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International Nuclear Congress

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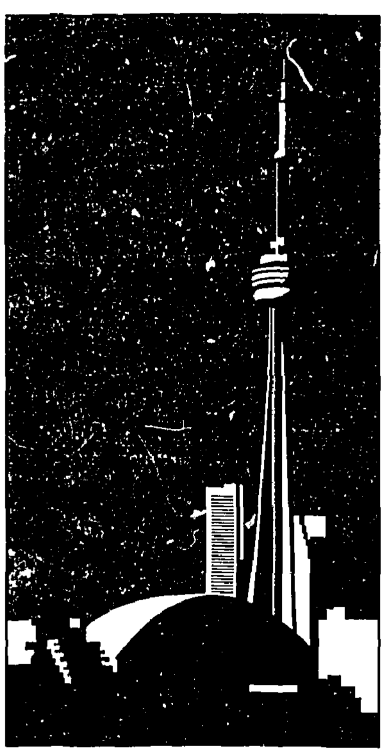
Technical Sessions Summaries



VOL 27 № 17

INTERNATIONAL NUCLEAR CONGRESS INC93

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INTERNATIONAL NUCLEAR CONGRESS 93

1993 OCTOBER 3-6
TORONTO, ONTARIO
CANADA

TECHNICAL SESSIONS

INVITED PAPER SUMMARIES

J. BOULTON

CHAIRMAN, TECHNICAL SESSIONS, INVITED PAPERS

Monday October 4

11:00 - 12:30 N1: Social Issues and Environmental Implications: Waste Management City Hall Room, 2nd Floor

Chaired by: Dr. T.E. Rummery, President, AECL Research

UK Perspective:
Mr. Michael Folger, Managing Director,
UK Nirex Ltd.

Japanese Perspective:
Mr. Noriaki Sasaki
Power Reactor and Nuclear Fuel Development Corporation

Canadian Perspective:
Dr Colin Allan, Vice President,
Environmental Sciences and Waste Management, AECL Research

US Perspective:
Mr. David Leroy, Former Negotiator,
Office of the US Nuclear Waste Negotiator

European Perspective:
Dr. Colette Lewiner, President & CEO SGN, France and
President ENS

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UK PERSPECTIVE ON SOCIAL ISSUES AND ENVIRONMENTAL IMPLICATIONS OF RADIOACTIVE WASTE MANAGEMENT

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United Kingdom

SUMMARY

This paper traces the history of radwaste disposal policy in the UK, and outlines the interaction between the industry which generates the waste, the government responsible for the national strategy for dealing with it and the public. It describes the formation of Nirex as the industry's vehicle for disposal with the clear remit also to win "a measure of support" for its proposals. The paper describes how lessons learned over the last decade have affected major decisions, by government and the Company, and have influenced our current approach.

In the United Kingdom the establishment of clear means for disposal of radwaste has been seen since the early 1970s as essential for the future development of the nuclear power industry. Since the early 1980s, attention has been focused on early development of a disposal route for intermediate level waste (ILW). Disposal of the small volume of high level heat-generating waste would be undertaken only after it had been vitrified and cooled in surface storage for at least fifty years. Since 1987, the national disposal strategy, defined by the government, has been that for all classes of intermediate level wastes, "early disposal in a deep facility is the right answer". Thus alongside a framework of regulation relating to safety and environmental aspects of disposal, the government has also set down the broad shape of the solution.

Nirex was formed in 1982 to co-ordinate the nuclear industry's responsibilities on radwaste disposal within the framework of government policy. Its initial attempts to develop a near surface disposal facility for the less problematic wastes did not come to fruition because of public and political pressures. This led to changes by government in the national disposal strategy and attention switched to the search for a deep multi-purpose facility. The consequent need to re-address the issue of siting allowed the principles of site selection to be re-examined.

In 1987 Nirex undertook a large scale public consultation exercise which had a major impact on its deep site selection exercise. In 1989, the Company announced that Dounreay in Scotland and Sellafield in North-West England were the first of twelve sites (selected because of the potential suitability of their geology) to be studied. In 1991, Sellafield was selected as the "preferred" site and investigations are now concentrated there. The paper describes how relations with the local community have progressed and describes how local concerns are being addressed.

A PERSPECTIVE ON THE MANAGEMENT OF RADIOACTIVE WASTES FROM NUCLEAR FUEL CYCLE IN JAPAN

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SUMMARY

Radioactive wastes such as low-level wastes (LLW) and high-level waste (HLW) from nuclear fuel cycle are conditioned, stored and finally disposed of. Basic idea of LLW disposal of the Japanese government is land disposal and sea dumping. In this context, the operation of shallow land burial of LLW from nuclear power stations was initiated in Rokkasho in 1992, while sea dumping is cautiously under investigation. The management policy of the HLW separated in a reprocessing plant is the solidification into a stable form, storage for the meantime and final disposal in a deep geological formation.

Recent topics of radioactive waste management in Japan, in addition to the operation of shallow land burial of LLW in Rokkasho, are as follows ; the cold operation of vitrification pilot plant(TVF) for Tokai Reprocessing Plant(TRP) has already started in Tokai, a commercial reprocessing plant with a capacity of 800 tons of spent fuel per year and an interim storage facility for HLW which is going to be returned from France and UK are under construction in Rokkasho, and an overall program for HLW has been submitted to the people from the government.

Geological disposal of HLW is on the stage of site-generic research and development, getting trust and understanding of the people on the geological disposal. To remove the mistrust of research site to be disposal site and to promote the research on the HLW disposal, it is considered highly important to provide a clear distinction between implementation of disposal and research and development as independent processes.

BUILDING CONFIDENCE IN DEEP GEOLOGICAL DISPOSAL OF NUCLEAR FUEL WASTE:

CANADA'S APPROACH

by

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Pinawa, Manitoba, Canada ROE 1L0

SUMMARY

At present used fuel from Canadian nuclear generating stations is safely stored at the station sites. Many years of experience have been accumulated with both pool storage and dry storage systems and supporting R&D indicates that these practices can be safely continued for many decades to come[1,2]. However, because the used fuel remains hazardous for thousands of years, it has long been recognized that such storage systems are not a permanent solution and that a passively-safe method of management that does not rely on institutional controls needs to be developed.

Like other countries, Canada is basing its plans for disposal of nuclear fuel waste on deep geological disposal, in the Canadian case in stable plutonic rock of the Canadian Shield. The most convincing way to demonstrate the long-term behaviour of such a disposal system would be to build one and monitor its performance in the long-term. Since such a procedure is not practical, society is faced with making decisions regarding the acceptability and the safety of deep geological disposal in the presence of unavoidable uncertainty. Thus the challenge that faces society and those charged with responsibility for nuclear fuel waste management is how to develop sufficient confidence to permit decisions to be made.

Is there a basis for such decision-making in Canada? The answer, in the view of the author, is yes. This confidence is based on:

- the technical approach, the use of multiple barriers for redundancy and defence in depth;
- the adoption of an observational approach to site characterization and to disposal vault design, construction, operation and eventually closure;
- an approach to the project, which is based on ongoing review and incremental decision-making and which recognizes that, throughout, the process must be flexible and responsive and that decisions are not irrevocable; and
- active and effective involvement of the public in the process.

NIMBY AND THE NUCLEAR NEGOTIATOR IN THE UNITED STATES

David H. Leroy
Former United States Nuclear Waste Negotiator

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(208) 342-000

SUMMARY

Nobody wants nuclear waste. But in the task of finding a repository for it in the United States, we must stimulate public interest which will lead to learning and eventually acceptance.

The original wide ranging, science-based concept for nuclear waste facilities in the US failed and was replaced with a single site at Yucca Mountain in Nevada. That choice has resulted in a standoff between the State and the Union.

Meanwhile the creation of the Office of the US Nuclear Waste Negotiator has resulted in an approach to the 50 States and 565 Indian Nations for either a permanent repository or a temporary facility. No real interest has been shown in the former, but a number of States and Indian Nations have shown interest in the temporary facility.

In seeking a site it has become clear that perception is reality. While the hard science must continue to advance, there must be a parallel effort to make people comfortable with nuclear power.

WASTE MANAGEMENT : TECHNOLOGICAL CHALLENGE OR SOCIAL ISSUE ?

by Colette LEWINER
European Nuclear Society, President
SGN, Chairman of the Board and Chief Executive Officer
– Société Générale pour les Techniques Nouvelles –
78182 Saint-Quentin-en-Yvelines Cedex (FRANCE)

Competitiveness, cleanliness towards the environment and consequent waste management, are the essential reasons for the achievements of nuclear energy in the world.

As regards the principles for waste management, since the origin, nuclear operators have been responsible for the wastes generated and have adapted appropriate management for each category of wastes. Besides, waste management costs are included in the cost of the nuclear kWh and are of the order of a few per cent.

Nuclear energy is far in advance in comparison with many other industries as concerns waste management.

Appropriate and definite solutions for the back-end of the fuel cycle are essential to a secure nuclear programme. In this respect, two alternative policies respectively the open fuel cycle and the closed fuel cycle are available to utilities.

The open fuel cycle policy consists in simply storing for final disposal the used fuel which are High Level Waste whereas the closed fuel cycle policy consists in reprocessing used fuel and recycling fissile products. The reprocessing/recycling strategy which offers a sustainable and environmentally safe solution is shared by a number of Western European countries : Belgium, France, Germany, the Netherlands, Switzerland, the United Kingdom as well as the former Soviet Union in Eastern Europe. Sweden and Finland have made the option for final storage. Whereas reprocessing and recycling are an industrial reality, the direct storage is still in a study phase.

The key concept for a safe disposal of nuclear wastes lies in minimization which may be mainly obtained by : the reduction of the volume and/or, by the reduction of the waste package.

For Low Level Wastes surface storage facilities are built that should remain under survey for 300 years. For High Level Wastes which are small in quantity (1/10), reducing their volume is still a concern for the nuclear industy, especially by a more efficient and elaborate partitionning and important R and D programmes are pursued.

The principle of underground storages in bedrocks associated with multiple barrier protections has been retained by almost all Western European countries and equivalent organizations have been set up and national bodies in charge of waste management have developped closed relationship in shared experiences and constitute a real specified network.

The future development of nuclear power depends largely on the safe management of the spent fuel and radioactive wastes. As a matter of fact, waste management is not a technological challenge, it is more a social issue that can only be achieved through excellence in communication with the public.

Monday October 4

11:00 - 12:30 N2: Social Issues and Environmental Implications: Reactor Safety
Dufferin Room, 2nd Floor

Chaired by: The Hon. E. Gail de Planque, Commissioner, US Nuclear Regulatory Commission

Safety of Soviet-designed reactors:
Dr. Claus Berke, President, Foratom

Safety of Aging Reactors:
Dr. H.J.C. Kouts, Chairman, International Safety Advisory Group of the IAEA

Recent Development in Engineered or Inherent Safety:
Dr. Fritz Ruess, Executive Vice President and
Mr. Dominique Vignon, Executive Vice President,
Nuclear Power International

Comparative Risk Perspectives:
Dr. Bernard L. Cohen, Professor of Physics and Radiation Health,
University of Pittsburgh

THE SAFETY OF NUCLEAR REACTORS IN THE FORMER SOVIET UNION

DR CLAUS BERKE, PRESIDENT, EUROPEAN ATOMIC FORUM (FORATOM)
15 rue d'Egmont, 1050 Bruxelles, Belgium

Summary

The potential problems posed by the nuclear power stations of the Former Soviet Union (FSU) raise some complex political and philosophical difficulties. On the one hand, these stations play a critical role in securing electricity supplies in large parts of the ex-USSR : for instance 25 % of Ukrainian electricity, and 55 % of Lithuania's electricity comes form nuclear plants. On the other hand, the poor maintenance and management record at the plants - and, in the case of the RBMKs, the questionable design of the plants themselves - has raised some clear concerns about the safety implications of continuing to operate these plants, both for local and indeed for more distant populations.

What is clear is that - if the stations of the FSU are to continue to play their essential role, in sustaining energy supplies - some of them need urgent attention and remedial work: Dr Berke's talk reviews the current state of the nuclear park in the former Soviet Union, the safety issues raised by the various reactor types in use, the current assessments of the various stations' safety, by the international authorities which have considered the problem, and the necessary conditions for carrying out the required remedial studies and hardware improvements.

Dr Berke's talk majors on the last of these aspects, and on the various political, financial and institutional obstacles which currently stand in the way of these urgently needed programmes. The specific issues which he examines in this connection are :

Δ The difficulties posed by the multiplicity of donors involved in the potential assistance programmes, and by the commercial sensitivities which parts of the assistance programmes inevitably entail;

Δ The key role of the G-24 Secretariat, in co-ordinating these various programmes, and the technical difficulty of the Secretariat's task;

Δ The roles of TPEG, ENAC, the EFCC, CASSIOPEE and other international organisations and consortia;

Δ The current financial restrictions on grant-funding (which in practice will not enable any major hardware improvements to be carried out), and the financial and political difficulties of loan-funded programmes (especially where there is an element of "conditionality");

Δ Last, and very crucially, the lack of any adequate and properly understood Civil Liability regime in most FSU countries, which opens up a perceived risk to donors that they themselves could be held financially accountable, if there were a serious accident at a station where they were involved in remedial programmes.

Dr Berke's talk puts forward some views on how these difficult problems could be addressed, and on how Western industry could best be helped to play the constructive role which it would like to, helping bring about the much-needed improvements in the FSU stations.

THE SAFETY OF AGING REACTORS

Dr. Herbert J. C. Kouts
Defense Nuclear Facilities Safety Board
Washington, D. C. 20004 USA

SUMMARY

Attainment of safety with older reactors has two principal components. The first is upgrading of safety where necessary to meet current criteria, as shown by modern safety analysis. The second is avoidance of degradation from effects of chemical change, wear, and neutron irradiation. Methods of avoiding reduction of safety from all such causes are known and are found to be effective, so that long-term reduction in safety is not an insuperable barrier to continued use of the plants to generate electricity.

RECENT DEVELOPMENTS IN ENGINEERED OR INHERENT SAFETY

Dr.-Ing. Fritz Ruess, Executive Vice President and
Michel Watteau, Executive Vice President

Nuclear Power International

SUMMARY

The national development efforts for future PWR's in France and Germany are now being combined into the development of the "European Pressurized Water Reactor" (EPR) with the support of the French and German utilities, who intend to use this design by the end of this decade.

The EPR is an evolutionary and innovative development based on the latest units in operation or under construction in France and Germany.

The basic design target is to further reduce the probability of accidents in particular severe accidents leading to large releases of radioactivity.

The defence-in-depth has been enlarged, providing mitigative features for low pressure core melt scenarios.

The paper presents the organization and the time schedule foreseen for the future development work, summarizes the safety objectives and describes the technical solutions selected so far in cooperation between Nuclear Power International, Siemens and Framatome on the supplier side and EDF and member of German utilities as the first potential users of the new design.

REACTOR ACCIDENT RISKS IN PERSPECTIVE

Bernard L. Cohen
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University of Pittsburgh
Pittsburgh PA 15260

Summary

According to the PRAs, reactor accidents may be expected to cause about 4 premature deaths per year in the United States. By comparison, coal burning air pollution kills thousands of times that number, and oil and gas, through air pollution, explosions, fires, and asphyxiation, kill hundreds of times that number.

It is sometimes argued that deaths from air pollution are less important because they are undetectable, but the same is true for the cancer deaths from nuclear accidents. If we are interested only in detectable deaths, the reactor accident toll is only a few deaths per century. The worst accident considered in the Reactor Safety Study, expected once in 10 million years, causes 3500 detectable deaths, a number already equalled by an air pollution episode from coal burning (London-1952).

Since large consequence nuclear accidents are hypothetical (in LWRs), they should really be compared with hypothetical accidents from other energy sources. There are many examples of these that are very much worse than the worst reactor accidents that have been discussed.

The biggest problem in obtaining public understanding of reactor accident risks is that it is irrational to treat risk questions without considering probabilities, and the public does not understand probability. One solution is to adopt a cut-off probability, such as not considering events expected less than once in 500 years. Buildings, bridges, dams, etc are designed to withstand a once-in-500-years earthquake, tornado, flood, etc, and smaller probability events are not even considered.

My preferred approach is to convert all risks to loss of life expectancy, LLE. The LLE for U.S. energy sources is: reactor accidents- 0.01 days (≈15 minutes), coal - 13 days, oil - 5 days, gas - 3 days. But by far the most dangerous energy strategy is energy conservation, with an LLE well over 100 days. If over-zealous energy conservation substantially reduces our wealth, the consequences could be very much greater.

If one believes that effects of radiation are uncertain, it is useful to compare reactor accident risks with other radiation problems. Radon in homes provides a 3000 times larger radiation risk, but arouses far less concern.

Other risks that give us an LLE of 15 minutes are listed, such as an overweight person increasing his weight by 0.005 ounces, or driving an extra half mile per year.

A bar diagram is provided comparing a wide variety of risks in terms of the LLE they cause. It demonstrates that the nuclear risk is truly trivial.

Tuesday October 5

08:30 - 10:00 N3: Social Issues and Environmental Implications: Economics of Electrical Generation City Hall Room, 2nd Floor

Chaired by: Mr. D.S. Lawson, President, AECL CANDU

Global Perspective:
Dr. Robert J. Saunders, Chief, Energy Strategy, Management and Assessment Division, The World Bank

UK Perspective:
Mr. Michael R. Kirwan, Executive Director, Finance, Nuclear Electric plc

Korean Perspective:
Dr. KunMo Chung, President, Institute for Advanced Engineering

Taiwan Perspective:
Dr. Kuang Chi Liu, Senior Vice Chairman, Atomic Energy Council

Canadian Perspective:
Dr. R.W. Morrison, Director General, Energy Mines and Resources, Canada

SOCIAL AND ENVIRONMENTAL IMPLICATIONS - ECONOMICS OF ELECTRICAL GENERATION

Global Perspective

Robert J. Saunders
The World Bank
Washington, D.C.

SUMMARY

The major changes in electricity consumption over the next 20 years will be in non-OECD countries. Almost all of the total world increase in electricity generation will be in developing or Central and Eastern European (CEE) countries. Yearly growth rates of energy demand of between 5 and 7 percent are quite possible for sustained periods in the coming thirty years. Given this rapidly growing electricity demand, major financial constraints, and widespread environmental problems, developing and CEE countries are experiencing increased pressures to produce and consume electricity more efficiently.

Experience has shown that when power sector programs and projects appear technically sound but fail to deliver results, the reasons in many instances are conflicting social objectives, overall weak country institutions, lack of an adequate legal framework, damaging discretionary interventions by governments, uncertain and variable policy frameworks, and a closed command-and-control decision-making process without adequate checks and balances. While all of these constraints are significant, it can be argued that the fundamental underlying sectoral problem relates to undue government interference in those day-to-day organizational and operational matters which should be under utility control. Such interference has undermined the accountability of those responsible for day-to-day management functions. It has influenced procurement decisions, mitigated against least-cost fuel choice, resulted in an inability to raise power tariffs to meet revenue requirements, restricted utilities' access to foreign exchange, mandated low managerial and technical salaries that are tied to low civil service levels, and promoted excessive staffing and political patronage.

Today, these large capital investment requirements, ingrained power sector inefficiencies, and desperate financial circumstances of many developing and CEE country power utilities have generated pressures for fundamental change. New approaches revolve around a framework for addressing the sector's financial, regulatory, and institutional issues, and around such effective reform mechanisms as greater transparency and public accountability in governing power sector institutions. Various approaches are being discussed and adopted in developed, developing and CEE countries. Among these are regulatory change, organizational change, commercialization and subsequently corporatization, and increased private sector participation in the power sector.

In the context of this changing industrial structure in the power sector in many parts of the developed and developing world, it can be argued that nuclear power will have difficulty competing with other energy sources at least over the next 20 year period. Even in countries that contemplate the nuclear power option in their electric generating mix, prospects for nuclear power are limited unless governments, regulators, utilities, and the nuclear industry together take coordinated action. The future of the nuclear industry will also depend on its ability to resolve issues such as public confidence, investment risk, and public acceptance of waste disposal and decommissioning plans. If nuclear power is to retain or obtain a role in the generating mix of countries contemplating the nuclear option, then the nuclear power industry must work closely with the public as well as all other concerned players.

THE ECONOMICS OF NUCLEAR POWER IN THE UNITED KINGDOM

M R Kirwan, Executive Director Finance, Nuclear Electric plc

Barnett Way, Barnwood, Gloucester, GL4 7RS, England

SUMMARY

The United Kingdom's choice of gas cooled reactor designs for its initial nuclear programmes has not, in retrospect, yielded the economic benefits that were anticipated. Nevertheless, existing Magnox and AGR plants now provide the lowest cost electricity in economic terms on the UK network (excluding low cost resource limited hydropower), and it will pay the UK to keep them running as long as possible.

Nuclear Electric's Sizewell B plant, a 1200 MWe PWR, is nearing completion to time and within its sanctioned costs. Its completion, commissioning and operation are also economically rational and cheaper than other potential capacity additions to the UK power grid.

The future of nuclear power is being considered by a Government Review this Autumn. If a prompt restart were decided on, it would be based on a twin repeat of the Sizewell design at the Sizewell site. Costs for the station (Sizewell C) are soundly based on existing experience and contractors' quotations. These give confidence that the plants should produce electricity at under 3p/kWh, a price at which they are projected to be competitive with coal or gas combined cycle generation plants that could be commissioned at the turn of the Century.

Additionally, nuclear power offers the UK a cost effective means of reducing acid gas and carbon-dioxide emissions in line with the Government's environmental targets and international commitments.

THE KOREAN PERSPECTIVE ON ELECTRICAL GENERATION

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Institute for Advanced Engineering
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Seoul, Korea

SUMMARY

The Korean energy sector is facing major challenges in the coming years. Firstly, it has to meet the rapid increase of energy demand. The annual growth rate of the total energy consumption during the past five years('88-'92) has been 11.3% and the elasticity of energy to GNP in 1991 was 2.62%. In spite of a vigorous campaign for increasing energy conservation and shifting the economic structure toward energy-saving industries, the elctric power consumption is expected to grow at the rate of 9.4% per annum during the next five years. It is planned to construct 27 new power generation plants with a total generating capacity of 12 GWe during that period.

Secondly, because of the increasing pressure to reduce consumption of low efficiency fossil fuels, there is a rapid increase of the demands for gas and electricity. The Korean strategy is to build more nuclear power plants and gas-fired(mainly LNG) units. The long-term power expansion plan (1991-2006) calls for a generation mix of 36.3% for nuclear plants and 21.8% for LNG units. Although coal-fired units will take a share of 34.1% by 2006 many will be integrated-gasification-combined-cycle(IGCC) plants. Two power generation technologies, nuclear and IGCC, will be developed intensely under ambitious research and development programs. The Korean research institutes are developing advanced standardized models of nuclear power plants and gas-fired units.

Thirdly, the Korean energy sector is also facing a need for rationalization movement. In previous years, the priority emphasis has been to secure supplies for the increasing demand. In the coming years, however, the major effort will be to increase efficiency of energy consumption. The energy sector will be free from tight regulation and the market forces will be allowed to determine the demand-supply mechanism. The electricity rate base will be more oriented toward load management. It will also reflect social and environmental costs. With the gradual introduction of solar electricity generation, wind power, fuel cells and other new generation technologies, the renewable energy will supply about 3% of the total energy by 2001. For this objective, the Korean government is accelerating R & D programs on new and renewable energy sources.

Lastly, the overall Korean energy strategy calls for enhanced public understanding of the integrated system of economy-energy-environment. We will stress for the positive synergetic effects between the subsystems of the total complex. Public acceptance is the key for a smooth sustainable development of the Korean economy. The Korean energy planners recognize the importance of openness and rational reconciliation for the future.

SOCIAL, ENVIRONMENTAL AND ECONOMICAL ISSUES OF ELECTRICAL GENERATION IN TAIWAN

KUANG-CHI LIU, SENIOR VICE CHAIRMAN, ATOMIC ENERGY COUNCIL
TAIPEI, TAIWAN, REPUBLIC OF CHINA

SUMMARY

As a country enjoying one of the fast growing economies of the world in the last decade, the installed power generation capacity in Taiwan lagged behind its economical growth to a great extent during this period. The reserve generation capacity dropped to a record low of 4.8% in 1991 and 6.7% in 1992. As a result, power interruptions occurred fourteen times in 1991 and two times in 1992. The construction of new generating units, together with improvements in energy management and energy conservation are among the top priorities facing Taiwan today in seeking for sustained future economical growth.

To meet this challenge, the government-owned utility developed a ten year development plan aiming at providing sufficient, reliable and low-cost electricity islandwide. The program called for a best power generation mix among hydro, coal, oil, gas and nuclear power, considering cost, environmental and social issues. Upon completion, the predicted reserve capacity will reach a much healthier 20%.

In this paper, details of the program in the aspects of cost, environmental and social implications will be discussed. Results from a multi-objective programming study by a local university will be briefly presented. As 93% of the total energy supplies are imported from abroad, the semi-indigenous energy source of nuclear power is particularly important to Taiwan in terms of long-term energy security and cost stability. However, social issues involved with the development of new nuclear stations have, to some extent, encumbered these advantages. Special discussion on these social issues on nuclear energy will also be presented.

COMPARATIVE COSTS OF ELECTRICITY GENERATION: A CANADIAN PERSPECTIVE

Dr. Robert W. Morrison
Director General, Electricity Branch
Department of Natural Resources Canada
580 Booth Street, Ottawa, Ontario, Canada K1A OE4

SUMMARY

The comparative cost of generating electricity from different energy sources and technologies is a key factor in electric utility decisions on capacity additions. As they struggle with declining load growth and concerns about the cost of electricity, utilities themselves are naturally keenly interested in the costs of the supply options available to them. Since electricity is so essential to the Canadian, and world economies, and utility activities have such far-reaching economic, social and environmental impacts, a much broader audience shares this interest in the economic analysis of generation costs.

This paper focuses on the internationally accepted Levelized Unit Energy Cost (LUEC) approach used to estimate the cost of various forms of generating electricity. It discusses the key factors impacting on costs of nuclear, coal and gas-fired electricity, presents some results from recent international and Canadian studies of generation costs and comments on the relation between such studies and the actual costs encountered in the real world.

A review of LUEC studies by the Nuclear Energy Agency of the OECD, the National Energy Board of Canada and Natural Resources Canada (NRCan), shows that there is consistency in the overall relative costs of electricity from coal, gas and nuclear power plants. In Canada, as in other parts of the world, no one fuel will be able to satisfy all regions and all circumstances. Nuclear power is competitive with coal and gas-fired generation in certain regions of Canada and of the world and will continue to be an appropriate and economic option. The studies also show that the comparative economics of nuclear, coal and gas-fired stations will depend on:

- the cost of fossil fuel and the impact of environmental policies on the LUEC of fossil plants;

- the future price and supply of natural gas and the ability of gas technologies to compete;

- the capital cost and performance of nuclear power reactors.

One can conclude, on the basis of these LUEC studies and actual cost results, that the challenge on costs for the nuclear industry in Canada and elsewhere over the next decade will be to operate existing nuclear plants consistently at high capacity factors, and to build new plants on schedule and on budget (having first gained acceptance from the public and orders from the utilities), and operate them at high capacity factors as well.

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Tuesday October 5

10:30 - 12:00 N4: Clearing the Public Acceptance Hurdle
City Hall Room, 2nd Floor

Chaired by: Dr. Ann S. Bisconti, Vice President, Research and Program Evaluation, US Council for Energy Awareness

Making the Case for Nuclear Power:
Dr. John H. Gittus, Director General,
British Nuclear Forum

A View from Europe:
Dr. P. Feuz, Secretary General,
European Nuclear Society/Executive Director, NucNet

An Analysis of Public Enquiries:
Mr. J.A.L. Robertson, Consultant

Evolution of Public Attitudes:
Mr. Gene Pokorny, Chairman,
Cambridge Reports/Research International

The Role of the Media:
Mr. Llewellyn King, President, King Publishing
(summary not available)

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"MAKING THE CASE FOR NUCLEAR POWER - ATTITUDES IN THE COMMUNITY"

John H Gittus F.Eng., D.Sc.
Director-General, British Nuclear Industry Forum

22 Buckingham Gate
London SWIE 6LB

SUMMARY

The nuclear industry gives a great deal of attention to public opinion about nuclear power and the industry itself. Public opinion surveys record a high level of acceptance that nuclear power will be necessary as part of a balanced energy supply pattern, and the public does not place nuclear waste or other perceived pollution problems high on their list of unprompted concerns. There is a marked dichotomy between these figures for acceptability and the favourability with which nuclear power is regarded, which has been adversely affected by accidents abroad and more recently by the surge of public sympathy for the coal mining industry.

Opinion surveys among members of key groups of opinion formers show that greater knowledge of energy and nuclear matters improves their perception of nuclear power. Representatives of these groups accept that the nuclear industry has a good record on operating safety. They believe that it is able to manage its waste successfully and expect that the waste can ultimately be disposed of. They also acknowledge that nuclear power is an important counter to global warming. However they need convincing that the cost of nuclear power stations can be justified at a time when there are adequate supplies of fossil fuel.

Finally a survey has been made to see who the public trust to make difficult decisions about science and technology. Scientists came top and politicians were the least trusted.

Given that scientists are more trusted than others on difficult issues such as nuclear power, they can claim the lead in helping the public to understand those issues. However they also have a responsiblility - to ensure that they communicate with the public on difficult issues in a way that the public can understand. We must all therefore look for ways of communicating our ideas clearly and simply so that the public can understand what underlies the difficult issues.

A VIEW FROM EUROPE

Dr. Peter Feuz, Secretary General, European Nuclear Society/Executive Director, NucNet
P.O. Box 5032, CH-3001 Berne

At first glance everything looks fine in Europe. Europe had 217 power reactors operating at the end of 1992. 12 of its nations nuclear generated more than 30% of their electricity - a figure which also applies to the European Community overall.

At second glance the situation is dismal. Very soon - except for France - additional nuclear capacity will come on line only in the central and eastern part of the continent.

There are many reasons for nuclear's stagnation. Fossil fuels are still cheap and abundant. Licensing procedures are unpredictable and make it too risky for many utilities to order more nuclear plants. But the main obstacles are politics and public opinion. It is difficult to say whether public opinion is bad because of lack of political leadership, or whether this leadership is lacking because politicians do not want to go against public opinion.

Nuclear has even become a party political issue where, for instance, Socialists fear to lose votes to the Greens. On the other hand, there are good examples in Finland and Sweden where the labor unions are important supporters of nuclear energy.

Unfortunately, in many countries not even the leaders of the nuclear industry support the cause vigorously enough. Many find it hard to look beyond the magic year 2000.

Public opinion polls show the situation as better than it really is. This illusion is because the pollsters' questions are often designed to obtain good answers. If we ask simply "Are you in favor of building additional nuclear plants?", there is nowhere a majority "yes".

Nuclear opponents, led by Greenpeace, have succeeded much more than the nuclear community has in making an impact on the public.

Nuclear is a scapegoat and has gone out of fashion.

In France, nuclear is a success thanks to leadership and excellent information programs. Different types of organizations in several countries are also doing outstanding information work and have achieved at least partial successes.

However, a real turnaround will be brought about only when information efforts are coordinated throughout Europe. ENS tries to contribute and runs a number of pan-European programs, and even worldwide ventures such as NucNet.

ADDRESSING PUBLIC CONCERNS: THE ROLE OF PAST INQUIRIES

J.A.L. Robertson
P.O. Box 2047, Deep River, Ontario
K0J 1P0

SUMMARY

Virtually all aspects of nuclear energy have been the subject of more than 30 national and international inquiries. The resulting reports make recommendations to improve the current situation, often with regard to maintaining or improving safety; some conclude that nuclear energy is not yet needed in their jurisdictions; and one recommends a moratorium on new nuclear power plants in Canada pending agreement on waste disposal. However, the most striking observation is that none rejects nuclear energy, when needed and under stipulated conditions.

Despite the endorsement of the nuclear industry's position provided by these inquiries, public acceptance has not been achieved. This is partly because very few people -- even in the nuclear industry -- are aware of the inquiries' findings, but more is involved. Study of the inquiries suggests that the industry and the public are talking past each other: While the industry concentrates on technical questions, the public is more concerned with ones broadly categorized as "ethics". These have been addressed by some inquiries, but much less thoroughly than the technical questions.

The irony of the current situation is that while members of the industry believe that for many applications nuclear energy is the energy source of choice, they fail to challenge the public conception that nuclear energy is a necessary evil. The obvious lesson to be learned is that the industry should devote more attention to what concerns the public. A recent publication presents the ethical case for nuclear energy from a proponent's angle.

The inquiries can be useful both in identifying the ethical issues, and in addressing the public's lack of trust in the industry and its regulator. Instead of saying "Trust us", the industry should be saying "Don't trust us -- check for yourself what independent inquiries have found".

Relevant Publications

"Nuclear Energy Inquiries: National and International", J.A.L. Robertson, Report AECL-10768, Published by AECL Research, Chalk River, Ontario, K0J 1J0 (1993)

"The Geometry of Nuclear Energy: Getting the Right Angle on the Ethics", J.A.L. Robertson, Canadian Nuclear Society Bulletin, Vol. 13, No. 3 (Fall 1992)

THE EVOLUTION OF PUBLIC ATTITUDES

LESSONS LEARNED FROM A REVIEW OF OPINION RESEARCH ON NUCLEAR ENERGY

Gene Pokorny
Chairman, Cambridge Reports/Research International
955 Massachusetts Avenue
Cambridge, MA, USA 02139

SUMMARY

After an examination of much of the public opinion research that has been conducted on the issue of nuclear energy over the last few decades, what strikes the author is not what has changed—or evolved to a higher or better state—but rather, how much has stayed fundamentally the same over the years of the nuclear debate.

The paper discusses three critical underlying patterns that exist in the data on public attitudes toward nuclear energy. Pointing out, where appropriate, what has changed, but also what has essentially remained constant over time. In addition after presenting each pattern the paper discusses some of its implications for those in the nuclear industry who are striving to "clear the public acceptance hurdle."

The patterns are:

Pattern #1

Since the advent of nuclear energy, public perceptions about the technology have been a mixture of both positive and negative views. Whether or not, in the final analysis, the public accepts the use of nuclear energy in general or favors specific nuclear facilities in their locality is determined by the relative weightings at any moment they give to these various positive and negative views and beliefs.

Pattern #2

Public perceptions of the changing benefits and costs of nuclear energy technology have been dramatically influenced over the last forty years by developments and trends in the broader society and culture, many seemingly unrelated to nuclear energy itself.

Pattern #3

The public consistently finds it easier to perceive the benefits in general of nuclear energy to the society at large than it does to perceive the benefits of specific nuclear facilities proposed in their localities.

Only by understanding these patterns and accepting their implications for nuclear communications and marketing can the nuclear industry hope to clear the public acceptance hurdle in the years ahead.

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INTERNATIONAL NUCLEAR CONGRESS 93

1993 OCTOBER 3-6
TORONTO, ONTARIO
CANADA

CONTRIBUTED-PAPER SUMMARIES

EDITOR: B. ROUBEN (AECL CANDU)

Acknowledgements:

The Editor would like to thank all reviewers on the Contributed-Paper Review Committee (see next page) for their help in reading and providing comments on the contributed abstracts.

The Editor is also very grateful to Cheryl Gaver and David Kidston for their invaluable assistance in assembling these Proceedings.

INTERNATIONAL NUCLEAR CONGRESS 93

1993 OCTOBER 3-6
TORONTO, ONTARIO
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CANADIAN NUCLEAR SOCIETY
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C1.2 *Radiological Impact of Fossil-Fired Stations in Ontario*
by A. Khan, S. Russell and H. Leung (Ontario Hydro)

C1.3 *Measurement of Neutron Radiation Exposure of Commercial Airline Pilots Using Bubble Detectors*
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C1.4 *Assessment of Operational Release Limits for CANDU-600 Nuclear Generating Station in Cernavoda, Romania*
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C2.2 *Options for Passive Containment Cooling in Next-Generation Nuclear Plant Designs*
by J. Woodcock, T.P. O'Donnell, J.A. Gresham (Westinghouse Electric Corporation)

C2.3 *Effectiveness of External Cooling and Associated Studies on Westinghouse AP600 Passive Plant*
by M.E. Wills and D.L. Paulsen (Westinghouse Electric Corporation), V. Notini and G. Invernali (Ansaldo)

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C3.2 *Safety Reassessment of the Hungarian NPP (The AGNES Project)*
by J. Gadó, L. Maróti, and I. Vidovszky (KFKI Atomic Energy Research Institute, Budapest, Hungary), J. Bajsz, A. Cserháti, J. Elter and S. Mikó (Paks Nuclear Power Plant Co.), E. Holló and Z. Téchy (VEIKI Institute for Electric Power Research) and K. Kovács (ERÖTERV Engineering and Contractor Co.)

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C4.2 *Decommissioning of Fort St. Vrain Nuclear Generating Station*
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C4.3 *Darlington Nuclear Generating Station Low Level Radioactive Waste Management*
by J. Hudson (Ontario Hydro)

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by J.E. Bergallo and G.N. Barceló (Comisión Nacional de Energía Atómica, Argentina)

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Chaired by: E. Price (AECL CANDU)

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by G.G. Elder (Westinghouse Electric Corporation)

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C9.3 *Prediction of Long-Term Fretting Wear Behaviour of Steam Generator Tubes*
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C10.2 *On-Line Reactor Building Integrity Testing at Gentilly-2*
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C12.2 *Challenges to Computing Buoyancy-Driven Flows in the Containment System of LWRs*
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by D.B. Sanderson, R.G. Moyer, D.G. Litke and H.E. Rosinger (AECL Research, WL), and S. Girgis (AECL CANDU)

C22.2 *Simulation of the Pressure-Tube Circumferential Temperature Distribution Experiments (Variable Make-Up Water Experiments)*
by M.H. Bayoumi and P.S. Kundurpi (Ontario Hydro), and W.C. Muir (IDEA Research International)

C22.3 *Liquid Relief Valve Failure Simulation in the Embalse Nuclear Power Station*
by S. Gersberg, J.R. Lorenzetti, D. Parkansky and J. Batistic (Comisión Nacional de Energía Atómica, Argentina)

C22.4 *Simulation and Analysis of a Main Steam Line Transient with Isolation Valves Closure and Subsequent Pipe Break*
by V. Stevanović and M. Studović (University of Belgrade, Yugoslavia) and A. Bratić (Thermal Power Plant Nikola Tesla-A, Yugoslavia)

Contributed-Paper Program

WEDNESDAY OCTOBER 6

10:30-12:00 Session C23: Physics 2
Huron Room
Chaired by: P. Akhtar (Atomic Energy Control Board)

C23.1 *Flux Mapping Theory Application for Channel Power Prediction*
by D. Brissette and M. Beaudet (Hydro-Québec, Gentilly)

C23.2 *A Review of the History-Based Local-Parameter Methodology for Simulating CANDU Reactor Cores*
by B. Rouben and D.A. Jenkins (AECL CANDU)

C23.3 *Validating the History-Based Diffusion Methodology for Core Tracking Using In-Core Detectors*
by A.C. Mao, B. Rouben and D.A. Jenkins (AECL CANDU), and E. Young and C. Newman (New Brunswick Power)

C23.4 *On-Line Heat Deposition Rate Measurements with a Quasi-Adiabatic Graphite Calorimeter in a Fusion Environment and Comparison with Calculations*
by O.P. Joneja and J.-P. Schneeberger (École Polytechnique Fédérale de Lausanne, Switzerland), R.P. Anand (Bhabha Atomic Research Centre, Bombay, India) and T. Buchillier (Institute of Applied Radiophysics, Lausanne, Switzerland)

10:30-12:00 Session C24: Life Extension
Kenora Room
Chaired by: K. Talbot (Ontario Hydro)

C24.1 *Planning the Retubing of a CANDU 6 Reactor*
by N.G. Craik (Canatom Inc.) and R. Baker (New Brunswick Power)

C24.2 *Improvements in Remote Removal Techniques for Active Components during Large-Scale Retubing of CANDU Reactors*
by W.J. Knowles (GE Canada Inc.)

C24.3 *Developments in Orbiting Tools for Refurbishment of CANDU Fuel Channel Components*
by M. Pollock (Spectrum Engineering Corporation Ltd.)

C24.4 *CANDU Single-Fuel-Channel Replacement Reducing Time and Radiation Exposure*
by T.A. Hunter and D.R. Pollock (GE Canada Inc.)

Contributed-Paper Program

WEDNESDAY OCTOBER 6

10:30-12:00 Session C25: Waste Management 2
Kent Room
Chaired by: J. Graham (British Nuclear Fuels Ltd.)

C25.1 *Progress at AECL's Underground Research Laboratory*
by G.R. Simmons (AECL Research, WL)

C25.2 *The Role of Engineered Barriers in the Disposal of Nuclear Fuel Waste - The Canadian Perspective*
by K. Nuttall and L.H. Johnson (AECL Research, WL)

C25.3 *Sulfur Polymer Cement, A Solidification and Stabilization Agent for Radioactive and Hazardous Wastes*
by G.R. Darnell (INEL-EG&G Idaho, Inc.)

C25.4 *The Cigar Lake Analog Study: An International R&D Project*
by J.J. Cramer and F.P. Sargent (AECL Research, WL)

10:30-12:00 Session C26: Accelerators & Industrial Radiation 2
York Room
Chaired by: A. Stirling (AECL Accelerators)

C26.1 *Nuclear Data for Feasibility Assessment of HLW Transmutation*
by M.A. Lone, P.Y. Wong, W.J. Edwards, and R. Collins (AECL Research, CRL)

C26.2 *Neutron Activation Analysis of Mortars from Stone Houses Built in Canada During the French Régime*
by H.W. Bonin, C. Bordeleau, J.R.M. Boulé and M.A.T. Lapointe (Royal Military College of Canada)

C26.3 *Handled Gamma Quant Generator in Density Measurement*
by A. Mozelev (Small Scale Research & Production Company RADICAL, Dubna, Russia)

Contributed-Paper Program

WEDNESDAY OCTOBER 6

10:30-12:00 Session C27: Human Factors 2
Peel Room
Chaired by: E. Davies (AECL Research, CRL)

C27.1 *Technology-Assisted Training in the Nuclear Regulatory Environment*
by D.J. Martin (Atomic Energy Control Board, Canada)

C27.2 *Alarm Processing for Diagnosis Using a Holographic Neural Network (HNeT)*
by J.E. Smith, M. Schwarzblat, and J.G. Sutherland (AND America Ltd.)

C27.3 *The Design and Implementation of an Operator's Performance Support System*
by R.J. Wilson (EACS - Engineering and Computing Services), A.A. Bokhari, W.J. Garland, W.F.S. Poehlman, and C.W. Baetsen (McMaster University)

C27.4 *Training Program Evaluation: Current Regulatory Activities*
by D. Tennant and R. Droll (Atomic Energy Control Board, Canada)

10:30-12:00 Session C28: Safety 5
Wentworth Room
Chaired by: A. Carnino (IAEA)

C28.1 *A Turbine Trip Transient Analysis with TRAC-BF1*
by J.L. François (Instituto de Investigaciones Eléctricas, Cuernavaca, México)

C28.2 *Should We Install a Software-Based Reactor Protection System?*
by G. Ives (Colenco Power Consulting Ltd., Baden, Switzerland)

C28.3 *Operating Under Fire the French Way*
by F. Bediou and J.P. Chatry (EDF-CIG)

C28.4 *Nuclear Plant Safety Enhancement by Early Identification of Slow Developing Abnormal Processes*
by V. Kotelenets, S. Korolev and M. Konovich (Yuzhnoukrainskaya NDP, Ukraine)

MONDAY OCTOBER 4

11:00-12:30 Session C1: Radiation and the Environment
Wentworth Room
Chaired by: R. Osborne (AECL Research, CRL)

C1.1 *Nuclear Electric Power Generation: An Evaluation of the Environmental Impact*
by R.W. Durante (Durante Associates)

C1.2 *Radiological Impact of Fossil-Fired Stations in Ontario*
by A. Khan, S. Russell and H. Leung (Ontario Hydro)

C1.3 *Measurement of Neutron Radiation Exposure of Commercial Airline Pilots Using Bubble Detectors*
by B.J. Lewis and R. Kosierb (Royal Military College of Canada), T. Cousins (Defence Research Establishment Ottawa), D. Hudson (Air Canada Flight Operations), and G. Guéry (Air France)

C1.4 *Assessment of Operational Release Limits for CANDU-600 Nuclear Generating Station in Cernavoda, Romania*
by D. Galeriu, N. Paunescu, N. Mocanu, R. Margineanu, and I. Apostoaie (Institute of Atomic Physics, Bucharest)

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SUMMARY

NUCLEAR ELECTRIC POWER GENERATION AN EVALUATION OF THE ENVIRONMENTAL IMPACT

By

Raymond W. Durante

Over the next 50 years, it is expected that world population will increase to somewhere between 9 and 10 billion people. In order to provide decent living standards for this population, adequate supplies of electricity will be needed. The choice of generation systems and the fuels used to provide the energy for this electricity production will have a significant impact on the environment. In fact, environmental considerations may emerge as the major decision making factor in the choice of electric power generating systems. For this 50 year period, there are basically five methods of generating electricity available to us: coal, oil, nuclear, gas, and hydroelectric. This paper examines the environmental impact of these systems to determine which has the least impact on the environment. The entire fuel cycle, beginning with the mining of the fuel, conversion, preparation, delivery, and storage and on to the plant operations, including land use, water use, noise, odor, and emissions, and finally the waste disposal factors are considered for each of the five systems. An attempt is made to apply a numerical value to the environmental impact and establish some ranking. While this may appear somewhat arbitrary, it is based on data and information already available. The results show nuclear power in a favorable light finishing second to natural gas fired plants. This assessment is made on the bases of environmental impact only. When other factors, such as long-term economics and availability of fuels are introduced, nuclear power will emerge as an even more attractive option.

In the United States and other parts of the world, some feel that a solution to the environmental problem is to simply not build anything. We know this cannot continue for any length of time and we face serious consequences if follow this path. We also know there will be an impact on the environment regardless of which system is selected. Therefore, it follows that the selection of energy strategies will coincide with the selection of environmental strategies and we must learn to quantify the real value of the environmental impact and factor it into on every decision.

RADIOLOGICAL IMPACT OF FOSSIL-FIRED STATIONS IN ONTARIO

Atika Khan, Sean Russell and Helen Leung

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Natural occurrence of radionuclides in fossil fuels makes emissions from fossil-fired stations a potential source of radiation exposure. This study provides an assessment of the radiological impact to humans in Ontario from fossil-fired stations (coal, oil and natural gas) using updated dose conversion factors. A 4000 MWe reference coal-fired station was chosen for assessing the radiological impact from coal. Similar sized reference oil-fired and gas-fired stations were chosen for assessing the impact from oil and gas, respectively. All stations were assumed to be located in Ontario. Actual emission data from a reference fossil-fired station were used in general, but where such data were unavailable, the most suitable data from the literature were used. Meteorological data from an Ontario coal-fired station site were used.

The study takes into account the dose to the maximum exposed individual from the entire fuel cycle, including the mining/extraction phase, the power generation phase and the waste disposal phase, for the fossil-fired stations. Comparison with a nuclear station was made for the power generation phase only, due to unavailability of data for the rest of the nuclear fuel cycle specific to Ontario. Available data for the mining phase and data from coal ash measurements carried out specifically for this study, were used in the present study. Public doses to members of the critical group beyond the boundary of the facility were evaluated using a detailed dose assessment model which included immersion in air, groundshine, beach sediment exposure, immersion in water, inhalation, and ingestion of soil, vegetables, fruits, animal produce, fish and water.

The pathway analysis for the power generation phase showed that an individual living near a 4000 MWe fossil-fired station with a 198 m emission stack could receive a maximum dose of 24 $\mu Sv \cdot a^{-1}$, which occurred for the coal-fired station. This corresponds to about 0.8% of the dose due to natural background radiation. When the same dose assessment model was applied to emissions from an equivalent size nuclear station with a 20 m emission stack, which is normal for the Canadian reactor designs, the projected public dose was estimated to be 120 $\mu Sv \cdot a^{-1}$, corresponding to about 4% of the dose due to natural background radiation. When comparison is made between model predictions and dose estimates based on environmental measurements, the pathway analysis model is conservative by at least a factor of 2.

For the waste disposal phase, measurements on a reference ash pile indicated a lower radon emanation rate compared to that from the soil, in general.

MEASUREMENT OF NEUTRON RADIATION EXPOSURE OF COMMERCIAL AIRLINE PILOTS USING BUBBLE DETECTORS

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SUMMARY

The International Commission on Radiological Protection has published new recommendations on radiation protection (ICRP-60). To reflect these new risk estimates, the regulatory agency in Canada, the Atomic Energy Control Board (AECB), has proposed to reduce the annual stochastic dose limit from 50 to 20 mSv for an atomic radiation worker, and from 5 to 1 mSv for the general public. These annual doses are expected to be comparable to those received by commercial air crew.

The measurement of the neutron component of the high-altitude radiation field produced by galactic cosmic rays has been difficult until the recent development of the BD-100R neutron bubble detector. In this study, these detectors were used by a total of 23 pilots from Air Canada and Air France over a period of one year. Each pilot carried one detector during their normal flight duties, providing a data base of about 600 flights. The present work yielded measurements of the neutron flux (0.8 to 3.3 n/cm^2.s), and the neutron dose equivalent rates based on an ICRP-60 recommendation (1.2 to 5.5 μSv/h). These measurements are in agreement with a previous study at the Ames Research Center using high-altitude aircraft and conventional neutron instrumentation. The total dose equivalents for the Air Canada flights are also consistent with predictions of the CARI code.

Considering that the neutron component contributes ~ 50% of the total dose equivalent, the total dose for a round trip for a Vancouver-Toronto route on the subsonic aircraft is 58 ± 13 μSv/trip. If the air crew member averages five such trips per month, this would yield an annual dose of 3.5 ± 0.8 mSv/y. The total dose for the air crew member would therefore exceed the proposed AECB limit for the general public.

Assessment of operational release limits for CANDU-600 Nuclear Generating Station in Cernavoda, Romania

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The nuclear energy option in Romania was taken in the 1970's by choosing the CANDU-600 type reactor for development.The first unit is planned to be completed at the end of 1994 in Cernavoda, Romania. Considering the real needs and resources of Romania, it is clear that nuclear energy is the only realistic solution and must be promoted safely. In the temperate climate of Romania, with high population density and agricultural land use in the near field, a generic approach using standard equilibrium models taken from Canadian or international literature cannot be used. Previous assessments done by the Romanian Nuclear Design Company and the Health Ministry have shown unrealisticaly high doses. For future licencing and operation, an environmental impact study done independently from the utility or designer is needed. Using the latest research results at various CANDU Nuclear Generating Stations and the experience acquired in Canadian laboratories, as well as a better knowledge of the Cernavoda site and the data base obtained in the preoperational survey, a full revision of the potential dose to the public was carried out. We reanalyzed the source term, the atmospheric and aquatic dispersion, reexamined the habits of the local population, and used recently obtained data for the the local transfer factors . We also ensured a realistic assessment of risk from tritiated water release by using the most recently quality factor, adding a human metabolic model which consider organically bound tritium, an improved model for plant and animal produce contamination, etc. The operational release limits were established for a maximum dose to the most exposed individual (MEI) of 0.3 mSv/a, a value lower than the usual 1 mSv/a limit, in order to follow the European trend and also to allow for the contribution to dose of unplanned incidents. Our results show that HTO and C-14 are the main contributors to the public dose, accommodating more than 95 % of the dose and that the ingestion pathways following atmospheric release contribute more than 80 % of the dose to the MEI. For the assumed source term, the operational release limits were found to be 3-10 times higher than routine release data obtained from actual CANDU-600 performance and extrapolating to plant maturity and 3 GWe/a (the planned full power for the Cernavoda site). We also analyzed the uncertainty in the assessment due to the changing habits in a society that is being transformed to a market economy. Recognizing the difference in the safety culture between Canada and Romania, we analyzed also the potential for increased routine (or incidental) emissions, due to inappropriate design, execution or operation(composition of annulus gas, quality of components, maintenance schedule, etc). Tritium releases are normally highest during station maintenance.We examined the effects of the maintenance scheduling on crop uptake, and consequent possibility of decreasing the MEI and local collective dose, by optimizing the maintenace schedule.

MONDAY OCTOBER 4

11:00-12:30 Session C2: Passive Safety
Kenora Room
Chaired by: Y.W. Na (KAERI)

C2.1 *Passive Emergency Heat Rejection Concepts for CANDU Reactors*
by N.J. Spinks (AECL Research, CRL)

C2.2 *Options for Passive Containment Cooling in Next-Generation Nuclear Plant Designs*
by J. Woodcock, T.P. O'Donnell, J.A. Gresham (Westinghouse Electric Corporation)

C2.3 *Effectiveness of External Cooling and Associated Studies on Westinghouse AP600 Passive Plant*
by M.E. Wills and D.L. Paulsen (Westinghouse Electric Corporation), V. Notini and G. Invernali (Ansaldo)

C2.4 *Considerations to Improve Decay Heat Removal by Natural Circulation under Accident Scenarios for Gentilly-2 Nuclear Generating Station*
by H.M. Huynh (Hydro-Québec), and J.-C. Amrouni and C. Hasnaoui (Énergie & analyses Énaq du Québec Limitée)

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PASSIVE EMERGENCY HEAT REJECTION CONCEPTS FOR CANDU REACTORS

by

N.J. SPINKS

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SUMMARY

A study is in progress at AECL to assess the safety and capital cost implications of a more extensive use of passive design features in CANDU reactors. The study is focussed on emergency heat rejection and applies passive design principles to enhance the independence of core cooling via the moderator as distinct from core cooling via the emergency coolant injection system. Moderator heat rejection is effected by heat transfer from a heavy water natural circulation loop to a light water natural circulation loop which includes a water jacket formed in part by the cylindrical wall of a steel containment vessel. The water jacket acts as an interim sink for heat from both the moderator and containment and ultimately transfers its heat to the outside air. As with current CANDU designs, depressurisation is effected via the steam generators which are then supplied with emergency water by gravity from an overhead tank.

The concepts have been applied to a 2 loop CANDU 6 plant and a preliminary assessment has been completed, using simplified (pseudo-steady-state) methods, of an in-core LOCA, the limiting accident for moderator heat rejection. It is found that some pressurization of the moderator is needed to maintain subcooling at the fuel channel elevation. By examining the dominant core-melt sequences for a conventional CANDU 6 design, core melt frequency appears to be improved by at least an order of magnitude. Capital cost seems to be competitive with conventional plant.

Further calculations are in progress to reassess moderator heat rejection using more sophisticated techniques. Core melt frequency will be assessed from first principles by identifying the dominant core-melt sequences for the passive design.

Design enhancements are described: improvements in the fuel channel design eliminate the need for subcooling of the moderator and in turn eliminate the need for moderator pumps even during normal operation.

OPTIONS FOR PASSIVE CONTAINMENT COOLING IN NEXT-GENERATION NUCLEAR PLANT DESIGNS

J. Woodcock, T.P. O'Donnell, J.A. Gresham

Westinghouse Electric Corporation, Nuclear and Advanced Technology Division,
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SUMMARY

Nuclear power plants in operation today utilize active safety-grade systems to obtain adequate heat removal from containments for postulated design basis or beyond design basis events. Reliable heat removal capability is typically obtained using redundant active systems and diverse mechanisms such as fan coolers or sprays to cool the containment atmosphere directly, while passive heat absorption by internal structures accounts for a small fraction of the early heat removal. Safety grade systems also require redundant support systems such as those which supply cooling water for pump seals and instrument air for valves. Although capable of providing for a safe response to postulated events, active systems add to plant construction costs and must be inspected and maintained throughout the life of the plant. Concepts have been proposed for new plant designs which provide heat removal from containment structures based primarily on passive principles. Advantages are -- the elimination of reliance on events such as pump start-up and valve actuation and a significant reduction in the number of safety-related components and systems in the plant. One promising concept is an extenal airflow path between the large steel containment shell and the concrete shield building (Figure 1) to provide a natural draft air flow. This concept is called a Passive Containment Cooling System (PCCS). Two particular PCCS designs have undergone detailed development. One design has been developed by an EBASCO team for use in a Heavy Water Reactor Facility (HWRF), and one design has been developed for the Westinghouse AP600 commercial plant. Analyses of the two PCCS designs have demonstrated that licensing or certification requirements can be met, demonstrating the flexibility and performance of the two designs.

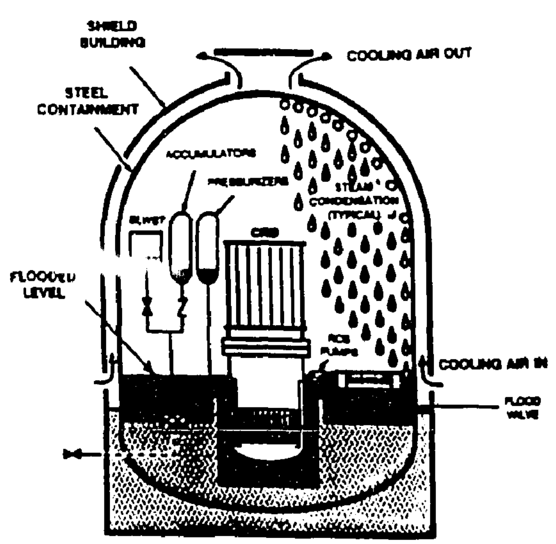

Figure 1 PCCS for Heavy Water Reactor Facility (HWRF)

EFFECTIVENESS OF EXTERNAL COOLING AND ASSOCIATED STUDIES ON W AP600 PASSIVE PLANT

Mark E. Wills - Westinghouse
David L. Paulsen - Westinghouse
Vinicio Notini - Ansaldo
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SUMMARY

In the AP600, the containment shell provides the ultimate barrier to the release of radioactive fission products in the event of either a design basis accident or even a severe accident. The containment is designed to be cooled by natural processes, in which heat is transferred from the interior to the exterior by conduction. Heat is then removed from the exterior of the shell both by radiation and free convection, and by the evaporation of cooling water applied to the external surface of containment. This paper focuses on the analyses performed to demonstrate the containment integrity for design basis events. In particular, the large break Loss-of-Coolant-Accident is examined because it provides a convenient means of evaluating the longer term performance of the passively cooled containment structure.

In order to obtain design certification, a Standardized Safety Analysis Report (SSAR) was prepared consistent with NUREG-0800, which documents analyses that demonstrate plant safety for design basis events. The WGOTHIC multi-purpose containment code was developed and used for the SSAR analysis, and is a Westinghouse enhanced version of the EPRI based GOTHIC code. The enhancements that Westinghouse made to the base code were to couple a mechanistic heat and mass transfer correlation for heat removal through the containment shell to the external nodes supported by the GOTHIC code.

Studies have shown that containment cooling is relatively insensitive to several parameters. In this study, the effects of varying the cooling water coverage fraction and also the time at which the shell receives cooling water have been examined. To assess the effect of each of these parameters, a 24 hour transient was examined, and the effect on the calculated containment pressure using the WGOTHIC code was obtained.

The most pronounced effect in the studies presented is that the time in which cooling water flow is initiated yields the largest deviation in containment peak pressure. There is a 5.2 psi difference in peak containment pressure between these cases, while the cases which examine the extremes in coverage area yield a peak pressure difference of only 2.6 psi. The peak shell heat removal rate does not vary significantly, with the exception of the reduced coverage case. The relative contributions of convection, radiation, and evaporation do, however, vary from case to case.

These studies provide further confirmation that the assumptions chosen for the SSAR analysis are realistic, but not overly conservative. Further conservatisms in the input model could be accomodated without violating safety criteria, but as these studies show, the use of nominal values in the safety analysis is appropriate, while maintaining adequate margin to design pressure. In summary, this study has demonstrated that the AP600 containment is a robust design, and is able to accomodate significant deviations in the ranges of system performance.

CONSIDERATIONS TO IMPROVE DECAY HEAT REMOVAL BY NATURAL CIRCULATION UNDER ACCIDENT SCENARIOS FOR GENTILLY 2 NGS

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Summary

A piping integrity study has been carried out to assess the integrity of the Primary Heat Transport System (PHTS) piping which could be subject to vibrations that may be caused by PHTS pumps operating under two-phase conditions after a postulated loss of coolant accident (LOCA). The study has shown that under accident scenarios, the PHTS pump trip is not required in the short term to preserve the Gentilly-2 PHTS piping integrity. Nevertheless, Hydro-Quebec has decided to proceed and study the implementation of an automatic PHTS pump trip system.

In the event of a loss of forced circulation of the coolant in the primary heat transport system, natural circulation of the coolant may be relied upon to transport the decay heat from the horizontal core to the boilers located above the core.

With the implementation of the Gentilly-2 automatic PHT pump trip system (reference 1), several additional accident scenarios will be added to the existing list of accident scenarios where the decay heat removal is relied upon through the effectiveness of thermosyphoning. In this paper, several considerations to improve the decay heat removal by natural circulation under accident conditions will be presented. The recommended and preferred modifications are discussed in details.

The modification of the loop isolation logic, which would enable the feed system to continue providing flow to both loops, the installation of two check valves between the feed pumps and the loops and the elimination of the thirty seconds delay on the steam generators crash cooldown are shown to provide a major improvement in the thermosyphoning capabilities of the intact loop.

MONDAY OCTOBER 4

11:00-12:30 Session C3: Safety 1
Kent Room
Chaired by: L. LeSage (Argonne National Laboratory)

C3.1 *The Distinctive Aspects of the Canadian Approach to Reactor Safety*
by F.C. Boyd (Wild & Boyd Management Advisors Ltd.)

C3.2 *Safety Reassessment of the Hungarian NPP (The AGNES Project)*
by J. Gadó, L. Maróti, and I. Vidovszky (KFKI Atomic Energy Research Institute, Budapest, Hungary), J. Bajsz, A. Cserháti, J. Elter and S. Mikó (Paks Nuclear Power Plant Co.), E. Holló and Z. Téchy (VEIKI Institute for Electric Power Research) and K. Kovács (ERÖTERV Engineering and Contractor Co.)

C3.3 *Some Aspects of Safety Characteristics of High Temperature Reactors*
by M. Šokčić-Kostić (Institute of Nuclear Sciences VINČA, Belgrade, Yugoslavia)

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THE DISTINCTIVE ASPECTS OF THE CANADIAN APPROACH TO REACTOR SAFETY

Frederick C. Boyd

Wild & Boyd Management Advisors Ltd.
Nepean, Ontario

SUMMARY

The origins of the Canadian approach to nuclear safety go back to the Montreal Laboratory during World War II and the design of the large, research reactor NRX, which began operation in 1947.

A serious "runaway" accident in NRX in December 1952 served as a catalyst for much of the reactor safety approach that still prevails. Siddall and Laurence both proposed a "risk" approach with a design target for a power reactor of 10(-5) serious accidents per year derived from a comparison with coal-fired electricity generating plants. This target was later lowered to 10(-6). Laurence, arguing that such a low probability could not be achieved, nor demonstrated with single systems, proposed separate and independent protective and containment provisions.

Subsequently, the concept evolved to one of a set of "special safety systems" including shut-down, emergency core cooling and containment,with each separate and independent of the "process" systems and of each other. They must have an unavailability of less than 10(-3) and be testable (during operation) to demonstrate it. Design dose values were set as performance criteria for "single" process system failures and "dual" failures of the process system and a "special safety system". Later, two separate, diverse, shutdown systems were stipulated to preclude runaway accidents. The CANDU designers introduced a "two-group" arrangement of systems to improve separation and developed "safety design matrices" to analyze potential cross-linkages.

Although the basic requirements for the special safety systems have been set out in regulatory documents the many subsidiary requirements and interpretations have not been documented making the Canadian safety approach very difficult for foreign purchasers of CANDU to implement. Also, in recent years the regulator increasingly is demanding apparently ad hoc requirements that deviate from the risk-based performance goals still nominally the basis for licensing. Ironically, this is occurring just when the USNRC - the author of much prescriptive regulation - is seriously examining risk-based performance criteria for regulation.

SAFETY REASSESSMENT OF THE HUNGARIAN NPP (THE AGNES PROJECT)

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SUMMARY

Paks Nuclear Power Plant is the only NPP in Hungary, it consists of four VVER-440/V-213 type units, which were put into operation in the years 1982, 1984, 1986 and 1987. The operating experience is fairly good, e.g. in 1990 two of the four units were among the top ten individual reactors in terms of cumulative load factors. During the ten years of operation no serious safety related problem occurred.

The main objective of the AGNES [1] (Advanced General and New Evaluation of Safety) Project for the reassessment of the safety of the Paks Nuclear Power Plant is to improve the safety culture of our nuclear technology. To ensure this the objectives are to be reached as follows.
- A report on the reassessment of the safety of the Paks Nuclear Power Plant has to be prepared, by using internationally acknowledged up-to-date techniques on the level of the nineties.
- The project should include the updating of design basis accident analyses, the performing of severe accident analyses and the preparation of a level 1 probabilistic safety analysis study.
- The project should help in determining the priorities for safety enhancement and backfitting measures and in identifying strategies for severe accident management.
- One of the objectives of the project should be the facilitating the preparation of a revised Safety Analysis Report, satisfying the requirements of the expected new Hungarian regulations.
The project has to be finished by publishing in 1994 a Final Report.

In the first period of the project's work the first version of the project's data base has been created. This data base contains NPP specific data and general VVER specific data. The analyses have started as well, some of them have been finished in the first period, most of them will be finished in the coming period.

The Final Report of the project will be prepared in two steps. In the first step a preliminary version [2] was prepared. This version describes the following topics: licensing of the Paks NPP, site description, basic design principles, description of the safety related systems, operating and safety instructions, operational experience, approved measures for safety enhancement, system analysis and description, analysis of design basis accidents, severe accident analysis, level 1 probabilistic safety analysis. The final version will deal with the same topics, with somewhat different content. The descriptive parts (e.g. site description, operational experience) can be considered as finally formulated ones in the preliminary version, parts describing the analyses will certainly contain more results in the final version, as the majority of the analyses has to be performed in the second part of the project's lifetime, i.e. in the coming year. The recommended measures for safety enhancement will be formulated only in the final version of the Final Report.

The present paper gives a short insight into the results of the analyses (system analysis and description, analysis of design basis accidents, severe accident analysis, probablistic safety analysis) performed.

SOME ASPECTS OF SAFETY CHARACTERISTICS OF HIGH TEMPERATURE REACTORS

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Nuclear Engineering Laboratory
POB 522, 11001 Belgrade, Yugoslavia

ABSTRACT

A numerical model for the description of water and air ingress accidents in the primary loop of the high temperature reactors is presented in this paper. The code TINTE developed in Institut fuer Reaktorentwicklung of Forschungszentrum Juelich, FRG (now Institut fuer Sicherheitsforschung und Reaktortechnik) is further improved and can be used for such an accident analysis. The thermofluid dynamics by different heat transport characteristics and the chemical processes between water, air and graphite on the basis of new experimental results are taken into consideration. The influence of the composition of cooling gas and graphite corrosion on the nuclear characteristics is also included. The developed computer code TINTE-C is tested on a fictious example of a water ingress accident at the AVR reactor.

MONDAY OCTOBER 4

11:00-12:30 Session C4: Operational Issues
Huron Room
Chaired by: A. Boothroyd (IAEA)

C4.1 *Projected Costs of Generating Electricity from Power Plants for Commissioning Around the Year 2000*
by M. Sakurada (OECD Nuclear Energy Agency)

C4.2 *Decommissioning of Fort St. Vrain Nuclear Generating Station*
by G.D. Schmalz (Public Service Company of Colorado)

C4.3 *Darlington Nuclear Generating Station Low Level Radioactive Waste Management*
by J. Hudson (Ontario Hydro)

C4.4 *QA in an R&D Environment*
by J.B. Hallett (AECL Research, CRL)

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PROJECTED COSTS OF GENERATING ELECTRICITY FROM POWER PLANTS FOR COMMISSIONING AROUND THE YEAR 2000

Michio SAKURADA, OECD Nuclear Energy Agency
Le Seine-Saint-Germain, 12, Boulevard des Iles, 92130 Issy-les-Moulineaux, FRANCE

Summary

An expert group of the OECD Nuclear Energy Agency (NEA) has updated a series of past studies on the projected costs of generating electricity. The main objective of the study is to review and explain the costs that would be expected for base load power generation technologies which could be commercially available for commissioning around the year 2000.

The study mainly focuses on generation costs of water-cooled nuclear plants (LWR and PHWR), coal-fired plants (pulverized coal combustion and atmospheric fluidized bed combustion) and gas-fired plants (combined cycle gas turbine systems). Generation cost data provided by 22 participating countries (including 6 non-OECD countries) have been analyzed on a common basis of standardised lifetime levelised cost methodology.

The series of reviews shows that the fossil-fired electricity generation costs, projected in constant money terms, have declined during the last decade, as a result of significantly lower fossil fuel prices than those expected in the past. On the other hand, the projected costs of nuclear generation have remained relatively stable over the same period. Thus nuclear power is not at present seen as having quite the same economic benefits as we thought in the early 1980s.

Despite this, at 5 % per annum (p.a.) real discount rate, using the reference performance assumptions, most participating countries project nuclear power to be the cheapest source of base load power from plants for commissioning around the year 2000.

The comparison is, however, sensitive in the majority of countries to the discount rates, to investment costs and plant performance of nuclear plants and to fossil fuel price expectations. At 10 % p.a. discount rate, the position of the less capital intensive technologies is greatly improved relative to the more capital intensive ones. As a result, only 5 of 14 countries providing nuclear and coal cost data project nuclear power to have a clear economic advantage over coal, and 3 of 9 countries providing nuclear and gas cost data project nuclear power to be cheaper option than gas combined cycle plants.

DECOMMISSIONING OF FORT ST. VRA!N NUCLEAR GENERATING STATION

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Overview of Fort St. Vrain Nuclear Generating Station

Fort St. Vrain is High Temperature Gas-Cooled Reactor. FSV was shutdown in August 1989. The decision to decommission FSV was based on technical and financial considerations. PSC has selected the DECON option. In November 1990, PSC submitted the Proposed Decommissioning Plan. The NRC issued in November 1992 the Decommissioning Order which established the requirements for carrying out decommissioning. FSV decommissioning commenced in August 1992 and is scheduled for completion in late 1995. FSV will then be repowered utilizing natural gas in a combined cycle.

Decommissioning Options, SAFSTOR vs. DECON

The purpose of decommissioning is to take a nuclear facility safely from service and to remove all radioactive materials to minimal levels of radiation so that the facility can be released for unrestricted use using one of three methods: DECON, SAFSTOR, or ENTOMB. In the case of FSV, only the DECON and SAFSTOR are considered practical.

In DECON, portions of the facility will be dismantled and removed or decontaminated to a level that permits the facility to be released for unrestricted use shortly after cessation of power operations. Three work elements exist with this option: decontamination and dismantlement of the Pre-stressed Concrete Reactor Vessel (PCRV); decontamination and dismantlement of contaminated or potentially contaminated BOP systems; and site cleanup and final site radiation survey.

SAFSTOR isolates the radioactively contaminated portions of the plant for long term storage and lay up. A portion of the work to place plant systems in a SAFSTOR condition is performed shortly after completion of defueling. The plant is then secured and enters into the SAFSTOR decay period, followed by additional work to complete the decontamination and dismantlement.

For either option, upon completion of the decontamination and dismantlement activities and the completion of the final surveys, termination of the license will be requested.

FSV PCRV Dismantlement

The major decommissioning task is the dismantlement and decontamination of the PCRV. Initial dismantlement includes the removal of selected internal components and removal of portions of the steam generators. The selected internal components are removed from the upper potion of the PCRV using the Fuel Handling Machine. The steam generator secondary assemblies are removed from the lower portion of the PCRV. The reactor cavity will be flooded with water to provide shielding for the workers. PCRV penetrations will be sealed, the Shield Water System will be connected, and the PCRV flooded.

To gain entry to the PCRV cavity, a plug of concrete will be removed from the top of the PCRV. The tophead plug will be cut into sections to allow them to be removed. The PCRV liner will be cut and removed. A work platform will be installed at the top of the PCRV. From this platform, workers will the remaining core components. The core barrel will be removed by cutting it into segments. The water level will be lowered removed in preparation of Core Support Floor removal. The CSF will be detached from the CSF columns and lifted with a hydraulic jacking system. The CSF will be sectioned into segments for disposal. Remaining radioactive components include the activated beltline concrete around the reactor core region and the PCRV liner. This section of PCRV sidewall will be removed by cutting and removing vertical segments.

DARLINGTON NUCLEAR GENERATING STATION
LOW LEVEL RADIOACTIVE WASTE MANAGEMENT

Janice Hudson
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Environmental Protection - Waste Management
DNGS Operations
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Traditionally, DNGS treated all waste produced in zone 3 areas as radioactive. This resulted in large volumes of materials being sent to the BNPD Radwaste Site for storage as low level waste which were in fact at least 75% inactive.

In October of 1992, DNGS implemented a waste management program entirely unique in Ontario Hydro which brought our low level radwaste production down dramatically, such that we are now 50% below the Nuclear Operations Branch target instead of being 200% above.

The DNGS Zone 3 waste processing systems consists of the following steps. All waste bags are barcoded and surveyed prior to delivery to the waste handling area. The barcoding system provides a complete history of all waste collected including pickup location, hazards, date and activity.

1. Each waste bag is checked for tritium (.5 MPC). Tritiated waste is dried in a fume hood for several days.

2. Bags free of tritium are placed in a waste bag monitor for determination of specific activity.

3. At this time waste with less than the alarm setpoint of 2000 nCi/kg is directed to a sorting table for opening. Waste with specific activity above the setpoint is treated as radioactive and is directed to a radwaste packing room.

4. We currently operate two sorting tables, one for likely clean and another for radioactive waste bags. The contents of the bags eligible for sorting are emptied onto the appropriate sorting tables. Each item is frisked piece by piece inside and out. Any materials displaying <u>no activity above background</u> are treated as inactive.

5. We will be installing ventilated sorting tables in the near future. This will allow us to sort waste bags up to 5 mrem/hr.

Our goal was to meet all required regulatory and internal targets with a simple, easy to maintain system with low startup costs. We have spend approximately $300,000 on equipment for this program which at the current waste production is saving Ontario Hydro $2.2 million over the next 5 years in avoided low level rad waste processing costs for DNGS.

QA IN AN R&D ENVIRONMENT

by

J.B. Hallett
Quality Assurance Coordinator
for
Engineering Technologies Division
AECL Research
Chalk River Laboratories

SUMMARY

Research and Development (R&D) organizations are adapting formal Quality Assurance (QA) programs designed to meet the requirements of manufacturing industries to the R&D work environment, which uses inspiration and innovation, combined with specialized knowledge, skills, equipment and facilities, to supply products that meet customers' requirements.

QA activities (planning and verification) provide assurance that work is carried out in an organized, controlled manner to meet customers' requirements profitably and the records show how this was achieved. In addition, they capture all opportunities to improve performance and identify new business opportunities (spin-offs). How much QA is applied is based on the consequences of failure (safety, costs and reputation) and the probability of that failure occurring, factoring in the experience of those doing the work.

A QA program is a documented management system for controlling and improving the quality of all activities related to conducting the business of the organization. The documents identify WHO is responsible for deciding WHAT (policies) has to be done, WHO approves the methods HOW (procedures), and WHO is responsible for verifying that the policies and procedures have been complied with. It provides assurance that:

- A clear and complete set of specifications reflecting customer requirements are developed (if this is not immediately possible then recognize that fact and plan accordingly).
- Clear, specific roles and responsibilities of the participants in the project, including the customer's, are described.
- Good project management practices are implemented.
- Good change-control and problem-solving procedures are in place and followed.
- Performance against customer requirements is measured and the findings acted upon.
- Potential spin-off opportunities are identified.

In R&D, the project quality plan bridges the gap between the program and the specific requirements for a project and its customer, and is proving to be the most effective way of providing the necessary flexibility required.

AECL Research recognizes that an effective QA program will be beneficial to its R&D activities. The pilot QA program implemented in the Engineering Technologies Division is aimed at providing quality products to customers in a cost-effective manner. The principles and practices developed in this program will be applied to all R&D activities in AECL Research.

TUESDAY OCTOBER 5

8:30-10:00 Session C5: Other Reactor Applications 1
Windsor East Room
Chaired by: P. Girouard (OECD Nuclear Energy)

C5.1 *Nuclear Energy, Environmental Problems and the Hydrogen Energy Economy*
by J. Rothstein (Ohio State University)

C5.2 *Issues Pertaining to Electrolytic Hydrogen Production Using Nuclear Power*
by E. Jelinski (Ontario Hydro) and J. Stephenson (Ontario Hydro, Retired)

C5.3 *Nuclear Hydrogen - Cogeneration and the Transitional Pathway to Sustainable Development*
by G.M. Gurbin (Integrated Energy Development Corp.) and K.H. Talbot (Ontario Hydro)

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NUCLEAR ENERGY, ENVIRONMENTAL PROBLEMS AND THE HYDROGEN ENERGY ECONOMY

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ABSTRACT

Hydrogen is an ideal fuel, its "ash" being pure water. It neither pollutes the environment nor contributes to global warming by producing greenhouse gases. Not occurring free in nature, it must be manufactured at an energy cost greater than the energy it provides. This is less serious than it sounds for it can be produced electrolytically from ocean or brackish water using renewables like wind and solar energy (or any other primary energy sources). Hydrogen is an excellent energy storage medium and transportable over large distances even more cheaply than electricity. Energy storage is mandatory for solar and wind energy or any primary energy source with temporal or geographical mismatches between patterns of power generation and patterns of use. Hydrogen can replace fossil fuels completely in many applications now, while in others, further development is required. Depletion of fossil fuel, as well as environmental factors, indicate the necessity of switching to hydrogen, and ammonia, made from it and often capable of replacing it. The magnitude of the infrastructure needed to manufacture the hydrogen needed to substitute for all fossil fuel probably requires capital investment at a rate beyond what a single generation would be willing to support, but environmental concerns and fossil fuel depletion may demand faster changeover. Nuclear power plants can be built fast enough to permit development of the hydrogen economy in time to prevent economic and environmental catastrophes. It is doubtful that any other approach can make this claim plausibly. Environmental fears of nuclear energy can be met by remote off-shore plant siting, inherently safe reactor designs, and waste storage at the plants until universally acceptable waste disposal methods are developed. Costs can be cushioned by synergetic production of oxygen, brine chemicals, deuterium and fresh water produced by multiple stage distillation of reactor cooling water, by in situ growth of hydrogen-based chemical industries, (e.g., ammonia), and by synergies between various industries.

ISSUES PERTAINING TO

ELECTROLYTIC HYDROGEN PRODUCTION

USING NUCLEAR POWER

by

Eric Jelinski; Technical Supervisor, Darlington Nuclear Generating Station, Ontario Hydro

John Stephenson; Retired, Health and Safety Division, Ontario Hydro

prepared for presentation at INC93
International Nuclear Congress, October 3, 1993
"Towards a better future"

SUMMARY

The timing and consequences of global warming resulting from the combustion of 'fossil' fuels are reviewed. The alarming consequences include melting of the polar icecaps and irreversible changes in climate and life as we see and experience it, that will take effect during our lifetime.

Use of the Index for Sustainable Economic Welfare (ISEW) is discussed and recommended for use because it enhances the value of hydrogen as a source of energy, relative to the energy produced by fossil fuels, which is subject to negative economic values associated with 'greenhouse' gas emissions, and other airborne pollutant emissions.

Analysis of "externalities" related to energy usage will result in identifying the "hidden" costs associated with the burning of fossil fuels, nuclear power, automobile and freight transportation. For example, the avoided emissions from an electric car are quoted at US $8000 to US $9000. In addition, the "externalities" associated with nuclear power, such as waste disposal, must also be considered.

When externalities are calculated and reported in the context of health and environmental costs avoided through the use of hydrogen technology, incentives for change may be realized. Incentives may be in the form of "Carbon Taxes", "BTU Taxes", and/or a form of a "health care tax" aimed at the harmful technology; in essence, providing a economic advantages for cleaner technologies using hydrogen.

The advantages of nuclear generating stations due to their near zero carbon dioxide emissions along with the base load generating characteristics of the nuclear stations, the daily variation of electrical demand in Ontario are discussed.

It is suggested that the un-used base load from nuclear electric stations be used to generate hydrogen gas which has application as a near zero 'greenhouse' gas energy source for road, air and space transportation; heating, and chemical feedstock applications. The current markets for hydrogen in Ontario, and recent developments in hydrogen production technologies are discussed along with energy costs, and methods of energy and environmental analysis.

The synergies between hydrogen and electricity such as the use of off-peak power to meet the Toronto Target for CO_2 reduction as well as avoided health and environmental impact due to the 'environmentally clean' properties of hydrogen are highlighted.

NUCLEAR HYDROGEN - COGENERATION AND THE TRANSITIONAL PATHWAY TO SUSTAINABLE DEVELOPMENT

By: Gary M. Gurbin - Integrated Energy Development Corp.
K.H. Talbot - Ontario Hydro

The atmospheric consequences of carbon and the evolution of world energy sources have resulted in a movement away from high carbon fuels, and a growing appreciation that the next generation of industrial development must be on a sustainable basis.

Although some legislation, such as the U.S. Clean Air Act, 1990, have resulted in a significant shift toward higher hydrogen and oxygen content transportation fuels, the net consequence can be negative to the global environment. The objective of sustainable development is clear, but the implementation remains elusive and lacking focus.

The Bruce Energy Centre has been evolving for nearly two decades, driven by a mission to commercially demonstrate the importance of integrating energy, the environment, and the economy in industrial development. The nearby Bruce Nuclear Generating Station "A" has provided process steam for operation of a fermentation alcohol plant, alfalfa processing plant and fullscale greenhouse.

The development of the next phase of the Energy Centre, in cooperation with Ontario Hydro, will see the introduction of a series of integrated energy processes whose end products will have environmental value added.

Cogenerated nuclear steam and electricity were selected on the basis of economics, sustainability, and "0" carbon emissions. The introduction of hydrogen to combine with CO_2 from alcohol fermentation provides synthetic methanol as a feedstock to refine into ether for the rapidly expanding gasoline fuel additive market. Large volumes of O_2 will enhance combustion processes and improve closed-looping of the systems.

Ammonia synthesis, municipal solid waste separation, cellulose conditioning, residual oil upgrading, aquafarming and additional greenhouse projects are in the planning stage.

In the implementation of the commercial development the first stage will require simultaneous electrolysis, methanol synthesis and additional fermentation capacity. Electricity and steam pricing will be key to viability and an 80 MW "back-up" fossil fuelled, back pressure turbine cogeneration facility could be introduced in a compatible manner.

Successful demonstration of transitional and integrating elements necessary to achieve sustainable development can serve as a model for electric utilities throughout the world.

TUESDAY OCTOBER 5

8:30-10:00 Session C6: Thermalhydraulics 1
Kenora Room
Chaired by: H.M. Huynh (Hydro-Québec)

C6.1 *NUCIRC Simulations of Asymmetries in CANDU6 Heat Transport Operating Conditions*
by M.R. Soulard and W.J. Hartmann (AECL CANDU), G. Hotte (Hydro-Québec), P.D. Thompson (New Brunswick Power), and P.L. Chang (Ontario Hydro)

C6.2 *One-Dimensional Model of Separated Two-Phase Flows*
by V. Stevanović and M. Studović (University of Belgrade, Yugoslavia)

C6.3 *Two-Dimensional Modelling of Fluid Flow in a CANDU-Type Header*
by D.J. Wallace and S. McIlwain (AECL Research, WL)

C6.4 *Moderator Flow Distribution in a Simulated Calandria Model*
by R. Nayak, P.K. Baburajan and K. Iyer (Indian Institute of Technology, Bombay)

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NUCIRC SIMULATIONS OF ASYMMETRIES IN CANDU6 HEAT TRANSPORT OPERATING CONDITIONS

by

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SUMMARY

The summary was not available at the time of printing.

ONE-DIMENSIONAL MODEL OF SEPARATED TWO-PHASE FLOWS

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A physical model and corresponding computer code have been developed for the prediction of two phase liquid and gas separated flows in horizontal and vertical pipes. These types of two-phase flows occurs in the components of the nuclear reactor coolant system during hypothetical loss-of-coolant accident, components of the chemical plants, etc. The model is based on one-dimensional mass, momentum and energy balance equations for three distinct phases: liquid film, gas phase and droplets entrained in the gas stream. In the case of a horizontal stratified flow, the liquid film momentum balance equation takes into account that the pressure drops are different in the liquid film and in the gas-droplets stream, because of the liquid film level change along the pipe. Corresponding closure laws are included in the model for the description of mass, momentum and energy transfer between the phases, and for the determination of interface surfaces between the fluids. The proposed system of ordinary differential equations is solved by the Runge-Kutta method. The modeling results have been verified against the experimental results available in the literature: flooding in the horizontal pipe, counter-current steam-subcooled water flow with the direct steam condensation, the conditions of the hydraulic jumps in the plane flow, and vertical annular flows with liquid entrainment. The numerical results have shown that the weak points in modeling these types of flows are the predictions of the direct steam condensation on the liquid film and liquid entrainment in the vertical annular flows.

A new simple correlation has been proposed for the entrainment in the vertical annular water-air or water-steam flow. The obtained results show that the liquid entrainment could be well correlated with the liquid film thickness, which is in correspondence with the surface wave behaviour:

$$We = F_1(\delta)^{F_2}\rho_F,$$

where are: We - rate of entrainment (kg/m2s), δ - liquid film thickness, ρ_F - density of liquid, and F_1 and F_2 are parameters which depend on the liquid film shape, wave amplitude, velocity and frequency.

Also, a new correlation has been proposed for the direct steam condensation on the liquid film. It is based on the turbulent convective heat transfer in the vicinity of the liquid film surface, and it is given in the form:

$$St = \frac{Re\theta}{I},$$

where $St = h/(\rho_F c_{pF} u^*)$ is Stanton number, $Re = \delta u^*/\nu_F$ is Reynolds number, u^* is friction velocity, $\theta = (T_{sat} - T_F)/(T_G - T_F)$ is nondimensional temperature, and I is integral expression derived from the energy balance of the turbulent boundary layer. The results are in good agreement with the experimental data.

TWO-DIMENSIONAL MODELLING OF FLUID FLOW IN A CANDU-TYPE HEADER

by

D.J. Wallace and S. McIlwain

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1993

SUMMARY

paper describes a preliminary study on the application of a two-dimensional prototype of the two-fluid, one-dimensional code, CATHENA. The study focuses on the use of a prototpye of CATHENA in investigating two-phase flow behaviour in a two-dimensional structure resembling a CANDU (CANada Deuterium Uranium) inlet header. Liquid levels and feeder flows obtained from CATHENA simulations of single- and double-turret injection of single-phase liquid and two-phase vapour/water are presented. A comparison of the calculated liquid levels with experimental data gives good agreement in the overall behaviour of the fluid in the header. The two-dimensional approach removes some of the empiricism built into the one-dimensional model, and as a result, increases the scope for understanding two-phase flow behaviour in CANDU headers under postulated upset conditions. Advantages and disadvantages of examining header two-phase flow behaviour in a multi-dimensional framework are discussed.

The work presented in this paper was funded by a CANDU Owner's Group (COG) agreement.

MODERATOR FLOW DISTRIBUTION IN A SIMULATED CALANDRIA MODEL

R.Nayak, P.K.Baburajan and K.Iyer*
Department of Mechanical Engineering
Indian Institute of Technology, Bombay 400 076, INDIA.

SUMMARY

The present work has been motivated by a specific system modification that was brought about by the failure of inlet flow distributor in the calandria of the Madras Atomic Power Station. The primary aim was to generate benchmark data using which the existing computer codes can be qualified.

The experimental set-up consisted of a cylindrical section 250 mm in diameter and 250 mm long simulating the calandria. Twenty one tubes distributed in a pitch to diameter ratio of 1.8 simulated the tube matrix. A centrifugal pump was used to pump in water from a storage tank into the test section through a rectangular inlet of 38 mm width and 225 mm length oriented at 45 to the vertical. Water flowed out of the model through a rectangular slot of identical dimensions as the inlet but directed vertically downwards. Suitable transition pieces were used to direct water from circular connecting lines of 37.5 mm in diameter to the rectangular inlet and outlet sections.

The flow of water through the test section was monitored using a calibrated orifice flow meter. The pressure drop across the test section was measured by a U tube manometer with carbon tetrachloride as manometric fluid. The velocity distribution in the set up was measured using a two colour Laser Doppler Velocimeter of DANTEC make employing a 5 Watt, COHERENT INNOVA 90 make argon-ion laser. To allow the laser beams scan the flow field, the end flanges were made of good quality perspex. A perspex window was also provided in the inlet section to monitor the inlet velocity.

Pressure drop and velocity measurements were made for water flow rates ranging from 6 l/s to 12 l/s. The results have been presented in the form of vector plots and stream line contours. The obtained results indicate that the flow is three dimensional and the normal assumption of isotropy of the tube resistance used in computer simulation would not be valid as water tends to follow the path of least resistance by aligning itself along certain favoured directions.

* To whom correspondence may be addressed.

TUESDAY OCTOBER 5

8:30-10:00 Session C7: Social Issues 1
Windsor West Room
Chaired by: F. De Galzain (OECD Nuclear Energy Agency)

C7.1 *Update on Canada's Fuel Waste Management Program: Preparing for the Environmental Review of the Concept*
by C.J. Allan, K.W. Dormuth and K. Nuttall (AECL Research, WL)

C7.2 *Research by British Nuclear Industry Forum into Public Support for Nuclear Power*
by N. Middlemiss (British Nuclear Industry Forum)

C7.3 *The Role of ANS in Enhancing Public Understanding of Advanced Nuclear Energy Plants*
by E.L. Quinn (MDM Engineering) and K.H. Turner (Dames and Moore)

C7.4 *Public Acceptance of Nuclear Energy in the Ukraine*
by N.N. Sappa (Kharkov Institute of Physics & Technology)

UPDATE ON CANADA'S FUEL WASTE MANAGEMENT PROGRAM:

PREPARING FOR THE ENVIRONMENTAL REVIEW OF THE CONCEPT

by

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Canada

SUMMARY

The Canadian Nuclear Fuel Waste Management Program (CNFWMP) was established in 1978 as a joint initiative by the governments of Canada and Ontario. Under the program, AECL is responsible for developing and assessing a concept to dispose of nuclear fuel wastes in plutonic rock of the Canadian Shield. Ontario Hydro has advanced the technologies for interim storage and transportation of used fuel.

The aim of the concept is to isolate the used fuel waste from the biosphere by a series of engineered and natural barriers. During the past fourteen years, AECL has carried out detailed studies on each component of this barrier system. A robust concept has been developed, with options for the choice of materials and designs for the different components.

The disposal concept is being reviewed under the Environmental Assessment and Review Process (EARP). AECL is the "Proponent" for this review, and will submit an Environmental Impact Statement (EIS) describing the disposal concept. The EIS has been written to respond to guidelines issued by the Environmental Assessment Panel responsible for carrying out the review. The future direction of the CNFWMP will depend on the recommendations of the Panel and on the resulting governmental decisions on the appropriate next steps.

If the concept review is completed by 1996, as currently expected, and the concept is approved, the many steps that would be involved with siting and construction of a disposal facility, mean that disposal would not begin before about 2025.

RESEARCH BY BRITISH NUCLEAR INDUSTRY FORUM INTO PUBLIC SUPPORT FOR NUCLEAR POWER

Nigel Middlemiss, British Nuclear Industry Forum
22 Buckingham Gate, London SW1E 6LB, U.K.

Summary

Attitudes in the community to nuclear power

a. The Public

i. The public *accepts* the need for nuclear electricity.

ii. In the context of its other day-to-day preoccupations there is little unprompted concern among the public about the nuclear industry.

iii. Fewer people are concerned about pollution from the nuclear industry than are concerned about atmospheric pollution, pollution of the rivers and seas and pollution from cars and from the chemical industry.

iv. Nuclear *electricity* has a better reputation than the nuclear *industry* as a cause of harm to the environment. Nuclear *electricity* is rated similarly to coal and oil generated electricity in this respect.

v. The nuclear industry and nuclear electricity are cited more frequently than the coal generated electricity industry as industries that are reducing the harm that they do to the environment.

vi. The more information the public has, the more *in favour* of the industry it is likely to be.

b. Senior decision makers and influencers ('experts')

vii. Virtually all experts accept the nuclear industry's credentials on operating safety, waste disposal and for being a counter to global warming.

viii. Experts require reassurance that the economics of the industry can be justified while there are adequate supplies of fossil fuels.

THE ROLE OF ANS IN ENHANCING PUBLIC UNDERSTANDING OF ADVANCED NUCLEAR ENERGY PLANTS

by E.L. Quinn
MDM Engineering

and

K.H. Turner
Dames and Moore

SUMMARY

The summary was not available at the time of printing.

PUBLIC ACCEPTANCE OF NUCLEAR ENERGY IN THE UKRAINE

by N.N. Sappa
Kharkov Institute of Physics & Technology
Kharkov, Ukraine

SUMMARY

The summary was not available at the time of printing.

TUESDAY OCTOBER 5

8:30-10:00 Session C8: Fuel & Fuel Cycles 1
Wentworth Room
Chaired by: I. Hastings (AECL Research, CRL)

C8.1 *Analysis of Advanced Fuel Cycles in Argentina*
by J.E. Bergallo and G.N. Barceló (Comisión Nacional de Energía Atómica, Argentina)

C8.2 *Advanced Fuel Cycle Options - Extended Burnup and Low Leakage Core Designs for Spent Fuel Volume Reduction*
by M.A. Feltus (Pennsylvania State University)

C8.3 *Application of Modern High Conversion Concepts to Pressure Tube Reactors with Breeding Capabilities*
by P.C. Florido, M.J. Abbate and A. Clausse (Comisión Nacional de Energía Atómica, Argentina)

C8.4 *Recovered Uranium in CANDU: A Strategic Opportunity*
by P.G. Boczar, J.D. Sullivan, and H. Hamilton (AECL Research, CRL), Y.O. Lee, C.J. Jeong and H.C. Suk (KAERI), and C. Mugnier (COGEMA)

ANALYSIS OF ADVANCED FUEL CYCLES FOR ARGENTINA

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SUMMARY

Our country has started the study of advanced fuel cycles for our nuclear power stations, all of then PHWR type, one of then CANDU 6, and two pressure vessel - KWU PHWR type.

In this work the review and analysis was focused on real impact in the fuel cycle economics for Argentinian case, because the technology for CANDU fuel fabrication, enrichment and reprocessing were developed in our country.

The fundamental condition was to use advanced fuel cycles that do not introduce any change in the nuclear power plant.

We focused this study over the current cycles proposed for PHWR reactors, including the use of slightly enriched uranium (SEU), and the strategy using plutonium recovered from PHWR or PWR type reactors.

In the case of the cycles with recovered plutonium, the possibilities analyzed were : spiking with Pu recycled from PHWR's only in channels in the external zone on the reactor core and the TANDEM fuel cycle whit uranium and plutonium provided by PWR reactors from Brazil in an optimum mixing with natural uranium.

This study was carried out with current models accepted to make this economical evaluations, developed for the special case of reactors with continuum refuelling.

The results show the possibility to reach very important economical improvements with several of nuclear fuel cycles analyzed, and an important reduction of natural uranium consumption per unit of generated electricity.

In the case of use of SEU the savings in natural uranium consumption will be around 25 %, and the economical savings will be around 25 % too, for enrichments levels lower than 1%.

In the case of recycled self generated plutonium as fuel cycle, uranium ore consumption will be reduce more than 50 %, but not economical savings can be achieved, with the actual expected cost for reprocessing, but in the case of TANDEM fuel cycle, it will be possible to obtain better economical impacts with the same ore consumption reduction.

It can be concluded that with the use of advanced fuel cycles the country can reach very important savings in the use of nuclear materials, and reduce the cost of actually generated electricity without any modification in our power plants.

ADVANCED FUEL CYCLE OPTIONS - EXTENDED BURNUP AND LOW LEAKAGE CORE DESIGNS FOR SPENT FUEL VOLUME REDUCTION

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This paper addresses the advantages and disadvantages of using very high fuel burnup, reinsertion, and low leakage designs in advanced fuel cycle light water reactor cores as a technique to reduce vessel fluence, and total volume of spent fuel discharged into the waste management stream. The results from the Penn State Fuel Management Package (PSFMP, i.e., LEOPARD, MCRAC, ADMARC, OPHAL computer codes) demonstrate how to attain practical high burnup core designs, that can be verified with design codes.

This study focuses on the practical use of such advanced fuel designs to: (a) achieve very high discharge burnups, (b) produce low leakage at the periphery, (c) have 18 or 24 month cycles, and (d) maintain safety margins, peak power levels, thermal-hydraulic limits, non-positive moderator temperature coefficients. Parametric studies that show the effects of batch enrichments, loading patterns, power distributions vs. burnable poisons (BPs), and batch loading options (3, 4, or split batches) are presented. Evaluations of practical and optimal extended burnup core designs, using the PSFMP, show that very high burnup core designs are not only attainable, but are most cost-effective and beneficial to the environment in terms of waste reduction.

The PSFMP expert system technique can be used with confidence to obtain a practical, efficient design. Zhian's optimization system, based on K_{inf} priority tables, BOC power distributions, using the Haling distribution for a first estimate to find BOC states, with constraint equations for discharge burnup, and on reducing cycle costs, can be extended to achieve higher exposures. This optimization technique uses a priority table to load fuel by ranking K_{inf} and core positions, determined by their neutron importance, and then modifying the loading pattern to reduce hot spots. Both fuel shuffling and split batch enrichments are used to obtain the best practical fuel design. The minimum amount of BPs (discrete rods or IFBAs) are found in the practical core design by using the Haling distribution to find the optimal core loading, and to minimize power peaks during depletion.

Although the optimum design is very sensitive to constraints imposed, all of the core patterns follow the reference core's priority table closely and produce similar optimum design results. Using information provided by position vs. K_{inf} priority tables and existing core patterns, it is possible to find practical designs by making make small adjustments. A LEOPARD cell depletion and one-dimensional diffusion analysis was used to approximate regional enrichments, core lifetimes, normalized power levels, and simple cost comparisons. Then, an expert system optimization found practical core loadings for a high burnup design, In general, a split batch approach with IFBAs proves to be the most effective design, in terms of low leakage, high burnup, and usage of low burnup assemblies in the core central location and strategic locations.

APPLICATION OF MODERN HIGH CONVERSION CONCEPTS TO PRESSURE TUBE REACTORS WITH BREEDING CAPABILITIES

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SUMMARY

The expected delay on the commercial availability of fast breeder reactors and the increment of the Plutonium amount stored have up-dated the interest on a better utilization of nuclear fuels using the current existing technologies. One of the most attractive options are the new designs of high conversion reactors in combination with advanced fuel cycles.

In this work, it is considered that the mature CANDU's technology can be adapted to a new kind of nuclear concepts, as the high conversion high leakage core. The analysis was focused in the use of light or heavy water as coolant, and the capability to get a Conversion Rate (CR) greater than one with a negative void coefficient.

The study started with the evaluation of different kinds of undermoderated cells. Based in the results of this study, a seed and blanket pancake geometry, consisting in sequences of fissile and fertile CANDU-like compact elements located in each channel, is proposed. Interesting properties were obtained using a compact geometry of pressure tubes without moderator in the calandria. The refuelling machine can be used as the main mechanism of reactivity control, and interesting properties were obtained.

Differents configurations of blankets and seeds are analyzed, from the point of view of neutronic, together with the economics of the corresponding fuel cycle. The isotopic composition of the plutonium corresponds to the standard burned fuel from PHWR and PWR.

It can be concluded that the negative void coefficient does not constitute a major problem to obtain high CR (even greater than one) for high compact lattices. The cost of the fuel cycle was found reasonable, and the CANDU technology was found in general compatible with high conversion concepts.

The most convenient plutonium composition is, both from a neutronic and economic point of view, the correspondent of a PWR, for which breeder conditions were found (CR>1) while keeping a negative void reactivity coefficient.

RECOVERED URANIUM IN CANDU: A STRATEGIC OPPORTUNITY

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SUMMARY

Recovered uranium (RU), a by-product from conventional reprocessing with a nominal U-235 concentration of 0.9%, is an attractive fuel for existing and future CANDU* reactors. Its use without re-enrichment in CANDU would yield several benefits. Uranium utilization (the amount of energy derived from the mined uranium used in the original PWR fuel) is improved by about 40%. Because of the neutron efficiency of CANDU and the neutronic characteristics of RU, double the energy can be extracted from the RU by burning it in CANDU, compared to re-enriching it as fuel for a PWR. Fuel burnup in CANDU would be about twice that of natural uranium, resulting in a smaller volume of spent fuel requiring disposal, and a commensurate reduction in back-end disposal costs. By flattening the channel power distribution across the reactor core so that all channels produce nearly the same power, RU offers a power uprating capability in new reactor designs, or in existing reactors where there is sufficient heat removal capacity. Fuelling costs would be significantly lower than natural or slightly enriched uranium (SEU) in CANDU.

The suitability of RU as a reactor fuel for CANDU was assessed in a joint program between AECL and COGEMA. RU powder, and pellets pressed from that powder, met CANDU fuel specifications. One issue that had been identified in an earlier assessment was whether trace amounts of Cs-137 in the RU powder would be released during sintering, and if so, whether this would condense in the cold part of a sintering furnace in a commercial fuel fabrication plant leading to a build-up in fields over time. This was assessed by sintering 4000 RU pellets in a furnace that had been designed with a cold-trap in which volatile Cs released during sintering would condense. This assessment showed that volatile Cs would not pose a radiological problem in a commercial fuel fabrication plant.

Fuel management studies were performed for a CANDU 6 reactor using the CANFLEX bundle with 0.9% SEU, representative of RU. These studies included time-average calculations, and a 120-day time-dependent refuelling simulation. A simple four-bundle shift, bi-directional fuelling scheme resulted in good axial power profiles, and a refuelling rate in bundles per day that was half that of natural uranium, and, in terms of channels per day, comparable to that of natural uranium. Peak time-average channel and bundle powers were 6406 kW and 757 kW, respectively. During the refuelling simulation, the maximum channel power varied between 6500 kW and 6700 kW, and peak bundle power was between 800 kW and 830 kW, well below licensing limits, and comparable to natural uranium. Peak element ratings during the refuelling simulation were below 45 kW/m, which would facilitate good fuel performance at extended burnup. Significant power boosting during refuelling occurred only for relatively fresh fuel, which is tolerant to power boosts. The reactivity worths of control devices is acceptable for safety and control functions.

Fuel cycle economics were assessed for RU in CANDU and re-enriched RU in a PWR. The potential savings in CANDU fuel cycle costs with RU are striking, and significantly greater than the potential cost savings for re-enriched RU in a PWR. With RU at no cost, front-end fuelling costs are reduced relative to natural uranium fuelling by 45% with natural uranium at $25/kg, and by 67% with natural uranium at $80/kg. With RU at natural uranium costs, the fuelling cost savings with RU are 28% for natural uranium at $25/kg, and 34% for natural uranium at $80/kg. Hence, there is a compelling economic incentive for using RU in CANDU.

In summary, excellent neutron economy creates a niche in which CANDU is uniquely suited for burning RU without re-enrichment.

* CANDU: CANada Deuterium Uranium. Registered trademark.

TUESDAY OCTOBER 5

8:30-10:00 **Session C9:** **Steam Generators 1**
Huron Room
Chaired by: E. Price (AECL CANDU)

C9.1 *Managing Steam Generator Margin*
by G.G. Elder (Westinghouse Electric Corporation)

C9.2 *Characterization of Wear Scars on Fretted U-Bend Steam Generator Tubes*
by E.E. Magel and M.H. Attia (Ontario Hydro)

C9.3 *Prediction of Long-Term Fretting Wear Behaviour of Steam Generator Tubes*
by M.H. Attia, E.E. Magel, E. Nadeau, H.L. Anderson and R.G. Sauvé (Ontario Hydro)

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MANAGING STEAM GENERATOR MARGIN

BY

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Steam generators or boilers represent the majority of the primary pressure boundary of the nuclear power plant. Over the past decade they have been a significant contributor to the unavailability of the nuclear power plant. However, a concerted program of inspection, preventative maintenance, and repair techniques can be employed to enable these components to perform reliably and to increase the availability of power plants. The key to reliable operation is to institute a rigorous program of managing the available margin inherent in these components and blending them with the available margin of the balance of the plant components. In this paper several techniques for managing steam generator margin such as sleeving, chemical cleaning, advanced inspection techniques, alternate plugging criteria, etc. will be explored and their impact on steam generator margin discussed. Another phenomena impacting the nuclear industry in recent years has been the apparent rapid growth of steam generator tubing corrosion as detected by eddy current inspection. On detailed analysis, however, it has been shown that in several instances this apparent rapid growth is actually due to an inspection transient. This degradation is actually growing at a much slower rate and is being controlled. The apparent rapid growth is due primarily to changes in the inspection guidelines and inspection equipment. When this is the case, the techniques discussed in this paper can be employed to maximize the operating time of the steam generators at full power. Several examples obtained from operating power plants will be discussed to illustrate these techniques.

CHARACTERIZATION OF WEAR SCARS ON FRETTED U-BEND STEAM GENERATOR TUBES

E.E. Magel[1] and M.H. Attia

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SUMMARY

Fretting wear of steam generator (SG) tubes is a source of great concern. A major research effort is being pursued at Ontario Hydro Technologies to develop the capability of predicting the long-term fretting wear behaviour of SG tubes, using short-term test results and finite element modelling of the dynamics of the tube/support system. A proper modelling of this system requires pre-knowledge of the alignment and concentricity of the tube with respect to the support. Since this information cannot be measured in-situ, a method has been devised to infer the operating conditions, namely, the tube orbital motion, alignment, and axial motion, from the topography of wear scars on tubes removed from the steam generator (see Figs. 1 and 2). To relate field measurements to laboratory-generated data base and code predictions, the relationship between the maximum wear depth and volumetric wear losses is investigated. The wide scatter in the variation of the average wear depth with the scar volume is significantly reduced by using a proposed weighted-average.

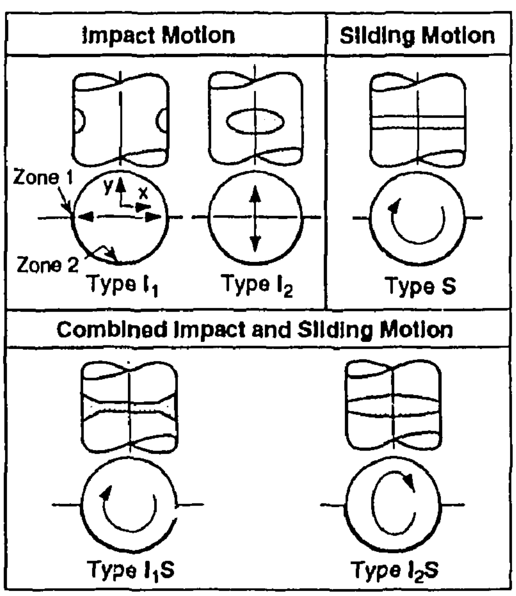


Fig. 2 Effect of tube orbital motion on wear scar profile

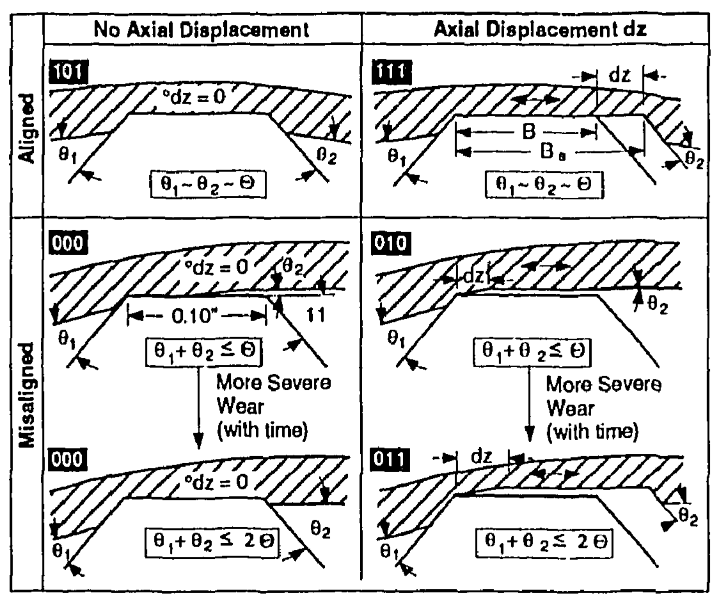


Fig. 3 Effect of tube alignment and axial motion on wear scar profile

[1] Currently with the National Research Council of Canada, Vancouver, B.C.

PREDICTION OF LONG-TERM FRETTING WEAR BEHAVIOUR OF STEAM GENERATOR TUBES

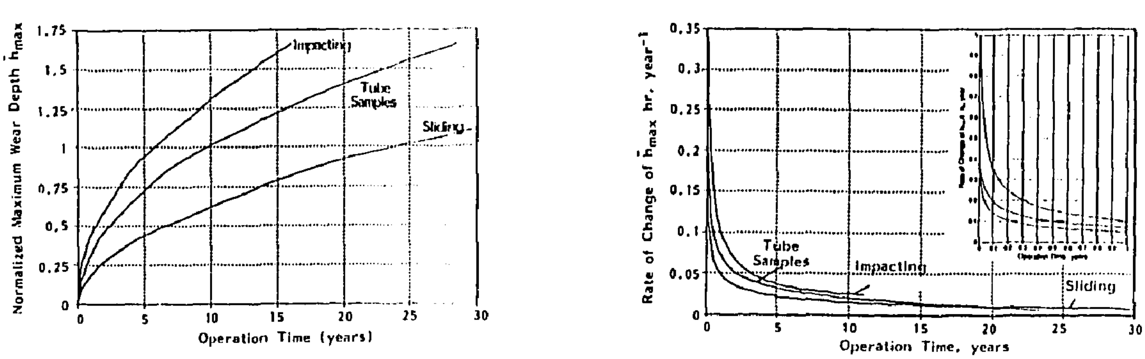
M.H. Attia, E.E. Magel[1], E.Nadeau, H.L. Anderson and R.G. Sauvé

Ontario Hydro Technologies
Toronto, Ontario, Canada

SUMMARY

Accurate prediction of the reduction in the wall thickness of the steam generator SG tube with time, due to fretting wear, is required not only at the design stage, but also for safe and reliable operation of the power plant. Available fretting wear data cannot be used directly since it is based on short-term tests, and correlates *volumetric* wear losses, and not *maximum* wear depth, to the operating conditions. In the present study, a method is proposed for estimating the upper- and lower- bounds of the time history of the maximum wear depth. The method, which accounts for the non-linear nature of the fretting wear process and closed-loop interactions at the tube/support interface, combines the short-term fretting wear data-base with the non-linear forced vibration analyses of the U-bend including clearances in the supports. The relationship between the wear volume and maximum wear depth, which is needed to apply this method, has been established and validated for the offset scallop bar geometry.

To demonstrate the proposed method, a hypothetical computer-simulated case study is has been presented and analyzed. A finite element model was constructed to simulate the non-linear impact forces, tube orbital motion and work rates at the supports. Analysis of the results indicated that the wear rate drops significantly after a short period of operation. This is due to the change in the tube dynamics with increasing clearance between the tube and offset scallop bar support. The analysis suggests also that this rate converges after a short period of operation for all types of tube motion. The time variation of the normalized maximum wear depth and its rate of change are given in Figs. a and b, respectively.



Time variation of (a) the maximum wear depth and (b) its rate of change

[1] Currently with the National Research Council of Canada, Vancouver, B.C.

TUESDAY OCTOBER 5

8:30-10:00 Session C10: Safety 2
Kent Room
Chaired by: J. Gadó
(KFKI Atomic Energy Research Institute)

C10.1 *Better Containment Systems for a Safer Nuclear Future*
by A. Turricchia (ENEL Spa, Rome, Italy)

C10.2 *On-Line Reactor Building Integrity Testing at Gentilly-2*
by N. Collins and P. Lafrenière (Hydro-Québec, Gentilly)

C10.3 *Overpressure Protection Analysis Methodology with RAMONA-3B*
by J.C. Ramos, J. Solis, and G. Cuevas (Instituto de Investigaciones Eléctricas, Cuernavaca, México)

C10.4 *Nine Mile Point Unit 2 IPE Results*
by R.F. Kirchner (Niagara Mohawk Power Corporation)

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BETTER CONTAINMENT SYSTEMS FOR A SAFER NUCLEAR FUTURE

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Summary

Modern containment systems exhibit enhanced capabilities to cope with "Severe Accidents", with respect to older types (which were designed for "Design Basis Accidents" only). However there is room for further progress.

A new containment system has been studied for advanced PWRs with the general design objective of limiting the accidental radioactivity release to the environment to less than one millionth of core inventory (of Cesium, Iodine and other aerosols) by using passive, simple and relatively inexpensive features. It has been named L.I.R.A. in relation to its capability of Intrinsecally Limiting Accidental Releases.The intent is to avoid, even in case of a severe accident, the evacuation of the population living near the nuclear power plant and to limit the contamination of the surrounding territory to such low values as to permit its continued and unrestricted use after the accident.

The design objective has been achieved by adopting the vapour suppression principle, with a drywell concentric with the wetwell, but still retaining a large and strong containment.

The integrity and leaktightness of the containment are preserved against the challenges posed by severe accidents by:

- making the suppression pool large enough to be able to absorb, without boiling, all the heat inputs generated in the first 24 hours;
- burning the hydrogen as it emerges above the suppression pool, by means of D.C. powered spark ignitors backed up by passive catalytic recombiners located near the top of the containment dome;
- locating, in the dry cavity below the pressure vessel, a large graphite slab (or a stack of graphite beams), acting as a solid and temporary heat sink of high thermal conductivity. The thin layer of molten corium spread over it is quickly solidified and cooled down; after this initial corium solidification, water from the suppression pool is allowed to gradually flood the cavity and act as the final heat sink without running the risk of steam explosion.

ON-LINE REACTOR BUILDING INTEGRITY TESTING AT GENTILLY 2

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In october 1992, Hydro-Québec performed the inaugural low pressure (3 kPa(g) nominal) Containment Integrity Test (CIT) at 100% F.P. The test yielded a conclusive leak rate of 0,895% Vol./day (orifice of 9.61 mm dia.) with an unprecedented precision of greater than 1%.

Following the test, the CIT System, based on the Temperature Compensation Method (TCM) was declared In Service. This represented the culmination of a (5) year development program to permit reliable containment integrity verification on-line (e.g. detection of a leak exceeding that of a 2.5 cm diameter opening).

The Gentilly 2 TCM uses a reference volume composed of an extensive tubular network of several different diameters. The reference volume concept eliminates the need to track numerous temperature points. The method includes numerous air sampling points thereby enabling the measurement of minute pressure variations of the reactor building independant of the spatial and temporal humidity behaviour.

This paper provides an overall assessment of the Gentilly 2 CIT System. In particular the Safety and Licensing implications are examined in light of the current regulatory position.

OVERPRESSURE PROTECTION ANALYSIS METHODOLOGY WITH RAMONA-3B

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SUMMARY

A methodology is presented where is proved that the Fuel Management System (FMS) is adequate to carry out one of several transients needed to get a licensing submittal to operate a facility. The paper describes where we are localized and the work that we have been doing in carrying out reload licenses analyses. Among the codes we have at present rely ECLIPSE, POLGEN, and RECORD for the generation of the nuclear cross sections. PRESTO-B is being used for the statics analysis and core following. RAMONA-3B is being used to perform the dynamic analyses and PETRA for collapsing the nuclear parameters from 3D to 1D. The methodology has been proven to successfully perform the analysis of the transients with the most limiting MCPR as established by the FSAR. Here is presented the analysis of the most severe overpresurization transient that for the case of Laguna Verde NPP is the MSIV Closure with high flux scram. It is needed to analyze because the ASME Code requires that each vessel designed to meet section III be protected for overpressure under upset conditions. The Code allows a peak allowable pressure of 110% of vessel design pressure under upset conditions. The design pressure of the CLV pressure vessel must not exceed 1375 psig. It was found that the peak pressure obtained with RAMONA-3B is approximately 50 psig lower than the peak pressure reported in the FSAR. This large difference was expected as the FSAR analysis was done with the old point kinetics and steam line models used in the 70's by GE.

A parametric study was carried out with RAMONA-3B taking the most important input parameters which may affect the limiting criterium, in this case, the maximum peak pressure in the reactor vessel. As a result of the above study, it was found that the code was only highly sensitive to the change of the recirculation pumps inertia and the control rod insertion time. The first with an increase of 20% produced a pressure increase of 9 psi, and the second with a change of 10% an increase of 6 psi. Therefore an effort has to be made in finding the best values for these parameters.

In the effort of validating the methodology and the models for the CLV BWR, some operational transients have been analyzed. The results for the simulation of a Load Rejection and a MSIV closure that have been recorded at the plant are presented here.

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Nine Mile Point Unit 2 IPE Results

Probabilistic Risk Assessment (PRA) studies of operating nuclear facilities are rapidly becoming an important tool for plant decisionmaking. The PRAs provide a technically sound and reproducible means of objectively assessing risk impact of most plant operating issues. In order to describe PRA, this paper will present an overview of the Nine Mile Point Unit 2 (NMP2) PRA. Further, this paper will also discuss some major applications of the PRA to demonstrate the power of the analysis.

Niagara Mohawk Power Corporation submitted the Individual Plant Examination (IPE) for NMP2 to the United States Nuclear Regulatory Commission (USNRC) on July 30, 1992. The NMP2 IPE is classified as a Level II PRA. This means that postulated accident sequences have been analyzed from initiation through to radionuclide release. NMP2 is a Type 5 Boiling Water Reactor (BWR-5) with a Mark II containment design. It operates at a capacity of 1080 MWe and is located on Lake Ontario in upstate New York.

Based on the IPE, it has been concluded that NMP2 has no unusually poor performance issues and poses no undue risk to the health and safety of the public. Two figures of merit typically cited in the above determination are Core Damage Frequency (CDF) and Early/High Release Frequency (ERF). While these figures do not entirely represent the value of the study and uncertainty exists in their exact value, they can be used as indicators and for comparison to other studies. The calculated CDF is 3.1×10^{-5} per year and the calculated ERF is 8.0×10^{-7} per year. Loss of injection scenarios dominate risk with Station Blackout (SBO) contributing highly to loss of injection frequency.

While no outstanding issues were raised that would question NMP2 risk, several insights arose that will lead to cost effective improvement in plant risk. SBO operating procedures under development benefitted from reference to the IPE and review by the IPE team. These procedures help to coordinate and prioritize the efforts required to terminate or mitigate an SBO. Key aspects of the procedure are: DC load shedding, operation of Reactor Core Isolation Cooling (RCIC), AC power recovery, ventilation system augmentation, and diesel fire pump operation. Another insight involved containment venting. Containment venting is a concern when the Residual Heat Removal (RHR) system and the condenser are unavailable. Design and procedural enhancements to allow greater capabilities in vent alignment were identified. In addition, the internal flood analysis noted a concern with emergency diesel generator cooling from the service water system in the vicinity of the emergency switchgear. A procedure for mitigating a break in these lines was developed.

Several issues arose that required additional information or study. These issues were classified as long term items and will be tracked by the IPE during "living PRA" and Accident Management issues. These issues include: service water recovery innovations, use of containment flooding, injection at or around the time of containment failure, standby liquid control recovery, and pedestal water injection. In addition, the importance of equipment maintenance both as a source of equipment unavailability and a means to maximize reliability were highlighted. As such, Reliability Centered Maintenance and continued PRA work have been integrated such that the programs are mutually benefitted.

TUESDAY OCTOBER 5

10:30-12:00 Session C11: Other Reactor Applications 2
Windsor East Room
Chaired by: C. Velez Ocon
(Instituto Nacional de Investigaciones Nucleares)

C11.1 *Description of the Korean Multipurpose Research Reactor*
by J.B. Lee and C.O. Choi (KAERI) and N.B. Dinh (AECL CANDU, Montréal)

C11.2 *The "RB" Reactor as a Source of Fast Neutrons*
by M.P. Pešić and M.J. Milošević (Institute of Nuclear Sciences 'VINČA', Belgrade, Yugoslavia)

C11.3 *The High Flux Reactor at Petten*
by J. Ahlf and G. Tsotridis (Commission of the European Communities Joint Research Centre, Institute for Advanced Materials, Petten, The Netherlands)

C11.4 *Safety and Radioprotection for the TdeV Tokamak Experiment*
by S. Chapados and J.-C. Amrouni (Énergie & analyses Énaq du Québec Limitée) and R.A. Bolton (Centre canadien de fusion magnétique, Varennes, Québec)

DESCRIPTION OF THE KOREAN MULTIPURPOSE RESEARCH REACTOR

SUMMARY

J.B. Lee and C.O. Choi, KAERI, P.O. Box 7, Daeduk-Danji, Taejon, 305-606, Korea
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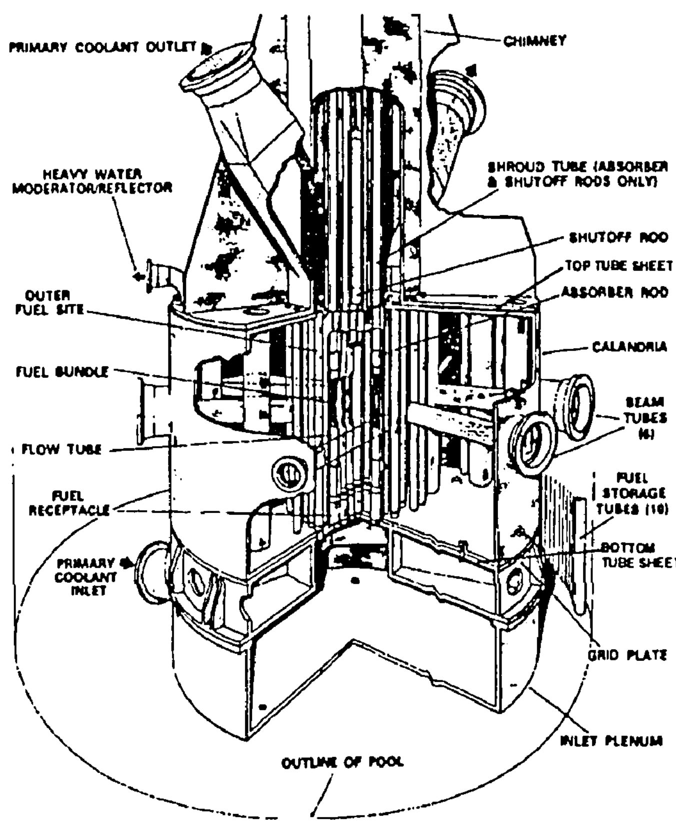
The Korea Atomic Energy Research Institute (KAERI), is undertaking the construction of a nuclear Research Reactor to be known as the Korean Multipurpose Research Reactor (KMRR). Atomic Energy of Canada Limited (AECL), have provided assistance to KAERI in the design of the fuel bundles, reactor structure, beam tubes, control rod mechanism and the reactor control. This paper describes the objectives of KMRR, the reactor concept, the control system and the results of simulations.

KMRR was designed of satisfy the Korean needs for the nuclear fuel and materials testing, the production of key radioisotopes, the neutron activation analysis and the neutron radiography.



Isometric View of KMRR Reactor Structure

KMRR is an open pool type research reactor with a forced upward light water moderator, and cooling flow, and a heavy water annular reflector. KMRR has 39 fuel sites, 7 beam tubes and 25 irradiation sites.

The KMRR core features a combination of light and heavy water reactor lattices. This combination provides extensive variety and flexibility of neutron quality in terms of energy and spatial distribution. When KMRR operates at full power (30 MWth) the maximum available thermal neutron flux is 5.3×10^{14} n/cm^2.s.

THE "RB" REACTOR AS A SOURCE OF FAST NEUTRONS

M.P.Pešić, M.J.Milošević

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SUMMARY

The 'RB' is a critical assembly designed in 1958 to operate using heavy water, natural metal uranium, 2% enriched metal uranium, and 80% enriched UO_2 fuel of Soviet origin. A study of the RB reactor as possible source of fast neutrons began in 1976 and four different version of fast neutron sources were designed up to 1990: an external neutron converter - ENC (1976), an experimental fuel channel - EFC (1982), an internal neutron converter - INC (1983), and a coupled fast-thermal core - HERBE (1990). An overview of applications and characteristics of each particular source of fast neutrons, including available irradiation space, neutron spectra and equivalent neutron and gamma dose rates is presented in the paper.

The ENC transforms thermal neutron leakage flux from the RB core into a fast, near to fission spectrum, neutron flux. It is designed as a wide Al box beside reactor core filled with segments of 80% enriched uranium fuel. The large experimental space and possibility of down-shifting of the ENC output fast neutron spectrum using screens of different materials are the principal advantages of the ENC. The shortcoming of the ENC is low intensity of the fast neutron flux.

The EFC was constructed of modified 80% enriched UO_2 fuel segments placed in a standard fuel channel tube of the 'RB' core, but without presence of the moderator around the fuel. Intensity of the fast neutron flux inside the EFC was elevated on account smaller available experimental space and softer neutron spectrum than it was in the ENC.

The INC fast zone (without moderator) was designed as an 80% enriched fuel elements annulus surrounded with blanket made of two layers of the natural uranium fuel elements in separate Al tanks. A central air hole is designed for irradiation purpose. The thermal zone of the INC is the 'RB' thermal core of 2% and 80% enriched fuel elements surrounded with heavy water reflector.

The HERBE fast neutron core in the RB reactor is recently designed coupled fast-thermal system aimed for verification of results of new developed design-oriented computer codes.

Determination of the fast neutron spectra and other relevant characteristics of the realized fast neutron sources (fields) at the RB reactor were the main experiments performed at the RB reactor in the last 18 years. Several new computer programs for the reactor calculations and experimental data evaluations were developed in the NET Laboratory of the VINČA Institute.

Safety analyses have showed that the RB reactor can operate safely with all types of developed fast neutron sources without any modification of the existing control and safety systems.

THE HIGH FLUX REACTOR AT PETTEN

by

J. Ahlf and G. Tsotridis
Commission of the European Communities Joint Research Centre
Institute for Advanced Materials
Petten, The Netherlands

SUMMARY

The summary was not available at the time of printing.

SAFETY AND RADIOPROTECTION FOR THE TDEV TOKAMAK EXPERIMENT

by

S. Chapados and J.-C. Amrouni
Énergie & analyses Énaq du Québec Limitée

and

R.A. Bolton
Centre canadien de fusion magnétique
Varennes, Québec

SUMMARY

The summary was not available at the time of printing.

TUESDAY OCTOBER 5

10:30-12:00 Session C12: Thermalhydraulics 2
Kenora Room
Chaired by: To Be Announced

C12.1 *A New Facility for the Determination of Critical Heat Flux in Nuclear Fuel Assemblies*
by R.A. Fortman, G.I. Hadaller, R.C. Hamilton, R.C. Hayes, K.S. Shin, and F. Stern (Stern Laboratories Inc.)

C12.2 *Challenges to Computing Buoyancy-Driven Flows in the Containment System of LWRs*
by A. Manfredini, F. Oriolo, A. Villotti (Università degli Studi di Pisa), and S. Paci (THEMAS s.r.l.)

C12.3 *Analysis of Moderator Flow and Temperature Distribution in the Calandria of Madras Atomic Power Station*
by S.P. Dharne and L.G.K. Murthy (Nuclear Power Corporation of India Ltd.) and U.N. Gaitonde (Indian Institute of Technology, Bombay)

C12.4 *Effect of Exit Boundary Conditions on Flow Pattern Transitions in Horizontal Fuel Channels of a PHWR*
by V.M. Wasekar and K. Iyer (Indian Institute of Technology, Bombay)

A NEW FACILITY FOR THE DETERMINATION OF CRITICAL HEAT FLUX IN NUCLEAR FUEL ASSEMBLIES

R.A. Fortman, G.I. Hadaller, R.C. Hamilton, R.C. Hayes, K.S. Shin, F. Stern
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A new facility for the determination of Critical Heat Flux in simulated reactor fuel assemblies has been constructed at Stern Laboratories for CANDU Owner's Group. The main circulating pump is rated 55 $l.s^{-1}$ at 380 m head and provides steady-state or transient light water flow at up to 14.6 MPa. Heat rejection from the primary loop is accomplished through either or both of two heat exchangers, and by feed and bleed using a separator and condenser. The system pressure is maintained by controlling the rate of bleed from the separator through a pressure reducing station to a low pressure condenser. The temperature is adjusted by controlling the rate of feed of cold makeup water in conjunction with control of the flows through the heat exchangers. Hydrazine is injected into the primary loop water to control the pH level between 7.9 and 8.5 to maintain the dissolved oxygen content below 5 ppb to minimize corrosion of the test loop components which are mainly carbon steel. The electrical conductivity of the loop water is monitored and generally held below 3 $\mu mho.cm^{-1}$ to prevent electrolytic corrosion.

Electrical power is provided by 8 individually controlled rectifiers, (3 units rated at 1.0 MW and 5 units rated at 2.5 MW) with a total rated output capacity of 15 Megawatts DC (ie. 60,000 amps @ 250 volts). The rectifier units are specially designed for low ripple DC output with a maximum ripple of less than 3% over the range of 25% to 100% power. The ripple frequency is 720 Hz. The supplies are remotely controlled using the laboratory computer system with custom, keyboard driven, software which outputs a control setpoint, handles the current sharing among supplies, and provides incremental control, ramping etc. of each power supply. For power measurement, current is measured by shunts in each power supply, and also the total current is measured by a Hall Effect device. Voltage taps are attached directly.

The facility is fully instrumented with many redundancies to ensure accurate measurements of power, flow, temperatures and absolute and differential pressures. All instrument signals are acquired by the laboratory computer system which consists of a DEC VAX 4000-100 computer, clustered with a MicroVAX 2000 computer and two VAXStation 3100 workstations running under the VAX/VMS V5.3-1 operating system. The system includes four CPI scanners with 120 input channels each for analogue to digital conversion, an HP A600 computer with 80 input A/D channels, a MicroMAC digital input/output system with various analogue and relay outputs for process control, magnetic disk units (350 Mb, 600 Mb and two 150 Mb), a 1 Gb/650 Mb R/W optical disk unit, a tape backup system, various graphics terminals, text display terminals, video display monitors, and graphics printers.

The data acquisition system is used for the on-line monitoring of the fuel string thermocouples for CHF determination. Ten thermocouple signals can be graphically displayed in real time on each of two VS3100 workstations and the operators can select on-line, via keyboard, the thermocouple channels to view and several (up to 10) groups of channels can be preconfigured for individual selection by keyboard entry. Temperature changes of the order of 1°C are readily detectable on the screen and dryout behaviour is easily distinguished on the monitored thermocouples. In addition, all of the thermocouple signals are continuously scanned by the data acquisition system and if any of the bundle thermocouple signals indicate dryout, that thermocouple identification channel is flashed onto the video screen to alert the test operators.

CHALLENGES TO COMPUTING BUOYANCY-DRIVEN FLOWS IN THE CONTAINMENT SYSTEM OF LWRs

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SUMMARY

The simulation of buoyancy-driven flows in a LWR multicompartment containment system, due to density gradients that may arise during a severe accident, were demonstrated to be a fundamental item to obtain a realistic description of the thermal-hydraulic transient and aerosol behaviour. These phenomena have a great influence on various aspects of the containment safety analysis; the proper evaluation of energy and mass transfers is fundamental, particularly during the long term phase of a severe accident sequence, for the prediction of pressure loads, hydrogen distribution and aerosol dynamic. Besides, natural circulation is the most important way to remove the decay heat power and therefore to maintain the integrity of the system in the new advanced nuclear power plants.

An advanced model for evaluating the temporal and spatial distribution of non condensable gases, including the simulation of buoyancy-driven flows, in a multicompartment containment system of a LWR is reviewed. The model employs an analogy technique with electrical networks to determine the convection flows among the containment compartments and evaluates, inside a single node, the profile of the vertical concentrations of steam and non condensable gases.

The proposed model has been used to investigate the natural circulation phenomena occurring in the HDR E11.2 and in the FIPLOC-F2 tests. The description of natural circulation through both closed loops for the test were correctly performed (Fig 1). The distribution of the flow velocity is characterized by the main convection loop which changes its direction under the influence of the different injections. In particular, the natural circulation models are able to describe the timing of flow reversals, leading to a good prediction of steam and air distributions and consequently of the temperature trends. The experimental conditions of the HDR E11.2 test simulate a SBLOCA followed by a hydrogen injection. The steam leaves the input location near the staircase and rises in the direction of the dome. The air already in the dome is thus displaced over the spiral stair into the lower region like in a plug flow. The injected steam strongly enriches volumes above the break location but it does not return down in a substantial quantity, through diffusion processes.

The comparison between the experimental data and the models predictions showed that an appreciable improvement in the modeling of the natural circulation phenomena exists. The errors on the temperature and gas distributions using these new models are reduced in a substantially way, necessary condition to perform a realistic prediction of a severe accident sequence. The comparison with results obtained in preliminary assessment analysis show the validity of the new natural circulation models and their usefulness for the coupled analysis required from the new generation of LWRs.

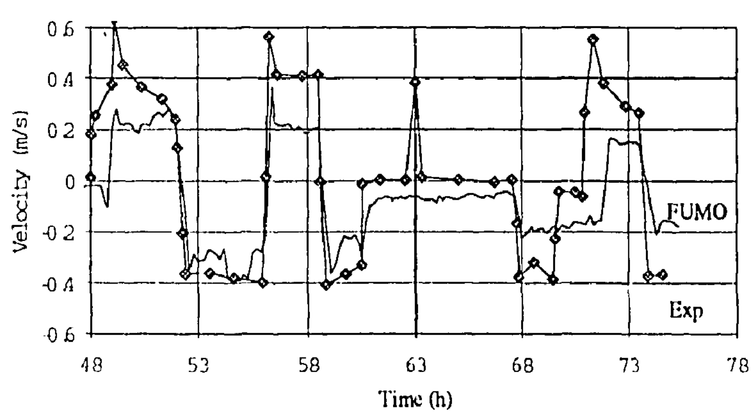

Fig. 1: Natural circulation velocity.

ANALYSIS OF MODERATOR FLOW AND TEMPERATURE DISTRIBUTION IN THE CALANDRIA OF MADRAS ATOMIC POWER STATION

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SUMMARY

A code to predict the steady-state as well as transient velocity and temperature distributions inside the calandria of Madras Atomic Power Station(MAPS) was developed. The exercise was necessary to compute, in reasonable detail, the flow fields and temperature distributions in MAPS calandria in its present mode of operation with original outlet serving as the inlet and the dump ports serving as outlets.

To predict the velocity and temperature fields of the moderator, a 3D finite-difference code(MFLO) was developed. The code solves the governing differential equations in a Cartesian coordinate system. The other characteristics of this code include a volume-based porosity to model the tube nest, a distributed resistance scheme to model the flow resistance of the tube nest, a control-volume based method for deriving the finite-difference form of the governing equations, and a 3D whole field iteration scheme based on the SIMPLE-C technique.

The predictions from the code include the 3D fields of velocity, pressure, and temperature. Results have been obtained for the cases of (a) original flow configuration and (b) modified flow configuration. Effects of reactor loads and moderator inlet temperatures have been studied with the help of the code.

The results obtained for the original flow configuration for 100% load show that (a) there is a significant effect of buoyancy forces, (b) there is stable thermal stratification in the moderator tank, (c) the cold moderator from the inlet manifold moves over the dump ports, distributes itself, and then rises through the core. The top sprays (cold moderator) deflect this upflow and the mixed stream then exits through the exit manifold. (d) It is noticed that there is hardly any short-circuiting of the flow from inlet to exit. (e) The specially-directed tubesheet sprays are found to be very effective in keeping the tubesheet cool.

The results obtained for the modified flow configuration show that (a) the flow is significantly governed by buoyancy, which makes the inflow turn and go down, in spite of having an initial upward velocity, (b) this stream gets diffused on the way due to heating and mixing, (c) the remaining zones of the calandria essentially experience an upward movement of the moderator and consequent thermal stratification, (d) the tubesheet sprays dictate the flow pattern and temperature profiles near the tubesheet, and (e) no significant pockets of stagnating fluid exist.

EFFECT OF EXIT BOUNDARY CONDITIONS ON FLOW PATTERN TRANSITIONS IN HORIZONTAL FUEL CHANNELS OF A PHWR.

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SUMMARY

In this paper systematic investigations have been carried out to study the influence of exit boundary conditions on the two-phase flow pattern transitions in the coolant channel of a Pressurized Heavy Water Reactor (PHWR).

An experimental facility consisting of a horizontal channel of 85 mm in diameter and 4.5 m long with provisions for the simultaneous introduction of air and water flow up to a maximum of 5000 lpm and 100 lpm respectively, was built. It was provided with suitable flow monitoring devices with an accuracy of better than 2.5% of the measured values. Suitable inserts to simulate the fuel bundles were fabricated. This included 1-rod, 7-rod and 19-rod assemblies.

Experiments were carried out to demarcate the flow pattern transitions in the horizontal channel with the inserts. To characterize the effect of boundary conditions, three kinds of exits, viz., a horizontal exit, a vertical exit, a vertical exit on a composite section simulating the shield plug assembly in a PHWR, were used. The flow patterns were characterized visually.

The observed flow patterns were classified as stratified, wavy stratified, wavy annular, annular and intermittent flows. The results clearly brought out the fact that there was a significant influence of the exit boundary on the stratified to wavy transition line and the stratified to intermittent transition line. These were primarily affected due to the varying position of the air-water interface in the test section, which was controlled by the nature of the exit boundary condition.

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TUESDAY OCTOBER 5

10:30-12:00 Session C13: Social Issues 2
Windsor West Room
Chaired by: R. Summers (Canadian Nuclear Association)

C13.1 *Perspectives of the Proponent and Initiating Department on the Federal Environmental Review of the Canadian Nuclear Fuel Waste Management Program*
by B. Gray (AECL Research, WL) and G. Underdown (Department of Natural Resources Canada)

C13.2 *Twenty Years of Nuclear Program Support for Social Science Research*
by D.R. Hardy (Hardy Stevenson and Associates)

C13.3 *Discussing Nuclear Energy Issues at School: How to Teach the Teachers?*
by F. De Galzain (OECD Nuclear Energy Agency)

C13.4 *Physical Models of Nuclear Public Acceptance*
by T. Ohnishi (CDC Research Institute, Japan)

V 2 of 3

PERSPECTIVES OF THE PROPONENT AND INITIATING DEPARTMENT ON THE FEDERAL ENVIRONMENTAL REVIEW OF THE CANADIAN NUCLEAR FUEL WASTE MANAGEMENT PROGRAM

by

BARBARA GRAY
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SUMMARY

Studies on disposal of used nuclear fuel in Canada have been carried out under a joint R&D program by AECL Research and Ontario Hydro. The work was formally initiated in 1978 by a joint statement from the governments of Ontario and Canada, and the mandate reconfirmed in 1981. The program of research has developed the technology and methodology for isolating used nuclear fuel from the biosphere through a series of engineered and natural barriers. The aim of the research has been to develop a full understanding of the underlying mechanisms of the behaviour of each part of this system. The work has been directed towards development of a concept for used nuclear fuel disposal. No site specific work has been, nor will be, carried out until there has been an environmental assessment and public review of the concept, and governments have made decisions on the safety and acceptability of the disposal concept, and the future steps that must be taken to ensure the safe, long-term management of nuclear fuel waste in Canada.

In September 1988, the AECL concept for geological disposal of nuclear fuel waste was referred to the Minister of Environment for environmental assessment and review under Section 13 of the Environmental Assessment and Review Process Guidelines Order. A Review Panel was appointed, and a Scientific Review Group of independent experts was established by the Panel to examine the safety and scientific acceptability of the disposal concept. Public hearings are expected to take place sometime in 1994.

In 1990, open houses and public scoping (information-gathering) sessions were held to provide information on the process for the review, and to identify issues that should be addressed in the formulation of the Environmental Impact Statement guidelines. The guidelines, which form the basis of the documentation that AECL is required to develop in preparation for public hearings on the concept, were issued in March 1992.

There are two features of the review which make it different from other environmental reviews. Firstly, the proponent, AECL, is presenting a generic disposal concept, in other words an approach to disposal, rather than a site specific disposal project. This makes the review very comprehensive, because technologies adaptable to a range of conditions must be considered. The review is necessarily somewhat abstract in that questions on site specific issues must be considered in the absence of a directly affected community. Secondly, the sensitivity of the public and the media towards the nuclear industry and to disposal of radioactive waste have resulted in a need to engender a very high level of confidence in the proposed concept.

The Canadian government is committed to policies and programs for the management of radioactive wastes from the nuclear fuel cycle, including nuclear fuel waste, that combine the need for public review, effective regulation, and the use of safe and environmentally sound technologies. As a whole, the objective is to ensure that our nuclear industries should be well placed to meet the highest standards of safety and environmental impact over the long term.

The paper will discuss the details of the review process and the progress to date. It presents some perspectives from both the initiating department and the proponent in the review, and describes the opportunities for technical and public contribution to the program and its review.

Twenty Years of Nuclear Program Support for Social Science Research

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The Canadian nuclear program has done much to support research and development in the physical sciences. The list of successes is long. And, in many areas, Canadian engineers and physicists have become world leaders. Yet equal in importance, but lesser known, are the social science research accomplishments resulting from nuclear program financial support. The societal pay-back of over twenty years of support for social science research has been a wealth of important: communications approaches: methods of relating to communities; social impact assessment studies; understandings of risk; approaches to site and route selection; and, other contributions to plan and program assessment. Social science research over the years is extensive, as seen in Figure 1.

As we track this research, we see that one of the most direct results of social science research support has been the contribution of our ability to understand the nature of contentious policy issues. For example, in the 50s, contentious public issues were seen to be a public relations challenge. With increased public awareness and the rise of participatory democracy in the 60s and 70s, sides were chosen and these issues became matters of approval. In the 80s and 90s, as a broad range of issues reach impasse, social scientists are beginning to further understand the nature of, and approach to contentious policy issues.

For example, with support from AECL, Hardy Stevenson and Associates recently completed several significant studies of the social and moral aspects of nuclear fuel waste management. With funding support we were able to further confirm our hypothesis that, to be effective in planning for the approval of contentious projects and policies, (not just nuclear) the proponent must understand the "socially systemic" nature of the issues involved.

A 'systemic' issue is one that: absorbs the interest of normally disinterested groups and individuals; has strong moral and ethical underpinnings; and, in some way, involves issues of human health and risk. Among the list of socially systemic issues, nuclear energy is not unique.

How would one approach a systemic issue that is different than how one would address normal issues To begin with, the approach to a systemic problem begins with the recognition of the depth of the issues involved. Developing a communications plan, or hiring a public relations consultant to do a public opinion poll, for example, is not likely to be effective. The issues are deep and complex. They cannot be addressed by finding the 'right spin'.

We suggest that the approach involved requires an understanding of the potential contribution of all of the social sciences to the development of acceptable policy choices. As indicated on Figure 2, appropriate action involves, understanding the problem to be a societal problem rather that an industry or section of society problem; breaking the issues down into its component parts; accepting the potential contribution of other disciplines; assessing what approaches are appropriate for what issues; understanding when an issue is ready to go forward for political choice; and focusing on what can change and what cannot.

The nuclear energy industry has been one of the major benefactors of social science research. As discussed above, social sciences are able to now provide sound research supporting better approaches to systemic policy issues.

DISCUSSING NUCLEAR ENERGY ISSUES AT SCHOOL: HOW TO TEACH THE TEACHERS?

Florence DE GALZAIN, Information Officer

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92130 ISSY-LES-MOULINEAUX (FRANCE)

1. Future generations will have a decisive role to play in the development of nuclear energy, given that policy decisions will continue to be made well into the future. It is thus at school that we need to make an effort now to inform and teach those who will be called upon in the future to make decisions regarding energy and environment.

2. Organizing a teaching programme on nuclear issues requires an integrated approach of societal factors involved in nuclear energy. Therefore, it should simultaneously explore the issues of nuclear energy under different but implementary approaches: Energy, Environment, Economy and Ethics.

3. Against this background, the question of the training of the teachers themselves takes a particular significance: are they properly prepared through their initial education to deal with present societal factors such as nuclear power and are they able to present the related options to pupils in an objective and fairly complete manner?

4. The specific role of the teachers' educators is essential here: they have to make sure that the teachers are provided with all the necessary information to not only understand but also to be able to explain in turn to their pupils the different implications of nuclear energy on society. They also have to help the teachers in organizing their teaching so that they could easily be integrated and adapted within the current educational programme.

5. The specificity of topics like nuclear or other current social issues calls for original and appropriate training and teaching methods. A new teaching approach could be proposed, based on a system of thematic sessions. Each of the sessions would be dedicated to a major current social issue and would involve both teachers from different disciplines and representatives from industry in the field under discussion. Various forms of complementary education for teachers have to be also introduced, such as site or plant visits, or training sessions with teachers having already gone through such programmes to exchange experience both on methods and results.

6. The OECD Nuclear Energy Agency is particularly well-suited to take an initiative for improving the situation of education in the field of nuclear energy. Thus the NEA has undertaken to develop an international understanding on the basic training and information requirements to assist secondary school teachers in discussing nuclear energy in an appropriately wide and balanced context at school.

PHYSICAL MODELS OF NUCLEAR PUBLIC ACCEPTANCE

Teruaki Ohnishi

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Two physical models were developed to study how nuclear public opinion varies under the influence of the newsmedia, mutual communication among the public, and sociopsychological pressure in the society. Using these models investigations were done on what condition must be satisfied for the amelioration of nuclear public acceptance in the future. In these models the amount of nuclear information released by the newsmedia is treated as one of the major exogenous variables. One of these models is a statical one where the psychological and social behavior of the public is treated statically, whereas it is treated dynamically in the other model.

In the statical model, the public are assumed to change their opinion in proportion to the amount of information provided by the media. Mutual communication among the public is assumed to begin round a few opinion leaders as seeds who are highly excited by the media, and the public are also assumed to change their attitude toward anti- or pro-nuclear states, being subject to the influence of mutual communication. In this model, the public are modelled as an aggregate of hexagonal cells adjoining to each other and the cellular automaton method is introduced to mimic the expansion of anti- and pro-nuclear thought. In the dynamical treatment, on the other hand, the multi-particle model is introduced to describe the sociopsychological interactions among the public and among the newsmedia. In this model, a state variable representing the status of anti- or pro-nuclear attitudes for each particle is introduced, and the social pressure which is exerted between individuals is calculated by assuming the form of potential between the particles. In these models, constants and parameters were determined so that the calculation reproduces the secular variation of Japanese public opinion concerning nuclear issues in the past.

By the simulation with the statical model, the follows became clear: Our society is a non-linear system where self-organizing process to form groups takes place. In the society constituted from only one population with an uniform characteristic, the trend of public opinion can turn over catastrophically when the extent of continuous effort to ameliorate the nuclear public acceptance through education, persuasion and advertisement exceeds a certain critical value. In the case when the amount of information concerning nuclear risk released by the media continues to be decreased from now on, the nuclear public acceptance can be ameliorated catastrophically only if the extent of the decrease exceeds some critical value. Also from the dynamical model, the follows became clear: The public have a psychological tendency for the cohesion to result in self-organization. The public make a synergetic system with the newsmedia in the society. The rapid rise and fall of the concern regarding anti-nuclear thought is induced in the public mind as a result of non-linear interaction between the public and the media. To ameliorate the nuclear public acceptance, it is quite important for us to spur the media so that the media inform the public of accurate nuclear information.

TUESDAY OCTOBER 5

10:30-12:00 Session C14: Fuel & Fuel Cycles 2
Wentworth Room
Chaired by: G.N. Barceló (CNEA, Centro Atómico Bariloche)

C14.1 *Recycling: The Advanced Fuel Cycle for Existing Reactors*
by G. Lamorlette (COGEMA)

C14.2 *The International Uranium Market*
by K.L. Smith (UNECO, Canada)

C14.3 *Well Field Development at the Crow Butte ISL Uranium Mine*
by G. Kirchner (Uranerz Exploration and Mining Limited), and G. Catchpole (Uranerz U.S.A. Inc.)

C14.4 *Uranium Ores Treatment and Uranium Dioxide Production for Fuel Elements Production for CANDU Nuclear Power Plant in Romania - Achievements and Available Assets*
by C. Bejenaru (Rare Metals Company, Bucharest), D. Georgescu (Institute for Rare and Radioactive Metals, Bucharest), and M. Bobe and A. Alda (Mineral Dressing Plant "R", Brasov)

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RECYCLING: THE ADVANCED FUEL CYCLE FOR EXISTING REACTORS

by

G. Lamorlette
COGEMA

SUMMARY

The summary was not available at the time of printing.

THE INTERNATIONAL URANIUM MARKET

K. L. Smith, UNECO *
International Nuclear Congress
October 3-6, 1993

Abstract

Any analysis of the international uranium market must recognize that the concept of "supply and demand" cannot be applied in the conventional manner. Because of the large inventories that are held by utilities, traders, producers and governments, procurement from producers does not reflect total "demand", and supply does not correspond to uranium production.

Western world uranium production is currently only about half of annual uranium requirements. The gap is being filled from excess utility inventories and by imports of low priced uranium from the Eastern Bloc countries. This situation is keeping the uranium spot-market price at an artificially low level -- below the cost of production for most western producers.

The paper reviews the basic mechanics of the uranium market, and includes some historical background in order to provide an understanding of how the uranium market arrived at its current situation.

The paper develops a possible scenario for the draw-down of inventory over the remainder of this decade, and estimates the potential supply from Eastern Bloc countries. The purpose of this "top-down" analysis is to forecast the required procurement of uranium from western uranium producers in the period up to 2010. The paper quantifies the assumptions needed to produce this analysis, and identifies the factors that could cause the forecast to change. Some of the questions that must be considered in producing such a forecast include:

- What is the total quantity of disposable inventory held by western world organizations, and would it be drawn down at the rate suggested in the paper?
- To what extent would western utilities be willing to depend on supply from Eastern Bloc countries? Would this reach 25% of total requirements. This might be an upper limit for some utilities, but is it perhaps too high for an average value?
- Does the forecast of imports from Eastern Bloc countries adequately allow for future agreements regarding the de-enrichment of Russia's large military stockpile of highly-enriched uranium?

Having generated a forecast of procurement from western producers, the paper compares this forecast with the existing production capability and concludes that the proposed expansion of Canadian production facilities will be needed by the latter half of the 1990s in order to avoid a production shortfall.

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SUMMARY

WELL FIELD DEVELOPMENT AT THE CROW BUTTE ISL URANIUM MINE

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The Crow Butte project, located in the northwest corner of the state of Nebraska, is one of four commercial in-situ leach (ISL) uranium mines currently operating in the United States. Commercial production at the mine started in April 1991, some 12 years after the discovery of the sandstone, roll front uranium deposit. Calculated reserves for the project are in excess of 30 million pounds U_3O_8 with an average ore grade of 0.25% U_3O_8. The ore body lies about 200 metres below the surface in a highly permeable aquifer with excellent confining shale/clay layers above and below the deposit. The exploration and development period included ore body delineation, pilot testing, environmental licensing, engineering design, and facilities construction. Production at the mine is planned to increase from the current annual level of 600,000 pounds U_3O_8 to one million pounds U_3O_8 in 1995. Commercial ISL mining operations, using alkaline leach chemistry and oxygen, have gone smoothly with no major technical or regulatory problems. Spacing between injection and recovery wells in the 5-spot and 7-spot patterns is nominally 20 metres. The mine, in year three of commercial production, is now operating in the third mining unit with some production still coming from the first and second mining units.

The method for establishing the locations of the injection and recovery wells, and the method of well construction have been modified since the first commercial mining unit, Mine Unit 1, was installed The initial method of well construction, referred to as the "cement basket" water well completion technique, produced good flowing wells but uranium recovery was less than expected based on earlier pilot tests. In a review of the production data from Mine Unit 1, it appeared that the recovery solution was being diluted because of overscreening of the ore zone. A second method of well completion, call "underreaming", was successfully tested and subsequently utilized to install wells in the third commercial mining unit, Mine Unit 3. The uranium recovery performance of Mine Unit 3, after five months of operation, has clearly demonstrated that the underreaming technique of well installation at the Crow Butte ISL mine is both cost effective and highly efficient at recovering uranium. Recovery and injection flow rates in the wells were not adversely impacted by the underreaming technique and, environmentally, both the old and new techniques prevent contamination of adjacent aquifers through complete cementing of the well annulus.

URANIUM ORES TREATMENT AND URANIUM DIOXID PRODUCTION FOR FUEL ELEMENTS PRODUCTION FOR CANDU NUCLEAR POWER PLANT IN ROMANIA-ACHIEVEMENTS AND AVAILABLE ASSETS

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ABSTRACT

In order to ensure the necessary fuel for NPP Cernavoda with 5 x 700 MW, which is now under construction, Romania has organized its own activity of exploitation and treatment of uranium ores based on the deposits located in the territory.

The exploitation started in 1952 with the deposit from the Apuseni Mountains, oneof exceptional grade, in 1956 began exploitation of deposits from the Banat area and in 1980 at one deposit in Oriental Carpathians.

The uranium ore production reached a peak in 1952-1961, for export purposes. The exploitation started again when the possibilites of ore processing whitin the country became actual by a new processing plant ready for operation in 1978 at Feldioara, district of Brasov. The technological flowsheet comprising basically the carbonate lixiviation in autoclaves with mechanical agitation, extraction-backextraction of the solubilized uranium usine RIP methode, uranium precipitationand dryng by atomising of sodium diuranate.

In 1985 a processing plant of the technical concentrates has been achieved and commissioned whitin the same plant, in order to obtain the sintering powder for the nuclear fuel production in the NPP Cernavoda, based on a classical flowsheet consisting mainly in solubilization in nitric acid, TBP solvent purification in mixer-settlers, precipitation with ammonium hydroxide roasting and hydrogen reduction of the pure ammonium diuranate to uranium dioxide.

After 1987, in order to ensure the demand of nuclear fuel for the five units of the NPP Cernavoda, the development with modern equipment and technology has been started. Due to the moderate and gradual program of commissioning of the NPP Cernavoda, after 1990 the investments ceased and the works already executed, about 30% at the new concentration plant and about 80% at the new refinery, are being preserved.

The paper presents some aspects regarding uranium romanian ores and methods, equipments, technical and economical parameters for the operating plants, introduction in this plant ofanew technology for achieving new technological parameters, similar to those obtained in others advanced plants and also the level reached and available assets for the partially executed installations.

TUESDAY OCTOBER 5

10:30-12:00 Session C15: Steam Generators 2
Huron Room
Chaired by: P. Lafrenière (Hydro-Québec)

C15.1 *Advances in Nuclear Steam Generator Technology for Improved Reliability*
by J.C. Smith (Babcock & Wilcox International)

C15.2 *Laser Welded Sleeving - A Proven Technology for Steam Generator Life Enhancement*
by B.R. Nair (Westinghouse Electric Corporation)

C15.3 *A Horizontal Steam Generator for Indian 235-MW Heavy-Water Nuclear Power Plants*
by D.R. Iyer (Nuclear Power Corporation, Bombay)

✓

ADVANCES IN NUCLEAR STEAM GENERATOR TECHNOLOGY FOR IMPROVED RELIABILITY

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N1R 5V3

Summary

Steam generator reliability problems arise mainly from tube corrosion issues. The majority of such problems today are OD and ID stress corrosion cracking problems in steam generators which contain mill annealed Alloy 600 tubing. Dealing with the problems requires detailed inspection and testing to quantify where and what size the cracks are, and a variety of repair techniques are then available. The ultimate solution to serious steam generator tube failures may be complete steam generator replacement; a process which is being undertaken in nuclear plants in numerous countries. The replacement steam generators are expected to provide significant improvements in reliability over the old steam generators due to improvements in technology in areas such as tube materials, and key design features such as tube supports. The nuclear industry has matured to the point where new steam generators are now fully expected to perform reliably through the life of the reactor system in which they reside.

LASER WELDED SLEEVING - A PROVEN TECHNOLOGY FOR STEAM GENERATOR LIFE ENHANCEMENT

Bala R. Nair
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SUMMARY

Laser welded sleeving was performed for the first time in the United States in April 1992 at the J. M. Farley Nuclear Plant Unit 2. In all, 68 tube support plate sleeves and 30 tubesheet sleeves were installed in two steam generators. This was followed by a larger sleeving campaign in the plant's Unit 1 steam generators in October 1992 when 148 tube support plate sleeves and 46 tubesheet sleeves were installed in three steam generators. The successful implementation of this new technology at Farley provides the industry with a field proven and effective option to repair steam generator tubes and maintain operating plant performance.

The laser welding was performed using a fiber optic delivery system to transmit light energy from a pulsed solid state laser located outside containment to the weld head which could be positioned remotely in the tubesheet and as high as the sixth support plate in the steam generator. The sleeve material was thermally treated Alloy 690 (UNS 06690). The joint design was a partial penetration, autogenous weld. All free-span welds, namely the support plate sleeve welds and the tubesheet sleeve upper weld, were thermally stress relieved to enhance stress corrosion life. Those welds were also required to pass a stringent ultrasonic test examination. All the processes for laser welded sleeving were performed remotely using the Westinghouse steam generator service robot, ROSA III.

The Farley campaigns showed that laser welded sleeving offers a high degree of process control not found with other methods and produces welds that can be fully inspected by ultrasonic examination. They also demonstrated the field hardiness of the sophisticated laser welding system.

SUMMARY

A HORIZONTAL STEAM GENERATOR FOR INDIAN 235 MW HEAVY WATER NUCLEAR POWER PLANTS

D.R.IYER
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Vikram Sarabhai Bhavan,
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In our country like in most part of the globe, the steam generators for the PHWR programme have been designed as a vertical type. This paper describes the design of horizontal steam generator for our programme. Such steam generators (S.G.) have successfully operated in VVER type PWRs upto a capacity of 500 MW. The main advantage of the horizontal steam generator is the large evaporation surface and ability to circulate coolant water without much concentration of impurities. The large volume of water makes the cool down during emergency shutdown of the reactor much easier. Besides, the S.G. is designed for the removal of residual heat during shutdown through natural convection. The main operating parameters of the steam generators are:

1. Primary (heavy water) flow = 3175 tons/hr,
2. Primary pressure = 91.4 kgf/cm2
3. Primary temperature at inlet and outlet = 293.4 deg.C and 249.3 deg.C respectively.
4. The secondary (steam) flow = 332.5 tons/hr
5. Secondary pressure = 40 kgf/cm2
6. Secondary temperature at the inlet and outlet of the steam generator = 171.0 deg.C and 250.6 deg.C respectively

The main parts of the horizontal steam generator are shell, feed water distribution leader, heat transfer surface in the form of S.S. or inconel 'U' tubes (both options are considered), primary headers (replacement for tubesheet in vertical steam generator) and pressure equaliser.

The proposed material for construction are: High strength low alloyed steel of type GOST-22K for the shell, Low Alloy Steel or stainless steel for the headers and stainless steel or incoloy for the heat transfer tubes.

The design takes into consideration the local manufacturing capabilities.

TUESDAY OCTOBER 5

10:30-12:00 Session C16: Safety 3
Kent Room
Chaired by: L. Simpson (AECL Research, WL)

C16.1 *A Kinetic Model for Fission-Product Release and Fuel Oxidation Behaviour for Zircaloy-Clad Fuel Elements Under Reactor Accident Conditions*
by B.J. Lewis (Royal Military College of Canada), D.S. Cox (AECL Research, CRL), and F.C. Iglesias (Ontario Hydro)

C16.2 *The Importance of Organic Compounds in Evaluating Accident Management Strategies*
by C.A. Chuaqui and J. Ball (AECL Research, WL) and R. Fluke, J. Edward and K. Weaver (Ontario Hydro)

C16.3 *The Radiolysis of Aqueous Organic Systems and Its Effect on Iodine Volatility*
by R.C. Quan, M. Mesbah-Oskui and G.J. Evans (University of Toronto)

C16.4 *A Neural Network Model of Volatile Fission Product Release from Fuel Elements and Fragments Under Severe Accident Conditions*
by W.S. Andrews and B.J. Lewis (Royal Military College of Canada) and D.S. Cox (AECL Research, CRL)

A KINETIC MODEL FOR FISSION-PRODUCT RELEASE AND FUEL OXIDATION BEHAVIOUR FOR ZIRCALOY-CLAD FUEL ELEMENTS UNDER REACTOR ACCIDENT CONDITIONS

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D.S. Cox, Chalk River Laboratories, Atomic Energy of Canada Limited - Research Company,
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SUMMARY

During a severe reactor accident fission products will be released from the degraded fuel in the reactor core. In addition, hydrogen will be generated at high-temperature by the steam oxidation of the core materials. This oxidation process will also influence the rate of fission-product release. Separate-effects tests performed out-of-pile at the Chalk River Laboratories (CRL) have provided a better understanding of the processes of fission product release during severe accident conditions. The annealing experiments were conducted in steam at temperatures ranging from 1200 to 1700°C with irradiated fuel specimens of uranium dioxide in the form of bare fuel fragments and a short-length Zircaloy-clad fuel element. The fission product release was monitored by on-line gamma ray spectrometry. The oxygen partial pressure was also measured with solid-state oxygen sensors, providing a calculation of the rate of oxygen consumption and hydrogen production in the fuel specimens.

Based on the CRL tests, an analytical model has been developed to describe the kinetic release behaviour of the volatile fission-product species (cesium) during high-temperature accident conditions. The physically-based model accounts for the kinetics of fuel oxidation as a rate-determining reaction at the fuel/steam interface. A more general framework is therefore provided to detail the influence of the atmosphere (i.e. oxygen potential) on the behaviour of the fission product release. Solid state diffusion in the fuel matrix is shown to be the rate-controlling mechanism in the early stages of release. The enhanced diffusivity of fission products in the hyperstoichiometric fuel is modelled with the assumption that diffusion takes place on vacant cation lattice sites. When the fuel reaches a state of oxidation of $x \sim 0.07$ for the UO_{2+x} phase, a more rapid release process occurs in accordance with first-order rate kinetics. The retarding influence of the hydrogen production on the fuel oxidation kinetics from the Zircaloy-steam reaction is also considered.

THE IMPORTANCE OF ORGANIC COMPOUNDS IN EVALUATING ACCIDENT MANAGEMENT STRATEGIES*

Claudio A. Chuaqui[1], Richard Fluke[2], Joanne Ball[1], Jeremy Edward[2] and K. Weaver[2]

(1). Fission Product Chemistry Section, Research Chemistry Branch, Whiteshell Laboratories, AECL Research

(2). Ontario Hydro

SUMMARY

One of the most important ultimate objectives of the Fission Product Chemistry Programme is to enable the prediction and control of iodine volatility in reactor containment buildings following an accident. The ability to provide a mechanistic determination of iodine volatility is significant in the Canadian setting because of iodine's important role in safety and licensing, emergency response and post-accident recovery operations. A mechanistic predictive capability is also important in the assessment of severe accidents, in accidents involving containment bypass and in accidents where release via waterborne pathways may be significant (e.g., as in Three Mile Island where containment sump water was pumped into the auxiliary building).

The tool being developed to meet this objective is LIRIC, a library of those reactions and rate constants which have some significance for iodine volatility. LIRIC is, therefore, a vehicle for codifying and structuring the important experimental information on iodine.

There is an experimental programme with three distinct elements supporting the development of LIRIC: (1) "all-effects" integral tests using the Radioiodine Test Facility (RTF), (2) a flexible programme of bench-scale studies and scoping tests, and (3) a programme of underlying fundamental work, including derivation of kinetic and thermodynamic data. The RTF attempts to provide qualified and controlled integrated tests involving iodine, which include all the effects that would be expected to containments post-accident. The bench-scale studies and the fundamental work, along with LIRIC, are intended to allow the RTF results to be interpreted mechanistically.

We describe here our work on the effects that the presence of organic materials in containment has on the volatility of iodine species in case on a reactor accident, the factors that affect it, and its implications regarding accident management strategies.

* The authors wish to thank the CANDU Owners Group, consisiting of AECL, Ontario Hydro, New Brunswick Power and Hydro Quebec for providing the funds to continue the research reported in this paper

THE RADIOLYSIS OF AQUEOUS ORGANIC SYSTEMS AND ITS EFFECT ON IODINE VOLATILITY

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Radioiodine (I-131) is an important fission product available for release from a nuclear reactor following an accident. This is because it has a high radiotoxicity and can assume many volatile (ie. airborne) forms, giving it the potential to be released from containment structures, causing significant environmental damage.

It is believed that certain organic compounds may enhance the formation of volatile iodine species. These compounds may exist in nuclear reactor containment structures in paints and wiring. The objective of this study was to identify specific organic compounds which may enhance or suppress iodine volatility. The experiments involved the irradiation (dose rate: 12.5 kGy/hr) of aqueous 10^{-5}M CsI solutions containing an excess (10^{-3} to 10^{-1}M) of an organic compound. These conditions, other than the concentrations of organic compounds, were selected so as to approximate the post-accident environment expected in a CANDU reactor containment structure. Many of the compounds were found to contribute to the formation of volatile iodine. The iodine partition coefficient (IPC=iodine concentration in the liquid phase/iodine concentration in the gas phase) varied, from as low as 300 for aqueous chloroform solutions, to as high as $1x10^5$ for aqueous phenol solutions. As a reference, a partition coefficient of 10^4 is often used in safety analysis.

It was established that organic compounds can have a significant impact on radioiodine volatility. Out of some 33 compounds tested, three distinct categories of IPC/pH behaviour were identified. The observed behaviour could be interpreted in terms of radiation chemistry: the interaction appeared to be for the most part indirect, through scavenging of free radicals, induced reduction in pH and consumption of dissolved oxygen. The importance of pH changes was clearly demonstrated in this work. The behaviour of the compounds in each group was explained through the preferential consumption of the oxidizing (OH) and reducing (e^-_{aq} and H) free radicals produced as a direct result of water radiolysis. Compounds which were found to enhance iodine volatility typically consumed e^-_{aq} and H in preference to OH. In some cases dissolved oxygen also played a significant role. This was confirmed in preliminary experiments conducted in a stainless steel vessel agitated so as to produce a well characterized rate of mass transfer. Addition of MEK to this system at constant pH caused much of the dissolved oxygen to be consumed, and a substantial increase in iodine volatility. Further investigation is required in order to clarify the relative importance of the various mechanisms by which organic compounds can affect iodine volatility.

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A NEURAL NETWORK MODEL OF VOLATILE FISSION PRODUCT RELEASE FROM FUEL ELEMENTS AND FRAGMENTS UNDER SEVERE ACCIDENT CONDITIONS

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SUMMARY

The accidents at Three Mile Island and Chernobyl have underscored the need for developing source terms for fission products during severe accident conditions. In order to refine the source term for CANDU fuel, annealing experiments under varying temperature and atmospheric conditions have been carried out at the Chalk River Laboratories (CRL) and have involved measuring the cumulative release of long-lived fission product species.

Annealing tests on light water reactor (LWR) fuel in steam at the Oak Ridge National Laboratory have led to the development of a correlation of cumulative release with temperature and time in the form of the CORSOR-M model. This model is an important tool used for source term prediction by the United States Nuclear Regulatory Commission (USNRC). The applicability of the CORSOR-M model to CANDU pressurized heavy water reactor (PHWR) fuel, however, has not yet been established.

Further, no comprehensive physically-based model has yet been developed for the release of fission products from either LWR or CANDU fuel in any environment other than steam. To redress this, an artificial neural network has been developed to model the cumulative fission product release of cesium-134 determined from the varied tests of three extensive experiments conducted at CRL: Hot Cell Experiment 1 (HCE-1), Hot Cell Experiment 2 (HCE-2) and Metallurgical Cell Experiment 1 (MCE-1). Based on a modified back propagation paradigm, a multidimensional correlation has been able to model the cumulative release fraction of fission product cesium under a variety of temperature, atmospheric (steam, air and inert gas/hydrogen), heating and sample conditions being represented by 14 input states. The network was trained with 3662 different input sets, or vectors, some of which were reproduced a number of times to produce a balanced training set of 12 516 vectors. The training was conducted over 7 epochs, or complete repetitions of the training set. An independent database of 396 vectors was used to test the trained network. A further data set comprising the results of a complete test not part of the training set was used to validate the network.

Networks with differing numbers of nodes in the hidden layer have been able to establish a good correlation between 14 input parameters or variables and the cumulative release fraction of cesium. The correlation coefficient for the relatively large test set varies between .93 and .95, depending on the architecture. The validation, although underpredicting the release fraction at lower and higher fractions, was successful in predicting within an overall standard error of 12% and in reproducing the non-linear response of cumulative release fraction with time. This suggests that a neural network model, which is capable of running in real time, can be imbedded into real-time computer codes used for modelling reactor accident phenomena.

WEDNESDAY OCTOBER 6

8:30-10:00 Session C17: Physics 1
Huron Room
Chaired by: To Be Announced

C17.1 *HUEMUL: A Transport Code for General Geometries Including Reactivity Devices - Its Validation Against Measurements*
by C.R. Calabrese, C. Grant, A.M. Lerner, C. Notari, and O. Serra (Comisión Nacional de Energía Atómica, Argentina)

C17.2 *A General Comparison of the Lattice Codes APOLLO-2 and DRAGON*
by A. Hébert (École Polytechnique de Montréal)

C17.3 *Burnable Poison: A Solution for Fuel Management in 1.2% SEU Fueled CANDU 6 MK1 Core*
by D. Serghiuta, E. Nichita, O. Nainer, and P. Laslau (Institute for Nuclear Research, Pitesti, Romania)

C17.4 *Selection of Materials with Low Induced Activity Following Neutron Irradiation*
by K.T. Tsang and C.R. Boss (AECL CANDU)

HUEMUL: A TRANSPORT CODE FOR GENERAL GEOMETRIES INCLUDING REACTIVITY DEVICES ITS VALIDATION AGAINST MEASUREMENTS

Carlos Rubén Calabrese, Carlos Grant, Ana María Lerner, Carla Notari, Oscar Serra
Comisión Nacional de Energía Atómica (CNEA)
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SUMMARY

The HUEMUL code was developed in Comisión Nacional de Energía Atómica (CNEA), Argentina. It solves the Two Dimension (2D) Transport Equation using the multigroup collision probability method for general geometries. Arbitrary segments and circunference arc combinations, permit a great flexibility in geometry treatment. On other hand boundary conditions in the form of an albedo matrix (J+/J-) are included for each face of the model, no matter the external face number.
Due to these characteristics HUEMUL becomes a useful tool to face a great variety of problems, specially the calculation of parameters associated to control rod and reactivity devices.

Several calculations were performed using HUEMUL:

a) Comparisons of results between HUEMUL and WIMS for a light water cell with 3% wt enriched UO2. They show a very good agreement in fluxes and multiplication constant k (better than 1.5% in fluxes and 0.5 mk in k)

b) Comparisons between HUEMUL calculations and copper activity measurement performed in the Canadian D2O facility ZED-2 with stainless steel adjuster rods. A very good agreement was obtained (better than 2%).

c) Comparisons between HUEMUL calculations and manganesum activity measuremets performed in the Argentinian H2O facility RA-2. The calculated values are within the measuremets with their experimental errors.

d) Rod calculations for the Atucha I power plant (CNA1). Comparisons were made between reactor calculations performed with PUMA and DELFIN using incremental cross sections calculated by HUEMUL and experimental data obtained at zero power. The results show a very good agreement (better than 2%).
The results obtained show that HUEMUL is a powerful tool to face 2D reactivity devices problems. It can be used in experimental and power reactors, specially for Atucha I and II.

A GENERAL COMPARISON OF THE LATTICE CODES APOLLO-2 AND DRAGON

by

A. Hébert
École Polytechnique de Montréal

SUMMARY

The summary was not available at the time of printing.

BURNABLE POISON : A SOLUTION FOR FUEL MANAGEMENT IN 1.2% SEU FUELED CANDU 6Mk1 CORE

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Summary

Utilization of burnable poison for the SEU fueled CANDU 6Mk1 core is proposed. The main incentives for this proposal are the reduction of void reactivity effects - due to increased fuel absorption - and the achievement of better axial as well as radial power shape control. The latter allows the preservation of construction parameters of the standard core (e.g. number and location of reactivity devices) . It also permits the use of regular shift fueling schemes , as opposed to the more involved "advanced schemes" , usually proposed for the SEU fueled core.

The paper makes an analysis of a 1.2% SEU - boron added - fueled CANDU 6Mk1 core. In this paper, ^{10}B was chosen as burnable poison. Comparison is made to a reference case consisting of a 1.2% SEU - no boron - CANDU core. Two-, and four-bundle regular shift schemes are considered for the boron case. A two-bundle regular shift scheme is used for the reference case.

The analysis of 35 ppm boron 1.2% SEU fuel for the CANDU 6Mk1 core revealed the following aspects:

- Significant reduction of the void effect is achieved (17.2 mk compared to 22.5 mk for 1.2% SEU, and 20.8 mk for natural uranium fuel)
- Adequate control of radial and axial power shape is obtained. Radial flattening and axial depression compensation are achieved.
- A four-bundle shift refueling scheme can be used. In spite of the higher CPPF, lower maximum bundle power is attained.
- Good fuel operating conditions are achieved. The lack of high burnup power boosts and the low linear power density permit the use of 37-rod fuel bundle.
- The maximum discharge burnup penalty is of only 7.1% compared to the no boron SEU fuel case.

As manufacturing of poison UO_2 pellets for PWR reactors is a well-known process, no significant difficulties in the manufacturing of UO_2 pellets - burnable poison added - for CANDU fuel are to be expected.

SELECTION OF MATERIALS WITH LOW INDUCED ACTIVITY FOLLOWING NEUTRON IRRADIATION

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SUMMARY

This paper will present a systematic approach to select a material that is least prone to neutron activation in a CANDU reactor. Hitherto, the way selection of such a material has been a review of the thermal neutron absorption cross sections, the stability of the daughter products and the type of decay. This approach, however, neglects the effects of epithermal and fast neutron activation in a typical spectrum. The approach that we adopted was to use the isotope generation and depletion code ORIGEN-S[1] to perform an activation analysis of 81 naturally occurring elements, from atomic number 1 to 83 in the periodic table. From these 81 ORIGEN-S outputs, the gamma energy and nuclear energy data were extracted and then integrated for a period of one week. Since radiation dose is proportional to the gamma energy release and the extremity dose is proportional to nuclear energy release, ranking the elements based on their relative amount of gamma and nuclear energy released over the integrating period will enable us to have an accurate choice of an element that is least prone to neutron activation. Based on the scenario we used for this paper, of all the transition elements, vanadium would be the preferred choice for tools that will be exposed to high neutron fluxes for short duration and at infrequent intervals.

REFERENCE:

1. O.W. Hermann, R.M. Westfall, ORIGEN-S User Manual - Draft, ORNL, February 1989

WEDNESDAY OCTOBER 6

8:30-10:00 Session C18: Plant Components
Kenora Room
Chaired by: H. Stremler (GE Canada, Inc.)

C18.1 *Improvements in the Fracture Toughness of CANDU Zr-2.5 Nb Pressure Tubes*
by G.D. Moan (AECL CANDU), J.R. Theaker, P.H. Davies, I. Aitchison, and C.E. Coleman (AECL Research, CRL), R.A. Graham (Teledyne Wah Chang), and S.A. Aldridge (NU-TECH Precision Metals)

C18.2 *Nuclear Sources of Hydrogen in CANDU Fuel Channels*
by M.A. Lone (AECL Research, CRL)

C18.3 *Using Improved Elastomers To Enhance CANDU Station Reliability*
by R. Wensel (AECL Research, CRL)

C18.4 *A Versatile Electrical Penetration Design Qualified to IEEE Std 317-1983*
by W. Lankenau and T.M. Wetherill (Imaging and Sensing Technology)

IMPROVEMENTS IN THE FRACTURE TOUGHNESS OF CANDU Zr-2.5Nb PRESSURE TUBES

G.D. Moan (AECL CANDU), J.R. Theaker, P.H. Davies, I. Aitchison, C.E. Coleman (AECL CRL), R.A. Graham (Teledyne Wah Chang, Albany, Oregon), S.A. Aldridge (NU-TECH Precision Metals, Arnprior, Ont.)

ABSTRACT

Programs carried out recently with funding from the CANDU Owners' Group (COG) in co-operation with the pressure tube manufacturers, TWCA and NU-TECH, have led to improvements in the fracture toughness of the Zr-2.5Nb pressure tubes being made for use in CANDU fuel channels.

In the first program it has been shown that the ranking of the fracture toughness (FT) of several different pressure tubes was unaffected by the test temperature and irradiation. Studies of the microstructures showed that differences in FT were related to the presence in the fracture surfaces of fissures, that were shown to be associated with regions that contained chlorine and carbon. The FT of the tubes was high if the chlorine concentration was low. It was also found that the tube with the highest FT had been made from an ingot that was made up 100% from recycled material, so that the material in the final ingot had been melted four times.

Trials were carried out using tubes made from ingot material that was deliberately melted four times. The FT of this material showed the expected improvement to high values. Other properties have been monitored and no deleterious effect has been found. The current practice is to use quadruple melted material for new pressure tubes.

In the second program the aim was to reduce the hydrogen concentration in new pressure tubes by determining the processing steps that led to increases in its concentration. The manufacturers have introduced changes to their procedures and it has been possible to reduce the specified maximum concentration from 25 to 5 ppmH. Tubes with lower hydrogen concentrations are less likely to initiate cracks by Delayed Hydride Cracking.

The paper will summarize the work carried out in the programs and will indicate the improvements that have been achieved in the pressure tube properties.

NUCLEAR SOURCES OF HYDROGEN IN CANDU FUEL CHANNELS

by

M. Aslam Lone
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In this paper we examine the nuclear processes that generate hydrogen (the word hydrogen herein, refers to any isotope of the element), and give quantitative estimates of their contributions to the ingress of hydrogen in the fuel channels of CANDU reactors. These channels are exposed to high fluences of neutrons and gamma rays of energies up to tens of MeV and the nuclear processes that can generate noticeable amounts of hydrogen ions are:

-(n,p), (n,d),(n,t), (γ,p) and (γ,d) reactions in Zr, Nb and impurities in the tube material,
-elastic scattering of neutrons from H and D atoms in the cooling water, and
-photo nuclear reaction with D in the cooling water.

A CANDU 600 fuel channel was considered as a reference. The production of hydrogen isotopes from nuclear reactions in the tube materials, including possible trace chemical impurities of $Z < 50$, were examined. The cross sections for nuclei with $Z > 50$ are too low to generate noticeable amounts of hydrogen. The ingress of hydrogen generated by neutron elastic scattering and photo nuclear reactions were evaluated. The collisions of fast neutrons with hydrogen and deuterium atoms in the water molecules produce knock-on ions that can implant in the fuel channels. Similarly, the gamma-ray-induced disintegration of deuterium will produce higher energy protons that can implant in the fuel channel.

The objective of this study was to examine the upper limits of the potential contributions from the nuclear processes. Thus where information on nuclear data and concentrations of chemical impurities was unavailable, upper bounds were adopted. The total ingress of hydrogen in the fuel channels was estimated assuming that the nuclear reaction products are trapped permanently in the fuel-channel material.

In a CANDU 600 reactor, the lattice array pitch is 28 cm and there is about 15 cm of heavy-water moderator between the adjacent fuel channels. With this configuration, the high-energy neutron flux in the pressure tube is predominantly due to the fission neutrons coming from the enclosed fuel channel. The neutron and gamma ray fluxes at the fuel channel location were obtained from reactor physics calculations with the codes WIMS and ANISN.

Cross sections for the nuclear reactions needed for this study were generally available from the compilations generated by the international nuclear data centers. The ingress of hydrogen in the pressure tube was calculated from the reaction yield of 1 g of the tube material placed in the neutron and gamma fluxes.

The total ingress from the nuclear sources over 20 years was calculated to be less than 4 ppm by wt. The study points to impurities like nitrogen that could generate high levels of hydrogen from the $^{14}N(n,p)^{14}C$ reaction.

USING IMPROVED ELASTOMERS TO ENHANCE CANDU STATION RELIABILITY*

R. Wensel

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SUMMARY

Much of the equipment in nuclear generating stations is critically dependent on the reliable performance of elastomeric components (O-rings, diaphragms, gaskets, etc.). These are prone to physical damage and are subject to change in material properties with time, temperature, fluid contact, and other environmental influences.

There are many different base elastomer types. Within each, there are many different compounds (specific combinations of ingredients). Effectively, these compounds are compromises between various functional properties. Selection, qualification, and specification of the best compound for particular applications requires compound-specific testing, with careful choice of performance criteria and test methodology.

An ongoing program that is improving elastomer performance, safety margins, and service life prediction in CANDU nuclear generating stations is described. The program encompasses:

- identification and retrofitting of improved elastomer compounds,
- development and implementation of new elastomer compound specifications and quality assurance procedures,
- development of guidelines for installation, handling, and inspection of elastomeric components, and
- development of a service and shelf-life database for these compounds.

Close interaction with station maintenance personnel helps ensure that work focuses on the highest priority problems. On-site work includes inspecting components for degradation, troubleshooting failures, monitoring replacement compounds during phase-in periods, and providing "workshops" on elastomer selection and application.

In the laboratory, simulated-service testing is conducted for compound selection and qualification, and for establishment of quality assurance procedures and procurement specifications. Selection is based on properties directly related to the service function. For example, O-ring selection warrants bench-scale testing of O-rings using the actual process fluid at representative pressures, temperatures, and clearances, and could include evaluation of compression set, age-hardening and extrusion resistance, in addition to leak-tightness. In some tests, certain conditions may be increased in severity, to assess margins of safety or to accelerate aging.

Compounds can vary widely in performance even with the same base elastomer type. Therefore, procurement and quality assurance specifications are tied to specific elastomer components, fabricated of specific elastomer compounds, for specific service equipment. For each application, attempts are made to qualify compounds from at least two different sources.

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* Paper offered to the Technical Review Committee for presentation at the International Nuclear Congress to be held in Toronto, 1993 October 3-6.

A Versatile Electrical Penetration Design Qualified to IEEE Std. 317-1983

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SUMMARY

Although worldwide demand for new construction of nuclear power stations has been on a decline, the available opportunities for the design and construction of qualified electrical penetrations continues to offer challenges, requiring a highly versatile design. Versatility is necessary in order to meet unique customer requirements within the constraints of a design basis qualified to IEEE Std. 317-1983.

This paper summarizes such a versatile electrical penetration designed, built and tested to IEEE Std. 317-1983. The principal features are described including major materials of construction. Some of the design constraints such as sealing requirements, and conductor density (including numerical example) are discussed.

The requirements for qualification testing of the penetration assembly to IEEE Std. 317-1983 are delineated in a general sense, and some typical test ranges for preconditioning, radiation exposure, and LOCA are provided.

The paper concludes by describing ways in which this versatile design has been adapted to meet unique customer requirements in a variety of nuclear power plants.

WEDNESDAY OCTOBER 6

8:30-10:00 Session C19: Waste Management 1
Kent Room
Chaired by: F. McDonnell (AECL Research, WL)

C19.1 *Mineralogical and Corrosion Problems Underlying the Choice of Materials for High-Level Nuclear Waste Disposal*
by M.B. McNeil (U.S.NRC) and T.L. Woods (East Carolina University)

C19.2 *Role of the Geosphere in the Canadian Concept for Nuclear Fuel Waste Disposal*
by C.C. Davison, F.P. Sargent and S.H. Whitaker (AECL Research, WL)

C19.3 *Fully Treated and Solidified Radioactive and Hazardous Wastes Belong in an Above-Grade, Earth-Mounded, Concrete Disposal Vault*
by G.R. Darnell (INEL-EG&E Idaho Inc.)

C19.4 *Long-Term Safety Assessment of the Disposal of Nuclear Fuel Waste*
by K.W. Dormuth, B.W. Goodwin and A.G. Wikjord (AECL Research, WL)

<u>Mineralogical and Corrosion Problems Underlying the Choice of Materials for High-Level Nuclear Waste Disposal</u>. M. B. McNeil, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington DC 20555 and T. L. Woods, Dept. of Geology, East Carolina University, Greenville NC 27858

SUMMARY: The U.S. Nuclear Regulatory Commission requirement that reasonable assurance be offered that high level waste packages remain intact for at least three hundred years requires that three different groups of technical issues must be addressed. First, the degradation characteristics of the candidate waste package materials under anticipated repository conditions must be determined, and some technique must be found for estimating degradation behavior over long periods. Second, some method must be found of estimating the uncertainties in the extrapolation which arise from inadequate understanding of the degradation processes. Third, an analysis must be performed of the stability of the corrosion process; that is, whether very minor changes in the initial conditions or parameters of the process lead to large deviations in the expected behavior.

The choice of material for a high-level waste package is generally between a material which shows great short-term corrosion resistance but for which no archaeological or natural analogue data are available and one which shows limited corrosion resistance but for which some analogue (but generally not very closely analogous) data are available. A natural tendency is to regard superior short-term performance as more important than analogue data of arguable relevance. This choice assigns a greater importance to the predicted behaviour than to the confidence with which the predictions can be made. Corrosion phenomena can show instabilities due to bi- or polyfurcation, and there exist data that indicate the potential for other types of long-term instabilities.

Consideration of all these various sources of uncertainty indicates that the ability to estimate low corrosion rates over long periods may be less important than the ability to project low <u>uncertainties</u> in the projected rates, and that this suggests that choice of a system for which natural analogue data exist may have substantial advantages with regard to support of long-term performance evaluations. The simultaneous use of electrochemical experimentation, mineralogical studies, and judicious use of natural analogues appears more likely to provide an adequate capability to predict, and to judge uncertainties in predictions of, long-term corrosion behavior than more straightforward approaches based only on short-term testing.

ROLE OF THE GEOSPHERE IN THE CANADIAN CONCEPT FOR NUCLEAR FUEL WASTE DISPOSAL

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SUMMARY

The Canadian Nuclear Fuel Waste Management Program was established in 1978 by the governments of Canada and Ontario to ensure the safe disposal of nuclear fuel waste. It is jointly funded by AECL and Ontario Hydro under the CANDU Owners Group. The disposal concept is being reviewed by a federal Environmental Assessment and Review Panel.

The Canadian concept for disposal of nuclear fuel waste is to place the waste in long-lived containers and emplace the containers, with sealing materials, in a vault 500 m to 1000 m deep in plutonic rock of the Canadian Shield. The disposal vault would be located and designed to inhibit release of contaminants from the vault and movement of contaminants to the surface environment in the groundwater within the surrounding rock.

The geosphere comprises the rock mass in which the disposal vault is constructed, including the groundwater pathways in the vicinity of the vault, especially all the pathways by which contaminants from the vault could enter the groundwater flow systems and eventually reach the surface environment.

The role of the geosphere within the disposal system would be:

- to protect the waste form, container and seals from natural disruptions and human intrusion in the long-term;

- to maintain thermal and geochemical conditions in the vault favourable for long-term waste isolation;

- to limit the rate at which contaminants could move from the vault to the surface: and

- to enable a safe working environment to be maintaiined during construction and operation.

On the basis of our field and laboratory investigations, we have concluded that currently available methods for site characterization and performance assessment are sufficient to determine whether or not the geosphere at any potential disposal site on the Canadian Shield would play its role adequately in contributing to a safe disposal system. We have also concluded that the conditions expected at vault depth in plutonic rock on the Canadian Shield are such that availability of technically suitable sites would be unlikely to be a limiting factor in implementing disposal.

FULLY TREATED AND SOLIDIFIED RADIOACTIVE AND HAZARDOUS WASTES BELONG IN AN ABOVEGRADE, EARTH-MOUNDED, CONCRETE DISPOSAL VAULT[a]

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SUMMARY

Imagine a single waste treatment and disposal complex where contact-handled low-level radioactive waste, Classes A, B, and C, and radioactively contaminated hazardous waste (mixed waste) would all be dealt with as a single waste in both treatment and disposal. Is this too simplistic, too complicated, too costly, or unacceptable considering today's national, state, and local regulations? Perhaps so, perhaps not. Regardless of the answer, the concept deserves serious consideration.

Current philosophy calls for separating various classifications of waste for separate treatment in different treatment facilities (or for no treatment at all) and then taking the various waste packages of many different sizes and shapes and placing them in separate disposal facilities with separate licensing. With some effort on our part, these unnecessarily complicated and expensive concepts of today can be turned around for the betterment of the nation in all respects.

Consider all of the busy, but unnecessary people involved in all of the separate organizations required in our currently planned and imagined individual concepts—managers, planners, design engineers, draftsmen, environmental engineers, architectural engineers, waste characterization specialists, clerks, secretaries, technicians, lawyers, technical editors, and on and on. We could all be doing constructive work elsewhere. We need a firm solution to our nuclear and hazardous waste problem so that we can get on with promoting nuclear power to provide economical electricity to recharge the batteries of the environmentally sound electric cars of the future.

With all of the combined waste fully treated to the highest standards, it could be placed in an environmentally sound abovegrade, earth-mounded, concrete disposal vault and licensed according to existing regulations, or more likely to new national, state, and local rules and regulations tailored to the proposed concept. In doing this, cost and radiation exposure to workers would be dramatically reduced while providing exceptional environmental protection. We can only hope that our congressional leaders and national, state, and local regulators would be willing if not pleased to bless the concept and allow it to happen with minimum documentation and oversight.

a. Work supported by the U.S. Department of Energy, Assistant Secretary for Environmental Restoration and Waste Management, under DOE Idaho Operations Office Contract No. DE-AC07-76IDO1570.

LONG-TERM SAFETY ASSESSMENT OF THE DISPOSAL OF NUCLEAR FUEL WASTE

by

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SUMMARY

AECL has proposed a concept for disposing of Canada's nuclear fuel waste. The concept has been referred for review under the Environmental Assessment and Review Process. The proposed concept is to emplace the waste in an underground disposal vault excavated 500 to 1000 m deep in the plutonic rock of the Canadian Shield. The disposal vault is to be passively safe, that is, long-term safety would not depend on institutional controls.

After vault closure, the disposal system would consist of the underground vault with its containers of waste, and sealed rooms and tunnels; the rock surrounding the vault; and the potentially affected near-surface and surface environment. The postclosure safety assessment integrates relevant information from site investigations, laboratory studies, expert judgment, and mathematical analysis, to evaluate how this system may be expected to function in terms of safety standards over many thousands of years.

For this assessment, we do not have a specific site and facility to evaluate. Therefore, we have specified an illustrative reference disposal system and evaluated its postclosure performance to demonstrate how an actual system would be assessed and to indicate what margins of safety might be expected. To make the reference system representative of a real system, we have used the geological observations at our Whiteshell Research Area to define the characteristics of the geosphere surrounding the vault and the groundwater flow system in the vicinity of the vault, including discharges to the surface. We derived the characteristics of the surface environment using information from appropriate locations on the Canadian Shield. The reference waste form is used CANDU fuel.

Our analyses of the reference disposal system indicate that humans and the environment would be effectively protected with a large margin of safety relative to existing regulatory standards. We have identified those parts of the system to which the safety performance is most sensitive. For example, the results are very sensitive to the layout of the vault relative to important features within the geosphere. We illustrate the use of performance assessment to establish design objectives and constraints.

In our opinion, the postclosure environmental impacts can be assessed with sufficient reliability to site and design a facility, based on the proposed concept, to meet regulatory standards; the assessment can be used to influence the facility design so as to optimize the margin of safety; the reliability of the assessments can be enhanced by gathering more detailed information during the construction and operation of the facility; and the methodology is compatible with current regulatory requirements and is sufficiently flexible to accommodate any reasonable changes in these requirements.

WEDNESDAY OCTOBER 6

8:30-10:00 Session C20: Accelerators & Industrial Radiation 1
York Room
Chaired by: T. Sasaki (JAERI)

C20.1 *AECL's IMPELA™ Electron Accelerators for Industrial Uses*
by A.J. Stirling (AECL Accelerators)

C20.2 *Advances in Radiation Processing of Polymeric Materials*
by K. Makuuchi and T. Sasaki (Takasaki Radiation Chemistry Research Establishment, JAERI) and A.C. Vikis and A. Singh (AECL Research, WL)

C20.3 *Effects of High Radiation Environments on Polymer Composite Epoxies*
by H.W. Bonin, H.M. Pak, V.T. Bui and D. Rhéaume (Royal Military College of Canada)

C20.4 *Some Calculational Results for Transmutation of Plutonium and Wastes in Blankets of Accelerator-Based Systems*
by B.P. Kochurov (Institute of Theoretical and Experimental Physics, Moscow, Russia)

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AECL's Impela™ Electron Accelerators for Industrial Uses

by

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Summary

In November 1992, AECL Accelerators completed the design, construction and commissioning of the world's most powerful 10 MeV electron accelerator in a commercial processing plant. The plant is owned and operated by E-Beam Services Inc.of New Jersey. The accelerator is AECL's Impela™ electron linear accelerator. It is characterized by its energy of 10 MeV and power of 50 kW. This high power level opens new horizons for the use of electron beams for modifying materials. The accelerator facility at E-Beam Services plant in Cranbury is a contract service centre. The products treated include medical devices that are sterilized, plastic pellets that are crosslinked, and formed plastic parts that are strengthened through crosslinking.

In executing the project in New Jersey, E-Beam Services designed and constructed the building, constructed the radiation shield, provided the personnel safety system and had the electrical and cooling services installed. AECL designed, built, installed and commissioned the accelerator. Both parties were under extreme pressure to meet the schedule. AECL's future clients and competitors were scrutinizing the progress minutely to assess the new technology. For E-Beam Services the pressure came from clients requiring firm dates before they could commit to processing contracts.

AECL's Impela™ is constructed to some 2300 controlled drawings, calling on standard and custom components from both foreign and local suppliers. The key item is the 3m high-purity copper accelerating structure. This consists of 58 close-tolerance cavities brazed face to face. Machining, tuning, brazing and vacuum testing were performed at AECL's former Medical Products plant.

The construction of the facility followed straight-forward building practices. E-Beam Services acted as its own general contractor, and purchased a prefabricated building. Accelerator installation was carried out by AECL staff and contractors form the local area.

The Impela™ accelerator at E-Beam Services was installed and commissioned fast and efficiently for the first model of its type and located outside the supplier's home country. It was completed a few days ahead of schedule due to excellent experience during commissioning. The elements which led to this success included

- Excellent and frequent communication between project managers of supplier and client,
- Competent project managers,
- Access to highly qualified contractors in the New Jersey area,
- Accelerator design complete and fully documented before construction, and
- Having a prototype operating prior to construction so that lessons were learned in the laboratory rather than in the client's plant.

The Impela™ is now operating at full power routinely in the E-Beam plant. Its installation will, in all likelihood, mark a new chapter in industrial radiation processing.

ADVANCES IN RADIATION PROCESSING OF POLYMERIC MATERIALS

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SUMMARY

In this paper we review recent advances in industrial applications of electron beam irradiation in the field of polymer processing at the Takasaki Radiation Chemistry Research Establishment (TRCRE) of JAERI, Japan, and the Whiteshell Laboratories of AECL Research, Canada. Irradiation of a substrate with ionizing radiation produces free radicals through ionization and excitation events. The subsequent chemistry of these radicals is used in radiation processing to substitute conventional processing techniques based on heating and/or the addition of chemicals. The advantages of radiation processing include formation of novel products with desirable material properties, favourable overall process economics, and often environmental advantages.

At AECL Research, carbon and aramid fibre-reinforced composites are being studied for a variety of structural applications. These composites, primarily used in the aerospace industry, are normally produced by thermal curing of epoxies. Ambient temperature production by radiation processing of equivalent acrylated epoxy matrices can be used to produce composites with reduced residual stresses and without any substantial emission of volatile compounds. Radiation induced degradation of materials as a result of bond scision can be exploited to process pulp and viscose. At AECL Research, studies of radiation processing of wood chips prior to pulping have shown that, in certain cases, pre-irradiation of the wood chips facilitates the pulping process, resulting in a significant decrease in the overall energy required, without any significant impact on the quality of the pulp. Also at AECL Research, studies have shown that radiation processing can be employed to produce better wood-fiber and mineral-powder filled thermoplastics for a variety of applications.

Curing of oligomer/monomer mixtures with low energy electron beams (EB) has been studied at TRCRE, where the mechanical properties of EB-cured films of mixtures composed of aliphatic urethane-acrylate oligomers and mono- or multi-functional monomers are being studied. These studies led to the development of a series of urethane acrylate type oligomers, suitable for pressure sensitive adhesives. Finally, a novel application of ionizing radiation, studied at TRCRE, is grafting functional groups onto polymeric substrates to produce materials with desirable physicochemical properties. For example, battery separators are prepared by grafting acrylic acid monomers on to polyethylene substrates; also adsorbents for ammonia and amines have been developed by grafting sodium p-styrenesulfonate and acrylic acid on to non-woven fabrics.

EFFECTS OF HIGH RADIATION ENVIRONMENTS

ON POLYMER COMPOSITE EPOXIES

H. W. Bonin, H. M. Pak, V. T. Bui and D. Rhéaume

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ABSTRACT

High polymer epoxy adhesives are more and more used in an increasing number of domains such as the naval and aerospace industries. In addition to excellent adhesion and high strength properties, they exhibit good weather resistance, low shrinkage upon cure, low toxicity, excellent resistance to corrosion and ease of use. Before using these high polymer epoxies in environments subjected to high gamma and/or neutron radiation, it is essential to acquire a good knowledge of their resistance to high fluences of ionizing radiation. Applications under such environments include composite materials in space vehicles and in the nuclear industry for nuclear reactor components or spent fuel storage containers.

In this work, the adhesive performance of a high polymer epoxy (a fast-setting 5-minute epoxy glue) was examined, resulting from various exposure durations to a high radiation (fast and thermal neutron, and gamma) flux produced by the SLOWPOKE-2 research reactor at Royal Military College of Canada. The tests used the ASTM D897 procedure on samples of epoxies subjected to various doses of radiation. These tests consisted in tensile strength measurements done on a fully computerized Instron Model 4206 tensiometer. The samples were made with two aluminum cylinders with flanges held together by a fine layer of epoxy glue prepared and applied in a very carefully controlled manner.

The first tests involved a batch of a dozen unirradiated samples which exhibited an average adhesive strength of 4 ± 2 kN. Batches of 12 similar samples were then irradiated in the SLOWPOKE-2 reactor pool using an "elevator" designed to bring and maintain samples against the reactor vessel at a position coincident with the mid-plane of the reactor core. Irradiation durations ranged from 1 hour to 24 hours, giving neutron doses calculated as ranging from 3 ± 2 mGy to 130 ± 50 mGy. The gamma ray doses were estimated as from 11 ± 5 mGy to 2.5 ± 1 Gy. The tensile strength tests performed on the most irradiated samples gave tensile strengths of more than 9 ± 1 kN. The graphs of the tensile strength versus irradiation times exhibited a sharp rise after three hours of irradiation, with the reactor at half-power, producing a thermal neutron flux of 5×10^{11} n/cm^2-s at an inner irradiation site (or 10kWth power).

Additional experimentation was performed in order to understand this unexpected tensile strength increase, in terms of damage mechanisms. Examination of both unirradiated and irradiated samples was carried out using methods such as neutron activation analysis and Fourier Transform Infrared spectroscopy, among others. These analyses reveal that the epoxies are affected by radiation even at relatively low fluences, and that the main effects are the modification of chemical bonds within the molecules into some complex cross-linking. The next phases of this research to be carried out include longer irradiations and selective exposures to either neutrons alone (thermal or fast) or to gammas alone, aiming at determining the effects of each type of radiation.

SUMMARY

SOME CALCULATIONAL RESULTS FOR TRANSMUTATION OF PLUTONIUM AND WASTES IN BLANKETS OF ACCELERATOR-BASED SYSTEMS

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Some calculational results of plutonium, fission products and Np-237 transmutation are presented.

Modification of computer code TRIFON , originally developed for reactor cell calculations, taking account of external source and zero boundary conditions with some iteration procedure for determination of flux distribution and the level of subcriticality, has been used for calculations with materials smeared inside cylindrical zones. Neutron balance relation and relation between power of subcritical system and multiplication factor were taken into account.

The first facility is accelerator/target/blanket system with heavy water blanket of ATW type. Weapons plutonium is loaded inside CANDU-type pressure tubes, fission product Tc-99 is dissolved in the moderator, surrounding the tubes. The level of subcriticality is controlled by continuous removing of Tc-99 dissolved in heavy water of blanket bulk. The loading of Pu is about ~ 300 kg. In the 2nd Case Pu loading is 5 times lower.

The value $k_{ef} = 0.97$ chosen leads to rather moderate accelerator parameters - for 1.6 GeV accelerator and Pb target proton beam current is 16 mA. With 300 loading of Pu and 3 tons of Tc-99 dissolved in heavy water the productivity of installation is

558 kg/year Pu fissioned

260 kg/year Tc-99 destroyed (converted to stable species).

For 5 times lower Pu loading with 1 ton Tc-99 dissolved Pu fission rate is the same (due to the same thermal power output), but transmutation rate of Tc-99 decreases to 89 kg/year due to higher relative rate of neutron captures in structural materials to fissions and increased leakage of neutrons.

In the 3d Case transmutation of Np-237 in two-stage process (involving Np-238 fission) is considered in a high-flux heavy-water blanket. Low level of criticality ($k_{ef} \approx 0.6$) were chosen to avoid overheating due to high fission rate, with low fissionable materials loading. The level of flux obtained in this system 3×10^{15} 1/cm^2 enables Np-238 fissions to be involved in transmutation process. About 485 kg of Np-237 are eliminated per year with 350 kg/year disappeared due to fissions of Np-238.

In the last - 4th Case - transmutation of Np-237 in a fast neutron spectrum blanket (thermal power output 0.87 Gwt) was considered. The amount of Np-237 transmuted is about 500 kg/year with 190 kg/year fissioned due to fission threshold reactions. Many of Np/Pu/Am/Cm isotopes are good fissionable materials in this Case.

WEDNESDAY OCTOBER 6

8:30-10:00 Session C21: Human Factors 1
Peel Room
Chaired by: L. Innes (Atomic Energy Control Board)

C21.1 *Coping with Human Factors in Nuclear Power Plants*
by J.-P. Clausner (OECD Nuclear Energy Agency)

C21.2 *Framework for Human Factors Input to Design Projects*
by J. Penington (AECL Research, CRL)

C21.3 *A Tool to Assist in Plant Data Monitoring and Diagnostics*
by P.D. Thompson and M.K. Gay (New Brunswick Power) and C. Xian and J.W. Thompson (Atlantic Nuclear Services)

C21.4 *Performance Support Systems and Artificial Intelligent Considerations*
by W.F.S. Poehlman, W.J. Garland, A. Bokhari and C.W. Baetsen (McMaster University) and R.J. Wilson (EACS - Engineering and Computing Services)

COPING WITH HUMAN FACTORS IN NUCLEAR POWER PLANT

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A paper for presentation at the **INC 93**
International Nuclear Congress and Exhibition
Toronto, Canada, October 3-6, 1993

Summary

In the wide area of Operating Experience and Human Factors, the Nuclear Energy Agency (NEA) which is one of the fifteen bodies that make up the Paris based Organisation for Economic Cooperation and Development (OECD), has devoted growing activities in the specific field of Human Factors. The Committee on the Safety of Nuclear Installations (CSNI) which is one of the standing Committees of the NEA, through its Principal Working Group on Operating experience and Human Factors (PWG1), has been particularly active, initiating studies and specialist meetings in the field of Human Factors.

Numerous studies have been and continue to be undertaken to improve the understanding of the various aspects of human behaviour. The purpose of these studies is to enhance nuclear power plant safety through improvements both in human performance and in man-machine interface. The CSNI, consequently, created, over the past ten years, several "task-forces" which carried out ten human factor-related studies on issues such as quantifying human behaviour, incidents involving cognitive errors, the use of digital computers in control rooms, regulatory approaches to maintenance practices at nuclear power plants, management of maintenance outages and the effects of advanced systems on operators.

The purpose of this paper is to give an overview of the OECD/NEA-sponsored activities in the area of human factors and present highlights of some of the conclusions that the NEA published.

FRAMEWORK FOR HUMAN FACTORS INPUT TO DESIGN PROJECTS - SUMMARY

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There is an increasing awareness of the impact that human behavior and human error has upon safety, and that the causes of accidents cannot be attributed solely to hardware failures. It is therefore essential that the human element is considered in the design of systems, to ensure that the design is compatible with operator requirements and is robust against operator error.

Atomic Energy of Canada Ltd. (AECL) have realized that working in the nuclear industry, which provides a potentially hazardous environment, fostering a commitment to the inclusion of Human Factors is essential in order to maintain high standards of safety and efficiency and provide a comfortable working environment. It is required that AECL demonstrates the incorporation of Human Factors into their projects in order to obtain and retain the site license. The regulators of the nuclear industry, the Atomic Energy Control Board (AECB), employ Human Factors assessors to monitor the acceptability of the inclusion of Human Factors in the design and modification of nuclear and associated facilities. The commitment of the AECB to Human Factors is demonstrated by the recent inclusion of Human Factors in their audit of the design process for AECLs new isotope producer, MAPLE X10.

Human Factors assessment methods are used to design systems, examine systems, specify features or identify deficiencies and suggest modifications which will improve the design from a human point of view. Human Factors should therefore be an integral part of the design process and the earlier in the design process these factors are considered, the safer and more economical the results will be. It is intended that in future AECL projects, a full-time Human Factors specialist will be dedicated to the project team.

The objective of producing a framework for the Human Factors input to projects is to provide the Project Manager and the Project Team with some indication of the scope and function of the work to be included and to ensure that Human Factors is adequately taken into account at all stages of the project. The paper will review each design stage to discuss the proposed scope of work.

The benefits of a comprehensive Human Factors program can be summarized as follows:

- ensures the safety of the operators, the public and the environment,
- ensures the operability of the systems by creating compatibility with the operators needs,
- improves the economics of the system by improving productivity, reducing down time and reducing the potential for injury,
- reduces the likelihood of costly modifications later in the system lifecycle, and
- ensures licensing requirements are met.

The project framework has been developed for future use within AECL and demonstrates their commitment to a safe and efficient working environment. Currently the framework is being applied to the design of the MAPLE X10 reactor, although the framework and associated Human Factors techniques were only developed in time to provide an input to the Design Definition stage.

A Tool to Assist in Plant Data Monitoring and Diagnostics

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At Point Lepreau Generating Station (PLGS), process data from approximately 2,500 transducers is measured every 6 seconds, and safety system data from approximately 250 transducers every 2 seconds. Additionally, the plant computers generate a variety of calculated parameters, using one or more of the "raw" signals as input. Until relatively recently this data was essentially transient, in that it was used, where required, for display purposes, or in control programs, and then discarded.

The recent installation of a data access system, referred to as the "Gateway", means that anyone on the PLGS computer network can have access to the data. The availability of large amounts of data means that reliable statistics may be derived for a wide range of data. By routine monitoring of these statistics, gradual deviations in transducer or process performance can be detected.

This paper describes an easy-to-use tool, the Plant Analysts Workbench (PAW) which can manage PLGS data in a straightforward manner. The short-term goal of the PAW project is to provide plant analysts with a tool that will enable them to easily manipulate large volumes of plant data, to deal with the idiosyncrasies of the data retrieval system, to present the data in a standard format, and to perform some relatively simple assessment functions on the data. This ability provides the safety analyst with the tools necessary to ensure that analytical plant models accurately reflect actual plant conditions.

In the longer term, the intent is to generate a suite of process parameter "statistical signatures", such that deviations from normal plant behaviour may be easily detected. Currently, examination of these statisitics is performed manually. The eventual aim is to develop an expert system to assess the statistics. It will be trained to identify normal plant behaviour, and isolate potential problem areas to the PAW operator.

Performance Support Systems and Artificial Intelligent Considerations

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and
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SUMMARY

Recent focus on intelligent performance support systems for reactor operators has resulted in several major reviews. Among top desirable features for these operator destined computer-based systems are human-centred design, intelligent behaviour and real-time performance. The philosophical approach of human-centred design, so necessary for the operator to be in control, is outlined and extended to include the concept of user mental models. Other fields apply artificial intelligent (AI) techniques to offer such approaches and operator companions for the nuclear industry seem also to be similarly amenable. Intelligent behaviour is extremely germane to these systems and includes two domains: (1)what information is to be conveyed to the operator under any given situation and (2)how that information can be optimally presented to the user to maximize data transfer and minimize the time required via the man-machine interface. AI can aid in the realization of these goals; however, such techniques are resource intensive and not easily adapted for real-time applications. The fundamental design principles of temporal and functional abstractions have given other knowledge-based control systems adequate response times for nuclear particle accelerators which are simpler systems but results seem promising for more complex problems. Other AI applications which appear fruitful to examine include intelligent and context sensitive help procedures (expert systems excel at explanatory applications) and the use of on-line and parallel AI paradigms for system validation to aid in licensing issues. Most of the above features are being either explored or implemented in an anthropomorphically designed, agent oriented approach and distributed architecture-based multi-computer system called the OPUS (OPerator / User Support) System. The overall approach is detailed and the impact of the methods to the field is considered.

WEDNESDAY OCTOBER 6

8:30-10:00 Session C22: Safety 4
Wentworth Room
Chaired by: D. Weeks (New Brunswick Power)

C22.1 *Reduction of Pressure-Tube to Calandria-Tube Contact Conductance to Enhance the Passive Safety of a CANDU-PHW Reactor*
by D.B. Sanderson, R.G. Moyer, D.G. Litke and H.E. Rosinger (AECL Research, WL), and S. Girgis (AECL CANDU)

C22.2 *Simulation of the Pressure-Tube Circumferential Temperature Distribution Experiments (Variable Make-Up Water Experiments)*
by M.H. Bayoumi and P.S. Kundurpi (Ontario Hydro), and W.C. Muir (IDEA Research International)

C22.3 *Liquid Relief Valve Failure Simulation in the Embalse Nuclear Power Station*
by S. Gersberg, J.R. Lorenzetti, D. Parkansky and J. Batistic (Comisión Nacional de Energía Atómica, Argentina)

C22.4 *Simulation and Analysis of a Main Steam Line Transient with Isolation Valves Closure and Subsequent Pipe Break*
by V. Stevanović and M. Studović (University of Belgrade, Yugoslavia) and A. Bratić (Thermal Power Plant Nikola Tesla-A, Yugoslavia)

REDUCTION OF PRESSURE-TUBE TO CALANDRIA-TUBE CONTACT CONDUCTANCE TO ENHANCE THE PASSIVE SAFETY OF A CANDU-PHW REACTOR

by

D.B. Sanderson*, R.G. Moyer*, D.G. Litke*, H.E. Rosinger*,
S. Girgis**

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SUMMARY

One of the ways to enhance the passive safety of a CANDU-PHW (CANada Deuterium Uranium - Pressurized Heavy Water) reactor is to reduce the moderator subcooling requirements during a postulated loss-of-coolant accident (LOCA). The increased moderator temperatures would enhance the heat transfer from the moderator to the surrounding shield tank during a postulated accident. This reduction in subcooling requirements can be achieved by incorporating a wire screen in the fuel channel annulus, right next to the calandria tube. The technique has been demonstrated to significantly reduce the moderator subcooling requirements so that the calandria tube was not forced into film boiling upon pressure-tube ballooning contact with 0°C subcooling outside the calandria tube.

An experiment was performed at AECL Research's Whiteshell Laboratories to investigate the changes in heat transfer characteristics between a pressure tube and a calandria tube with a wire screen placed in the fuel-channel annulus. This paper describes the results of this experiment. Results from computer simulations performed to assess the effect of the wire screen on the performance of a CANDU fuel channel during selected LOCA scenarios are also presented.

SIMULATION OF THE PRESSURE TUBE CIRCUMFERENTIAL TEMPERATURE DISTRIBUTION EXPERIMENTS (VARIABLE MAKE-UP WATER EXPERIMENTS)

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SUMMARY

In a number of postulated loss-of-coolant accident (LOCA) scenarios, some of the fuel channels are predicted to initially experience periods of stratified flow. During this time, the steam generated will flow to the top portion of the pressure tube and thereby expose the top part of the pressure tube to superheated steam. The exposed part of the pressure tube and fuel elements will heat up as a result of heat transfer by radiation and steam convection. This results in a circumferential temperature gradient around the pressure tube which could result in a nonuniform or localized pressure tube strain which could lead to failure of the pressure tube prior to contacting the calandria tube.

The pressure tube circumferential temperature gradient experimental program (PT-DELTA T) has been undertaken at AECL-WNRE under COG to investigate the potential of pressure tube rupture during ballooning. The experimental results were also used to validate the computer code SMARTT (Simulation Method for Azimuthal and Radial Temperature Transients) [1] which is one of the analytical tools used in the safety analysis of CANDU reactors. In the fourth test series, Variable Make-Up Water series, water is injected into the pressure tube at a controlled and declining rate to study the effect of a gradual decrease in make-up water flow rate and to simulate the decreasing density driving head as the liquid level in the feeder decreases.

The main objective of this paper is to discuss the simulation results of the first Variable Make-Up Water experiment using the SMARTT code and to compare with the experimental results for code verification. The SMARTT computer code predicts fuel sheath and pressure tube thermal and mechanical behaviour under asymmetric coolant conditions such as stratified or decreasing coolant flow in the channel. The code predicts the pressure tube circumferential temperature distribution and its effect on pressure tube ballooning. The code also predicts whether the pressure tube will rupture prior to contacting the calandria tube or balloon into contact with the calandria tube.

Two parameters are chosen to show the comparison between the SMARTT predictions and the experimental results, namely, the pressure tube and the heater sheath temperature transients. Excellent agreement was obtained between the SMARTT simulation and the experimental results for this experiment after adjusting the measured power to be consistent with the results of heat balance across the test section

REFERENCES

[1] K.E. Locke, et al, "SMARTT - A Computer Code to Predict Fuel and Pressure Tube Temperature Gradient Under Asymmetric Coolant Conditions", Proc. 6th Annual CNS Conference, Ottawa, June 1985.

LIQUID RELIEF VALVE FAILURE SIMULATION IN THE EMBALSE NUCLEAR POWER STATION

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Simulations of failure (open) of a liquid relief valve in the Embalse Nuclear Power Station heat transport system have been performed using the Firebird III computer code.
As only one loop was simulated (1 circuit model) we have made some assumptions to take into account the influence of the pressurizer and the unbroken loop.

- We have assumed that the two loops depressurize at the same rate.
- Since the unbroken loop compensates the loss of inventory in the broken loop, the discharge rate flow through the liquid relief valve was reduced to half of its calculated value. However, this assumption was not considered when the degasser condenser mass and energy balance was performed.

Results show that, without any operator action , and if the degasser " bottles up" (closure of a valve downstream of level control valves), the heat transport and the degasser pressure increase, reaching the relief degasser condenser valve opening.
To avoid this loss of inventory, three operating procedures have been analyzed:

1) An initial reduction of power (0.5%/s).
2) The same power reduction following the pressurizer level recovery.
3) A trip when the pressurizer level falls below 7.5 m.

Results of the simulations indicate that the most convenient action is to reduce power when the pressurizer level stops decreasing.

SIMULATION AND ANALYSIS OF A MAIN STEAM LINE TRANSIENT WITH ISOLATION VALVES CLOSURE AND SUBSEQUENT PIPE BREAK

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Simulation and analysis of a real main steam line break transient at the coal fired, 300 MWe Thermal Power Plant Drmno are presented. Prior to the incident, the Plant had been operating at nearly full power with a steam leakage observed acoustically. Main events of the transient were the closure of isolation valves in front of the high pressure turbine, the opening of the by-pass line, and 20 seconds later a pipe break in front of one isolation valve. Intensive pressure waves were generated and they propagated through the pipe network of the steam line, causing high fluid dynamic forces on structure. The transient has been simulated by the computer code TEA-01, based on the Method Of Characteristics with three characteristics directions.

Several main steam line boundary conditions have been modeled and verified. The main steam line has been modeled according to the actual plant geometry, while the steam boiler has been represented as one node with heat source and corresponding in and out flows. In order to derive the sensitivity of the systems behavior to the instants of the isolation valve action, to the time duration of the valve closure, and to the by-pass system's action, as well as to obtain the most conservative case of the possible system parameters change during transients, various modeling scenarios were prepared, for the following time intervals:

- less then one second in order to simulate the pressure waves propagations and transient fluid dynamic forces during and after the isolation valve closure;
- a few seconds in order to predict the global pressure in the steam lines and steam boiler after the isolation valves closure; and
- a few minutes in order to simulate the system's blowdown after the pipe break. Results show pressure wave propagations, global pressure change, steam flow, and hydraulic forces on the part of the steam line where the break took place, during the valve closure and during the steam blowdown. They are used for the prediction of the dynamic structural response of the main steam pipeline. Some of the numerical results are compared with plant data logger records.

The described simulation methods and procedures could be applied in the analysis of the similar abnormal transient at the main steam lines and steam generators of a nuclear power plant.

WEDNESDAY OCTOBER 6

10:30-12:00 Session C23: Physics 2
Huron Room
Chaired by: P. Akhtar (Atomic Energy Control Board)

C23.1 *Flux Mapping Theory Application for Channel Power Prediction*
by D. Brissette and M. Beaudet (Hydro-Québec, Gentilly)

C23.2 *A Review of the History-Based Local-Parameter Methodology for Simulating CANDU Reactor Cores*
by B. Rouben and D.A. Jenkins (AECL CANDU)

C23.3 *Validating the History-Based Diffusion Methodology for Core Tracking Using In-Core Detectors*
by A.C. Mao, B. Rouben and D.A. Jenkins (AECL CANDU), and E. Young and C. Newman (New Brunswick Power)

C23.4 *On-Line Heat Deposition Rate Measurements with a Quasi-Adiabatic Graphite Calorimeter in a Fusion Environment and Comparison with Calculations*
by O.P. Joneja and J.-P. Schneeberger (École Polytechnique Fédérale de Lausanne, Switzerland), R.P. Anand (Bhabha Atomic Research Centre, Bombay, India) and T. Buchillier (Institute of Applied Radiophysics, Lausanne, Switzerland)

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FLUX MAPPING THEORY APPLICATION FOR CHANNEL POWER PREDICTION

by

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SUMMARY

The summary was not available at the time of printing.

A REVIEW OF THE HISTORY-BASED LOCAL PARAMETER METHODOLOGY FOR SIMULATING CANDU REACTOR CORES

by
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SUMMARY

The history-based local-parameter methodology represents a fundamental refinement in the evaluation of nuclear lattice properties in core-tracking simulations for CANDU reactors. This paper reviews the basis, development, and advantages of the history-based methodology.

The standard method of evaluating lattice properties for reactor-core neutronics calculations is by interpolation in tables of nuclear cross sections versus irradiation. The calculation of these "fuel tables" is normally performed with some basic simplifying assumptions. For instance, certain cell parameters – such as flux level, fuel temperature, coolant density, etc. – are given "effective" uniform or core-average values. These approximations overlook the spatial variation of these parameters, which can translate to significant differences in lattice properties between different fuel bundles at the same irradiation but at different locations in core.

A local-parameter methodology is one which attempts to reduce these approximations by building into the reactor model lattice properties appropriate to local conditions. The history-based method is the latest advance in local-parameter methodologies for CANDU reactors. It was designed to work in the context of core tracking, where simulations to follow the core history are performed typically at intervals of a few Full Power Days (FPD).

The totally new feature of the method is that fuel tables are <u>not</u> calculated. Instead, a cell calculation is carried out for every single fuel bundle in core over the small irradiation increment associated with the specific bundle and the core-tracking time step simulated (say ~3 FPD).

Each cell calculation therefore need not individually span the entire irradiation range experienced by a bundle over its lifetime, whereas a fuel-table calculation must. This makes the initiative of following each fuel bundle's history with its own specific cell calculations manageable when use is made of an appropriately trimmed (but logically complete) version of POWDERPUFS-V, the empirical, recipe-based lattice code developed for application to the CANDU heavy-water moderated lattice. The use of the history-based method with other lattice codes will be tied to the availability of fast-running versions of the codes and further advances in computer speed.

The history–based methodology possesses many advantages which allow more realistic and more accurate modelling of the neutronics of the core:

- Any number of cell parameters can be customized to the lattice calculations for each specific fuel bundle.
- The fact that each cell calculation spans only a small irradiation step allows the modelling of <u>changes</u> in local conditions during a bundle's residence time. For instance, changes in power, moderator poison concentration, etc. can be modelled as they occur. Thus, they affect lattice properties at the proper time in the bundle's history, <u>not</u> for the entire residence time of the bundle. (This is the source of the "history-based" label for this methodology.)
- "Instantaneous" core perturbations – e.g., power manoeuvres, addition or removal of moderator poison, loss of coolant, etc. – can be modelled more accurately.
- Since a cell calculation is performed for each bundle in any case, bundle-specific concentrations of the saturated fission products (e.g., Xe-135, Rh-105, Sm-149, . . .) are easily calculated. A fission-product driver coupled to the history-based methodology allows steady-state, transient, and long shutdown concentrations of these fission products to be calculated and incorporated in the lattice-property calculation. This is very much more difficult to do when working with fuel tables.

The history-based method has been implemented and is operational on the standard CANDU core-tracking program, Reactor Fuelling Simulation Program (RFSP).

VALIDATING THE HISTORY-BASED DIFFUSION METHODOLOGY FOR CORE TRACKING USING IN-CORE DETECTORS

by

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SUMMARY

A refinement of the three-dimensional diffusion reactor fuelling simulation program for CANDU reactors was developed. This program refines the lattice properties which take into account the past burnup history of each individual bundle. This past burnup history includes fuel temperature, coolant temperature, coolant density, burnup, flux or power level, as well as saturated fission-product concentration in each fuel bundle.

The purpose of this paper is to validate the history-based local-parameter method for core tracking. This method was applied to the core-tracking simulations of the Point Lepreau Generating Station for a period of approximately 18 months from 1991 September 30 to 1993 April 29. The validation of the method was performed by comparing calculated flux shapes to core fluxes measured by means of in-core vanadium detectors. The results were compared with those obtained from a conventional, i.e., non-history-based, calculation. In general, the maximum channel power is about 1.5% lower in history-based calculations, while the maximum bundle power is about 4.5% lower, and the maximum channel power ripple is about 1% lower. The standard deviation differences between calculated and measured fluxes obtained from the history-based simulation is significantly improved compared to that from the conventional method.

ON-LINE HEAT DEPOSITION RATE MEASUREMENTS WITH A QUASI-ADIABATIC GRAPHITE CALORIMETER IN A FUSION ENVIRONMENT AND COMAPRISON WITH THE CALCULATIONS

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Summary

The energy deposition profile in a (d,t) driven fusion system is expected to be spread out, because most of the reaction energy is carried away by the fast neutrons. These highly penetrating neutrons undergo host of nuclear reactions in a fusion system and therefore for any accurate estimation, a precise knowledge of neutron transportation and energy deposition due to all the possible reactions is required. A direct heating rate measurement by a calorimeter in a fusion system would provide a valuable data for comparison with the calculations, and the necessary confidence in designing and constructing the ambitious fusion machines such as ITER, NET and FER.

Measurements were carried out by a calorimeter consisting of four graphite zones, separated from each other by small vacuum regions. In order to insulate the inner regions from any environmental temperature fluctuations, a double shield arrangement is employed. The calorimeter is operated in the quasi-adiabatic mode i.e the difference in temperature of the core and the jacket is always kept constant through out the period of measurement. The temperature of various regions is measured by micro-thermistors wired with platinum. The central core region of the calorimeter measures 16 mm in dia and 3 mm in height and the overall dimensions are 34 mm dia and 18.1 mm height.

The heating rate for 300 mA beam current and 165 KV operating voltage of the generator was found to be 30.48 μw/g $\pm$ 3.4%. The experimental reproducibility was better than 1%. The main source of errors comes from the calibration of the calorimeter (2.5%) and the error due to the beam current variation was $\leq$ 1%. The value obtained by Monte-Carlo calculations for the same power level of the generator was 31.99 μw/g $\pm$ 2.1 %. The gamma heating constitute 26 % of the total. The main source of error is from the determintaion of source strength (2 %) and a statistical accuracy of 0.9%. The C/E was found to be 1.05. Experiments are underway to determine heat deposition profile inside a large block of graphite as well as for different fusion materials.

WEDNESDAY OCTOBER 6

10:30-12:00 Session C24: Life Extension
Kenora Room
Chaired by: K. Talbot (Ontario Hydro)

C24.1 *Planning the Retubing of a CANDU 6 Reactor*
by N.G. Craik (Canatom Inc.) and R. Baker (New Brunswick Power)

C24.2 *Improvements in Remote Removal Techniques for Active Components during Large-Scale Retubing of CANDU Reactors*
by W.J. Knowles (GE Canada Inc.)

C24.3 *Developments in Orbiting Tools for Refurbishment of CANDU Fuel Channel Components*
by M. Pollock (Spectrum Engineering Corporation Ltd.)

C24.4 *CANDU Single-Fuel-Channel Replacement Reducing Time and Radiation Exposure*
by T.A. Hunter and D.R. Pollock (GE Canada Inc.)

PLANNING THE RETUBING OF A CANDU 6 REACTOR

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SUMMARY

The reactor fuel channel pressure tubes at the New Brunswick Power Corporation CANDU 6 Nuclear Generating Station at Point Lepreau, are known to have a number of design related mechanisms which will limit their performance life.

Most of these mechanisms are of concern only near the end of the nominal 30 year life of the pressure tubes. However, one mechanism, caused by movement of spacers away from their correct design position, is pressure tube to calandria tube contact which may result in cracking of the pressure tube material. It is estimated that this could be a significant operating problem as early as 1998.

In 1993, utilizing the latest techniques in pressure tube inspection and maintenance technology, both Ontario Hydro and NB Power demonstrated for the first time that pressure tube to calandria tube contact could be eliminated without pressure tube replacement. This process, known as Spacer Location and Relocation (SLAR) offers an alternative to retubing and is being considered by NB Power for implementation in 1995.

Consequently, NB Power has been studying Large Scale Fuel Channel Replacement (LSFCR) either for implementation in 1998 or 1999 as a fallback position to SLAR or in 2008 when the pressure tubes will require replacement for other reasons. Either strategy would have the objective of achieving a 40 to 50 year plant life.

LSFCR has been undertaken at the four Ontario Hydro Pickering A Reactors. Plans for retubing the four Ontario Hydro Bruce 'A' units are well advanced. However, this is the first time that the methods of undertaking an LSFCR on a Candu 6 have been studied in any detail. The initial focus of the Lepreau LSFCR planning is to see how the equipment and facilities similar to that used in the Pickering and Bruce LSFCR operations, can be applied to Lepreau.

NB Power prepared a 3D CAD's model of areas that will be involved in LSFCR and in the movement of large/long components in and out of both east and west fuelling machine vaults, from the main airlock, through the reactor building, including existing cranes.

NB Power's studies on a method of LSFCR of the Lepreau 1 Candu 6 show that the reactor building and facilities are significantly different to the Pickering and Bruce layouts and present some major problems. This paper describes potential solutions to these layout problems.

* Canatom Inc.

IMPROVEMENTS IN REMOTE REMOVAL TECHNIQUES FOR ACTIVE COMPONENTS DURING LARGE SCALE RETUBING OF CANDU REACTORS

by

W.J.Knowles

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ABSTRACT

This paper describes the evolution of the "On Face" tooling and the techniques used for the remote removal of the radioactive pressure tubes and end fittings from CANDU reactors.

Starting with a brief description of Pickering units 1 and 2 techniques, the improvements for units 3 and 4 are discussed together with a forward look at the techniques being developed for the Bruce A reactors.

The "P1/P2" tooling design was done on a crash basis with little development time available for sophisticated tooling. The resultant tooling was adequate but did not allow for techniques that could reduce cycle time and manrem reduction.

The greater lead time afforded for the P3/P4 work enabled more effective tooling to be developed. Two tools, the End Fitting Removal Tool and the Pressure Tube Push Tool along with their operation are described. These new tools and procedures enabled significant gains in productivity and manrem reduction to be made. The improved statistics are reviewed in the paper.

The Bruce reactor channel is different to that at Pickering but similar tooling can be used. The main difference is that emphasis has been placed on increasing the amount of remote working that can take place before local manual intervention is required for further setup work. This is achieved by the use of remotely controlled shield plug handling machines and the ability to handle multiple end fitting removal work stations. This equipment is briefly described.

DEVELOPMENTS IN ORBITING TOOLS FOR REFURBISHMENT OF CANDU FUEL CHANNEL COMPONENTS

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ABSTRACT

Ontario Hydro's Bruce Nuclear Generating Station "A" is to undergo a Large Scale Fuel Channel Replacement (LSFCR), more commonly called a "retube", in order to extend reactor life.

One of the primary operations of the retube involves precisely severing the End Fittings (E/F) from their attachment points to the tube sheet on the face of the reactor. The area to be cut is a weld on the outboard side of the expansion bellows assembly or the east side and through a thin steel shell (stop collar assembly) on the west side. Once the cutting and other related operations are complete, the old E/F is removed and a new one is installed and welded at the location of the cut. Access is difficult because of the physical restrictions of surrounding lattice sites, feeder pipes and the obstructions on the target E/F itself.

The paper discusses the development of specialized cutting tool used to remove existing components, cleaning and welding equipment for replacement components and the orbiting tools use to support and rotate the tools around the component. Design criteria for the tooling include quick and easy assembly by personnel with limited training, light weight for ease of handling, low radiation dose and high reliability. It includes discussions of the various options considered, the reasons for the chosen solution and the operation of the tools.

The development of the above tooling has evolved based on the experience from previous retube, rehabilitation and single channel replacement operations on various Ontario Hydro reactor All tooling has been developed using the procedures laid out by Ontario Hydro's "Program of Quality Engineering for Fuel Channe Replacement Tooling and Equipment".

CANDU Single Fuel Channel Replacement

Reducing Time and Radiation Exposure

by

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There has been a large number of single fuel channel replacement (SFCR) projects on Candu reactors, particularly at Pickering, Bruce and Darlington Generation Stations. To date there has been just one replacement on the 600 MW reactors and that was at Point Lepreau.

GE Canada has played an active role in fuel channel replacement work starting with Canada's first power reactor, NPD. With the most recent project on Darlington Unit 2, GE Canada supplied the replacement tooling and supervised the replacement work. At Point Lepreau a crew of 9 supervisors and technicians was supplied to train the workers and direct the work.

With ever increasing replacement costs there is an ongoing incentive to improve replacement tooling and procedures to reduce reactor down time and radiation exposure. This paper outlines improvements made on several tools including feeder blanking plates, feeder winches, television cameras, end fitting angle measuring tools, feeder bolt removal tools and reactor face shielding.

The paper also discusses the latest SFCR concepts and procedures as used recently to successfully replace two fuel channels on Darlington Unit 2. Emphasis was on a trained dedicated crew with well defined responsibilities and with efforts well recognized. In each case only about 75 hours of in-vault time were required for the actual replacement work in the interval between installation and removal of the maintenance platforms. This represents a reduction in the amount of time traditionally used to replace fuel channels. The paper gives ideas which can lead to further improvements.

WEDNESDAY OCTOBER 6

10:30-12:00 Session C25: Waste Management 2
Kent Room
Chaired by: J. Graham (British Nuclear Fuels Ltd.)

C25.1 *Progress at AECL Research's Underground Research Laboratory*
by G.R. Simmons (AECL Research, WL)

C25.2 *The Role of Engineered Barriers in the Disposal of Nuclear Fuel Waste - The Canadian Perspective*
by K. Nuttall and L.H. Johnson (AECL Research, WL)

C25.3 *Sulfur Polymer Cement, A Solidification and Stabilization Agent for Radioactive and Hazardous Wastes*
by G.R. Darnell (INEL-EG&G Idaho, Inc.)

C25.4 *The Cigar Lake Analog Study: An International R&D Project*
by J.J. Cramer and F.P. Sargent (AECL Research, WL)

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SUMMARY
PROGRESS AT AECL'S UNDERGROUND RESEARCH LABORATORY

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The Underground Research Laboratory (URL), a research and development facility, has been constructed by AECL Research near Pinawa, Manitoba for the Canadian Nuclear Fuel Waste Management Program (CNFWMP). The CNFWMP is developing the technology to engineer the safe and permanent disposal of Canada's nuclear fuel wastes. A formal public review of this technology is now under way as part of the Federal Environmental Assessment and Review process.

During the development of the CNFWMP in the 1970s, the need for an underground facility to conduct large-scale multidisciplinary experiments and demonstration tests was recognized. An underground research laboratory at a reasonable depth with a vertical access shaft was selected. The initial phases of the URL Project were Site Evaluation and Construction.

The Site Evaluation Phase, a comprehensive surface-based geotechnical characterization of the URL, began in 1980 with regional reconnaissance studies and then more detailed site characterization of the land leased for the URL. Monitoring of many of the instruments installed during Site Evaluation will continue at least until the end of URL operation.

The Construction Phase began with construction of the shaft collar in 1983 and ended in 1990 with completion of underground access and services to all levels. Particular emphasis was placed in the Construction Phase on the development and demonstration of characterization and testing methods applicable to exploratory excavation and to disposal vault characterization during construction and operation. A full underground geotechnical characterization program was conducted during this phase. Combining the geotechnical activities, many of which were of an R&D nature, and the construction activities in an effective and safe manner was one of the major challenges. A project management and control system was developed during the Construction Phase to manage this integrated project effectively.

The URL currently comprises surface support facilities, a vertical shaft to a depth of 443 m, small shaft stations at depths of 130 m and 300 m, and major testing levels at 240 m and 420 m. A small ventilation shaft connects the 420 Level, the 240 Level and ground surface. Approximately 80 to 100 AECL and contract staff work on projects in the URL directed to investigating issues relevant to nuclear fuel waste disposal.

The present phase, the Operating Phase, began in 1989 and will continue until 2011. AECL's Operating Phase activities have been consolidated into nine multidisciplinary studies and experiments, which address the following subjects:

Characterization Methods Development and Demonstration,
Solute Transport Studies,
Excavation Response Studies,
Vault Sealing Studies, and
Disposal Vault Emplacement Room Simulations.

Of the nine planned activities, seven are under way and two are scheduled for implementation in the future. The status of the seven activities that are in progress are discussed in the paper.

The URL and AECL's studies have attracted the interest of national and international organizations and agencies. Several of these have either arranged to have specific tests conducted in the URL or to participate in AECL's planned activities.

As the experiments and studies at the URL have progressed, we have learned much that is essential to showing the feasibility and safety of AECL's concept for nuclear fuel waste disposal. We have demonstrated many aspects of geotechnical characterization from the surface and underground and are now conducting studies into specific issues of rock strength, rock failure, solute transport and excavation disturbance. Specific studies in the area of vault sealing technology are in progress and planned. As well, the expertise that has been developed in the URL project is contributing to the preparation of documentation for the disposal concept review and will contribute to the regulatory and public review process.

THE ROLE OF ENGINEERED BARRIERS IN THE DISPOSAL OF NUCLEAR FUEL WASTE - THE CANADIAN PERSPECTIVE

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SUMMARY

The concept developed in Canada for disposing of nuclear fuel waste is disposal at a depth of 500-1000 m in plutonic rock of the Canadian Shield. The waste would be isolated from the biosphere by a multi-barrier system consisting of engineered components supplementing the natural containment potential of the host-rock. The key engineered barriers are the waste form itself (used Candu fuel), the container that holds the waste form, and sealing materials used to envelop the container and to backfill and seal the vault excavations.

During the past fourteen years, AECL has carried out a comprehensive research program on the design, materials selection, and performance assessment of engineered barriers. The research has recognized the generic nature of the concept development phase of the program whilst taking account of the key parameters characteristic of the plutonic rock disposal environment. The research approach has included studies of underlying processes, large-scale component testing and demonstration, conceptual engineering, model development and the study of natural analogs. On the basis of the research we have:

- developed a number of container designs that meet the primary structural requirements that will exist in a disposal vault.
- demonstrated that both copper and titanium can provide long-term containment of the waste; depending on material choice and container wall thickness, lifetimes between 10^3 and 10^6 years are considered achievable.
- demonstrated that intact used fuel bundles are a highly durable waste form from which most of the radionuclides will be released at a slow rate controlled by the low solubility of the UO_2 fuel matrix.
- demonstrated that cement- and clay-based materials can provide effective sealing of excavated openings and a physico-chemical environment that ensures that radionuclide transport between the container and the host rock occurs primarily by diffusion rather than by groundwater flow.

The results have demonstrated that there is considerable flexibility in the selection of materials and designs for engineered barriers for a disposal system in plutonic rock that could meet potential constraints that might be imposed by conditions at an actual disposal site.

SULFUR POLYMER CEMENT,
A SOLIDIFICATION AND STABILIZATION AGENT FOR
RADIOACTIVE AND HAZARDOUS WASTES[a]

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SUMMARY

There are no known perfect solidification and stabilization agents for radioactive or hazardous wastes, so the search continues for individual agents for specific wastes. The U.S. Department of Energy began testing sulfur polymer cement (SPC) as a radioactive and hazardous waste solidification and stabilization agent because of its unusual properties. SPC is a sulfur polymer composite material that begins melting between 110 and 120°C (230 and 248°F), with an optimum pour temperature between 130 and 140°C (266 and 284°F). The compressive strength of SPC upon cooling averages 27.6 MPa (4,000 psi). Its mechanical strengths continue to increase for at least two years to approximately triple the original strength. As a proven construction concrete, SPC has demonstrated the ability to survive for years in acids and salts that destroy or severely damage hydraulic concretes in months or even weeks.

Perhaps SPC's strongest selling point is that it will always melt and pour at approximately 135°C. This feature will allow hazardous or radioactive waste specimens that do not pass the required tests to be remelted and reformulated until they do pass.

Heavy loadings (5 wt%) of the eight toxic metals have been combined individually with SPC and 7 wt% sodium sulfide nonahydrate ($Na_2S \bullet 9H_2O$). The leach rates for mercury, lead, chromium, and silver oxides were reduced by six orders of magnitude, while arsenic and barium were reduced by four. With ever increasing emphasis on high-temperature treatment of radioactive and hazardous wastes, the ability of SPC to stabilize incinerator ash and its volatilized toxic metal contents is encouraging.

All SPC used in tests to date was formulated for the construction industry. The preceding tests and many others conducted in other countries like Denmark, Japan, France, Germany, and the Netherlands strongly suggest that SPC shows great promise for further development. Ion-exchange resins that failed miserably in routine SPC testing passed in excellent fashion when the temperature and duration of heating was increased. An additive dramatically improved the leach resistance for toxic metals. The search is just beginning for different configurations of SPC that can accommodate higher loadings of various difficult-to-stabilize wastes.

a. Work supported by the U.S. Department of Energy, Assistant Secretary for Environmental Restoration and Waste Management, under DOE Idaho Operations Office, Contract No. DE-AC07-76ID01570.

THE CIGAR LAKE ANALOG STUDY: AN INTERNATIONAL R&D PROJECT

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The Cigar Lake uranium deposit is located in northern Saskatchewan, Canada. The 1.3-billion-year-old deposit is located at a depth of about 450 m below surface in a water-saturated sandstone at the unconformity contact with the high-grade-metamorphic Canadian Shield. The uranium mineralization, consisting primarily of uraninite (UO_2), is surrounded by a clay-rich halo in both sandstone and basement rocks, and remains extremely well preserved and intact. The average grade of the mineralization is 8 % U with locally grades as high as 55 - 60 wt.% U.

The Cigar Lake deposit has many natural features which parallel the features being considered within the Canadian concept for disposal of nuclear fuel waste. The study of these natural structures and processes provides valuable insight toward the eventual design and site selection of a nuclear fuel waste repository. The main feature of this analog is the absence of any indications on the surface of the rich uranium ore 450 m below. This indicates that the combination of natural barriers has been effective in isolating the uranium ore from the surface environment. More specifically, the deposit provides analog information relevant to the stability of UO_2 fuel waste, the performance of clay-based barriers, radionuclide migration, colloid formation, radiolysis, fission product geochemistry and general aspects of water-rock interaction. The main geochemical studies on this deposit focus on the evolution of groundwaters in the deposit and on their redox chemistry with respect to the uranium, iron and sulphide systems.

AECL, through generous cooperation from the owners of the Cigar Lake deposit, has conducted analog studies since 1984. International participation started in 1989 through collaboration with the Swedish Nuclear Fuel and Waste Management Company (SKB) and, more recently, with the Los Alamos National Laboratory (LANL). The studies are aimed at providing data and information for use in the safety assessment of the Canadian, Swedish and United States disposal concepts.

The geochemistry of the naturally occurring radionuclides ^{99}Tc, ^{129}I and ^{239}Pu in the deposit is being investigated. This study includes the calculation and measurement of the production of these radionuclides in the high-grade ore, and the measurement and modelling of their mobility within the deposit using the unique facilities at LANL.

This paper summarizes the results of these studies and their value in building confidence in the geological disposal concept for nuclear fuel waste.

WEDNESDAY OCTOBER 6

10:30-12:00 Session C26: Accelerators & Industrial Radiation 2
York Room
Chaired by: A. Stirling (AECL Accelerators)

C26.1 *Nuclear Data for Feasibility Assessment of HLW Transmutation*
by M.A. Lone, P.Y. Wong, W.J. Edwards, and R. Collins (AECL Research, CRL)

C26.2 *Neutron Activation Analysis of Mortars from Stone Houses Built in Canada During the French Régime*
by H.W. Bonin, C. Bordeleau, J.R.M. Boulé and M.A.T. Lapointe (Royal Military College of Canada)

C26.3 *Handled Gamma Quant Generator in Density Measurement*
by A. Mozelev (Small Scale Research & Production Company RADICAL, Dubna, Russia)

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NUCLEAR DATA FOR FEASIBILITY ASSESSMENT OF HLW TRANSMUTATION

by

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Over the years, many options for managing the HLW have been proposed. One of these suggests partitioning the long-lived radionuclides from the HLW and converting them to short-lived or stable nuclides by neutron-induced nuclear transmutation (P-T). With transmutation the effective half-life of a radionuclide is $T_{1/2}^{eff} = T_{1/2}^{nat} / (1 + 1.44\, \sigma\, \varphi\, T_{1/2}^{nat})$, where σ and φ are, respectively, the effective cross section and the neutron flux. The objective of the P-T option has been to reduce the long-term (> 1000 years) hazard of HLW to a level low enough to permit disposal in a repository requiring medium-term integrity and institutional control.

For maximum effectiveness, the P-T option must reduce the inventory of both long-lived fission products and higher actinides. The cost of waste transmutation must be a modest increment on the cost of nuclear power generation, and any increase in the short-term radiological risk should not negate the expected long-term benefit. Thus an effective P-T option would constitute an advanced fuel cycle requiring significant developments in techniques for fuel reprocessing and partitioning, and the generation of economical sources of neutrons for transmutation of radionuclides.

The transmutation options have been reviewed by many agencies. Over the last couple of years, there has been a resurgence in activity in this field. The nuclear data committees of IAEA and NEA that coordinate the evaluation of data for nuclear-power-related technologies have encouraged international cooperation. Improved data is needed for spallation reactions that could be used to generate high fluxes of neutrons, the transmutation cross sections of radionuclides, and the neutron-induced reaction cross sections used to assess radioactivity and radiation damage. The data on transmutation cross sections of radionuclides need to be examined carefully, since there are often few corroborative measurements and in some cases there is a large spread in the reported values. One such case is ^{90}Sr, where two measurements had reported a thermal-neutron-induced transmutation cross section of 800 mb and 14 mb.

Total neutron yields from spallation reactions in lead at proton energies between 0.4 and 2 GeV were investigated theoretically. At 1.6 GeV the integral neutron yield from a 100 cm diameter, 100 cm long lead target is calculated to be in the range of 46 to 57 n/p, depending on the level density parameter used in the evaporation model of the neutron emission. These yields can provide high neutron fluxes for transmutation of actinides and the radiologically significant fission products ^{99}Tc and ^{129}I. The thermal-neutron induced transmutation cross section of ^{90}Sr was measured at the NRU reactor with an activation technique. The value obtained is 9.7 ± 0.7 mb. This low value rules out the feasibility of thermal-neutron-induced transmutation of this radionuclide.

NEUTRON ACTIVATION ANALYSIS

OF MORTARS FROM STONES HOUSES

BUILT IN CANADA DURING THE FRENCH REGIME

H. W. Bonin, C. Bordeleau, J.R.M. Boulé and M.A.T. Lapointe

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ABSTRACT

The mortar used in the construction of stone houses and buildings in Canada during the French regime is known to have better properties, in particular better longevity, than even the modern mortars. However, the exact recipe for these mortars has been lost, as it was transferred verbally from the masons to their sons or successors. This paper covers part of a project attempting to find out the recipe of the ancient mortar from careful analysis of samples of ancient mortars provided graciously by the Québec Ministry of Cultural Affairs. These samples come from the remains of three stone buildings in the old part of Québec City: the second house of Champlain (1624), the Gervais-Beaudoin house (1682) and the "Magasin du Roy" (The King's Store)(1680-1700). For the sake of comparison, several mortar samples from buildings and houses in the Kingston area were also analyzed.

The analysis work presented here was based on instrumental neutron activation analysis (INAA), using the SLOWPOKE-2 nuclear research reactor at Royal Military College. The main elements detected and measured with the INAA technique were Ca, Al, Fe, K, Na, Ba, Mg, Ti, Cl and Mn. One of the difficulties with neutron activation analysis is that silicon has a (n,p) reaction with thermal neutrons, producing ^{28}Al, the same radioisotope resulting from the (n,γ) reaction with ^{27}Al. Therefore, Si had to be determined by means other than nuclear. The results from INAA were quite surprising, because the French regime samples had significant differences in their elemental compositions. The samples from the second house of Champlain and the Gervais-Beaudoin house are quite similar, with 8-10% calcium, 6% aluminum, and 2% iron, for the most abundant isotopes (excluding silicon). However, the "Magasin du Roy" samples exhibited some 24% calcium, 25% aluminum, and negligible concentrations of iron and other elements. For this last set of samples, the relative concentrations are quite similar to those of the St.Mary's Cathedral and the modern mortars in Kingston.

INAA produced results on the elemental content of the samples with a 5% accuracy, however, it cannot provide information on the molecular arrangement of the materials, and other, non-nuclear, methods are needed to obtain information here. The research is now performed using several conventional methods including infrared spectrometry, X-ray spectrometry, and other examination methods such as porosity, density and micro-hardness tests. Despite its limitations, INAA remains a key analysis method for this project since its high accuracy will be put to good use in the quantitative dosing of the various components of the French regime mortars.

HANDLED GAMMA QUANT GENERATOR IN DENSITY MEASUREMENT

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The up to date ways of the density measurement are based usually on the spectrometry of the scattering irradiation excited by a different radioisotopic sources.
In this paper, the principally new source - gamma quant generator is proposed.

The essential part of this arrangement is the source of artificial gamma radiation. The main advantage of this kind source in density measurement application is possibility of remote handling that excludes any polution in the case of damage.

The preliminary computation and mathematical simulation have revealed that the gamma quant generator has advantages upon the traditional radioisotopes sources in technical characteristics.

In the course of investigation the model of generator was created. The carrying out researches with this model have shown as well the expediency of subsequent investigations as the confirmation of the theoritical computations.

As a result of the researches the next parameters have been received:

1. Dimensions:

Diameter	70 mm
Length	300 mm
Weight	2 kg

2. Temperature up to 120 C
3. Registration precision of measured parameters is 10% at transposition speed 500 m/h
4. Electron source specification :

Kinetic energy	500 KeV
Current density electron beams	0.1 - 10 A/cm
Pulse duration	0.1 - 10 mcs
Beams dimension	0.1 - 20 cm
Impulse frequency	0.5 - 50 Hz

The attained high intensity of the irradiation and small pulse duration (5 10 ns) have required the making of a new measure method and special measure eqipment.

Also, with the same energy level as this of a radioisotopic sources, the high irradiation intensity per impulse allows to raise the reliability of measure results in compairison with a radioisotop sources.

With high speed detector in the couple, gamma quant generator can be used in geophisical investigations, defectoscopy of borehole casing columns, rock density measure, etc. This device can provide an effective work as well for boreholes in exploitation as for scientific researches.

Data acquisition and process-control system is PC-based. There are data acqusition and control cards housed within PC, with leads extending to remote sensors. The icon-based programming environment provide to acquire data, log it to disk, and graph it with great flexibility.

WEDNESDAY OCTOBER 6

10:30-12:00 Session C27: Human Factors 2
Peel Room
Chaired by: E. Davies (AECL Research, CRL)

C27.1 *Technology-Assisted Training in the Nuclear Regulatory Environment*
by D.J. Martin (Atomic Energy Control Board, Canada)

C27.2 *Alarm Processing for Diagnosis Using a Holographic Neural Network (HNeT)*
by J.E. Smith, M. Schwarzblat, and J.G. Sutherland (AND America Ltd.)

C27.3 *The Design and Implementation of an Operator's Performance Support System*
by R.J. Wilson (EACS - Engineering and Computing Services), A.A. Bokhari, W.J. Garland, W.F.S. Poehlman, and C.W. Baetsen (McMaster University)

C27.4 *Training Program Evaluation: Current Regulatory Activities*
by D. Tennant and R. Droll (Atomic Energy Control Board, Canada)

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TECHNOLOGY-ASSISTED TRAINING IN THE NUCLEAR REGULATORY ENVIRONMENT

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Introduction

One contributing factor to the safety of nuclear installations is adequately-trained personnel. While this is self-evident for the operational staff of a nuclear utility, it applies equally to staff of a nuclear regulatory agency. In late 1989, the Atomic Energy Control Board (AECB), Canada's nuclear regulatory agency, set up a Training Centre to oversee the production of training programs for AECB staff and for training staff of foreign regulatory agencies who had requested assistance from Canada.

In setting up the Training Centre, the natural intent was to prepare as high a quality of courses as possible. However, we realized that the *mechanics* of delivery of courses can materially affect their quality and effectiveness. This paper describes briefly how we address the issue of enhancing the effectiveness of courses by using appropriate technology.

Technology for three types of materials

The Training Centre's courses rely on computer technology to enhance the effectiveness of courses. The technology is applied to produce three types of materials:

1. **Printed materials**: produced using "desktop publishing". The pages are designed using aesthetically-pleasing layouts; typefaces, styles and sizes are chosen for maximum clarity; diagrams are produced to illustration-level quality.

2. **Video-based materials**: Training can be assisted, and content made more assimilable, by using videocassettes of the concepts under discussion, and so the Training Centre has set up a video editing station. Video sequences can be intercut with graphics and titles produced on a Macintosh. A VideoToaster allows professional-quality transitions and edits.

3. **Computer-based materials**: the Training Centre has adopted the use of comput[illegible]d interactive modules of two types: **Lecturer support modules** use a computer connected to a liquid-crystal display (L[illegible]) which sits atop a high-intensity overhead projector (of the type used for slide acetates). The audience sees on a standard projection screen what is on the computer monitor. The module itself contains text, colour photographs, animations and video segments. Thus, a rich environment of support material is available to the lecturer to assist in the presentation of the lecture content. **Self-paced modules** are interactive computer documents which reside on the computer on the trainee's desk. Digitized sound is used in the modules, and the sound track can easily be replaced with one of a different language. Digitized video sequences are also incorporated into the self-paced modules. They are particularly effective when used in conjunction with animations: the animation shows the concept involved, while the video footage shows what the real world application looks like. One crucial aspect of the use of technology to enhance effectiveness of courses is that the technology must not get in the way: it must be user-friendly, easy to set up and easy to use. The Training Centre's choices of hardware and software reflect this requirement.

Conclusion

The mechanics of presenting material can impede or enhance the flow and clarity of course information, and so can materially affect the quality and effectiveness of a training course. This paper has described briefly how the Training Centre of the Atomic Energy Control Board addresses the issue of enhancing the effectiveness of courses by using appropriate technology: desktop-publishing, video and computer-based interactive modules. Further information can be obtained by writing to the author.

ALARM PROCESSING FOR DIAGNOSIS USING A HOLOGRAPHIC NEURAL NETWORK (HNeT)

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SUMMARY

The purpose of using a neural network for alarm processing is to provide a process plant operator with a tool for rapid diagnosis of plant "upset" conditions. Consider that if a major process failure occurs at a nuclear generating station, many alarms will be triggered in the control room. In such cases, operators need help to determine which alarms are most significant, what the actual upset is and what needs to be done to correct it. Neural networks are a means of using a computer to emulate human thought and learning processes, and are patterned after the human brain as a structure of cells and interconnections. As such, they work very well in applications involving pattern recognition. Alarm processing is such an application.

Alarm processing using HNeT involves more than just the neural network for the overall process. Initially the alarm patterns which will occur for various plant upset conditions have to be generated. Then the alarm patterns and the upset conditions have to be modelled in a stimulus/response form that HNeT can handle. That is, the alarm pattern provides the stimulus, and the proper diagnosis is the response. Then the neural network has to be set up and trained. Finally, there has to be some post-processing to present the results in a form an operator can use to diagnose the upset.

Alarm patterns are generated using fault trees for the plant systems. Analysis of the fault trees provides the causes of upset conditions, and from these causes, the alarms which are triggered can be determined.

Modelling the alarms and the upset conditions as a stimulus/response pattern involves presenting the alarms as a set of binary inputs, indicating whether a particular alarm is on or off, and presenting each unique upset condition as a coded identifier. Alarm patterns become the stimuli. The upset condition identifier is the desired response.

The work reported in this paper was completed in January 1993. The system used was a hypothetical system, based on a typical nuclear power plant service water supply system. The HNeT network used was a single cell with very modest processing power. This has been sufficient to identify up to nine separate upset conditions from as many as sixty expected alarm patterns. A system to handle a complete plant will be more complex than this, but still tractable on a micro-computer. However, as proof of principle, a single cell works.

At the time of writing, (April 1993), work is proceeding with a real nuclear power plant system, and an improved version of the HNeT network.

In a practical situation, there will also need to be both pre-processing of inputs to HNeT and post-processing of the output from HNeT. Alarm signals from either an annunciator system or a control computer need to be delivered to the network in a form which it can recognize and process. Similarly, a coded identifier of an upset condition is of no use to an operator for diagnosis. The identifier needs to be translated into a descriptive form that leads the operator to a correct diagnosis of the upset and the appropriate corrective actions. These are areas for future work, when a practical neural network system has been demonstrated for a real plant, in software only.

THE DESIGN AND IMPLEMENTATION OF AN OPERATOR'S PERFORMANCE SUPPORT SYSTEM

by

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Summary

The objective of the current OPUS (Operator/User Support) System project is to produce an intelligent performance support system for the operators and maintainers of the Central Sampling and Condenser Leak Detection System at the Pt. Lepreau Nuclear Generating Station in New Brunswick. The design of the OPUS system addresses issues that have been raised in recent reviews and takes advantage of the experience that has been gained in the area of operator support systems over recent years. The fundamental design revolves around a user-centred concept since this will help to address one of the major shortcomings of past designs which is lack of operator acceptance. OPUS is also an active system in that it is designed to be responsive to changing circumstances. The model will eventually run on different platforms, for both performance and geographical reasons, and is multi-tasking in order to realize the inherent parallelism that is present. Each task is a separate software process and, to allow for incremental growth, proactive, persistent and intelligent modules are used. This is singularly useful in lessening the software maintenance requirements. An anthropomorphic three level (strategic, tactical and operational) model has been adopted that utilises the blackboard architecture. The consequential and concomitant aspects of communication, synchronisation, and organization of the computational flow for real-time performance create problems of their own, not least of which is the intertwined effects of temporal and logical correctness. The communications traffic is handled both asynchronously and synchronously and decomposed into four distinct categories. In order to develop the concepts on a real problem, the system for sampling the secondary side chemistry at the Pt. Lepreau Nuclear Generating Station was chosen. This phase is a single processor realisation but it is expected to rapidly expand into a multi-processor multi-platform implementation encompassing a number of operating systems.

TRAINING PROGRAM EVALUATION: CURRENT REGULATORY ACTIVITIES

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As part of the Operator Certification Division's (OCD's) "Initiatives" program, systematic evaluation of nuclear utility initial training programs has begun. Evaluation of continuing training programs will commence as soon as practicable. These efforts are aimed at assuring that nuclear utility training programs are effective and are adhering to the internationally-accepted principles of the Systematic Approach to Training (SAT). Each Canadian nuclear utility has already undergone two partial regulatory evaluations by members of the Training Program Evaluation Section (TPES) of OCD. TPES views SAT as consisting of three parts: (1) Prerequisites phase (analysis, design and development), (2) Training phase (implementation), and (3) Evaluation phase. Where nuclear utilities lack documentation containing acceptable criteria for the effective implementation of a systematic approach to training, the TPES is developing in a consultative fashion a set of standard criteria as the basis for utility training programs.

WEDNESDAY OCTOBER 6

10:30-12:00 Session C28: Safety 5
Wentworth Room
Chaired by: A. Carnino (IAEA)

C28.1 *A Turbine Trip Transient Analysis with TRAC-BF1*
by J.L. François (Instituto de Investigaciones Eléctricas, Cuernavaca, México)

C28.2 *Should We Install a Software-Based Reactor Protection System?*
by G. Ives (Colenco Power Consulting Ltd., Baden, Switzerland)

C28.3 *Operating Under Fire the French Way*
by F. Bediou and J.P. Chatry (EDF-CIG)

C28.4 *Nuclear Plant Safety Enhancement by Early Identification of Slow Developing Abnormal Processes*
by V. Kotelenets, S. Korolev and M. Konovich (Yuzhnoukrainskaya NPP, Ukraine)

A TURBINE TRIP TRANSIENT ANALYSES WITH TRAC-BF1

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SUMMARY

This paper shows the work performed to analyze the behavior of a BWR/5 nuclear power plant in the event of a turbine trip. This analysis was based on results obtained by the TRAC-BF1 code, a well-known best-estimate code used for transient analyses of boiling water reactors. The work undertook an evaluation of several safety parameters, such as dome pressure rise, maximum temperature in the fuel, maximum reactor power, and others.

A model of the BWR/5 plant developed with TRAC-BF1 used one equivalent main steam line from the vessel to the turbine control valves and one equivalent recirculation loop with one recirculation pump from the suction in vessel level 3 to the discharge at the jet pumps. The 20 jet pumps were lumped into an equivalent one. The vessel modeled has nine axial levels and two radial cells. The core was located at levels 4 and 5 and the control guide tubes at levels 2 and 3. The safety systems, low pressure and high pressure core sprays, were at the top of level 5, the steam separators and dryers in level 7 and the safety-relief valves at level 9, the vessel's top. The simulation took into account the conditions at 104.3% power, 100% core flow, the beginning of reactor life (BOL) and the bypass-on and bypass-off conditions were analyzed. The first step to achieve the simulation was to reach a steady-state condition comparable with the initial conditions pointed out before. Afterwards, the TRIP capability of the code was used to start the transient.

The sequence of events obtained with TRAC-BF1 starts with the closure of the turbine stop valves at 0 seconds, followed by the beginning of the SCRAM at 0.28 seconds, the recirculation pumps trip at 20 seconds for the bypass-on condition and at 0.15 seconds for the bypass-off condition. The opening of the first group of relief valves occurs at 0.48 seconds. For the case of bypass-on condition, the bypass valves open at 0.11 seconds. The results show that the dome pressure rise, due to the closure of the turbine stop valves, produced a steam void collapsing, forcing the water level descent in the downcomer region. The pressure rise reached its peak at 1.7 seconds after transient initiation. The 60% percent opening of relief valves were observed in the bypass-on condition and the 80% opening in the case of bypass-off, because their pressure setpoint was reached. Reactor power decreases asymptotically to 7% of rated value due to the control rod effect, which is greater than the positive effect due to void collapsing. The results of this work show that the values reached by the different parameters of the plant stayed low enough not to produce any effect on the safety of the plant. These results were compared with those of the Transient Safety Analysis Design Report provided by the reactor supplier and were found in quite good agreement.

SHOULD WE INSTALL A SOFTWARE-BASED REACTOR PROTECTION SYSTEM

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SUMMARY

Digital technology applied to protection systems can provide more rapid, comprehensive and accurate protection. It is not subject to age-induced factors such as drift, thereby eliminating the need for frequent maintenance, and it facilitates diagnostic self-testing.

The potential for reduced quantities of cables compared with traditional systems can also achieve a corresponding reduction in civil costs. However, although digital protection can help to eliminate some of the characteristic weaknesses of traditional systems, new technology introduces new uncertainties for all associated institutions.

Computer based protection systems have most of their complexity built into software which is subject to error. Because there are normally three or four lines of protection, any failures in a single-version software has the potential to cause all protection lines to fail simultaneously.

Functional or system diversity is therefore a prudent provision. Quantifying the reliability of software in large systems is not currently possible. Formal proof of dependability by functional testing of software is not feasible because the number of tests required becomes prohibitive.

Full static analysis of the software by diverse means is gaining acceptance, but is expensive on resources.

Subsequent modifications to protection systems may introduce problems of retesting, verification and validation.

Because licensing authorities generally feel more comfortable in licensing well-proven systems, innovative technology may be at a disadvantage. Public acceptance is influenced by a public increasingly conscious of the environment, and sceptical of nuclear power. Safety-critical software acceptability is influenced by international factors such as standards, politics and financing.

The dominating aspect of safety is how to produce and assure error-free software to ensure the potential for enhanced performance is realized. Despite the current uncertainties, experience will generate confidence and it is believed that digital protection systems will be part of the nuclear plant future.

OPERATING UNDER FIRE

THE FRENCH WAY

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ABSTRACT

Fire protection in nuclear plants usually has three aspects : prevention, detection and fire fighting. Prevention mainly consists in avoiding that a single fire may render two redundant safety systems unavailable. Detection aims at rapidly locating a starting fire, giving the alarm, and sometimes initiating automatic actions.

Fire fighting is organized to extinguish any fire fast. On French plants, an operator is sent to confirm the fire and to extinguish it. In case of difficulty, he isolates the area, informs the control room who will call for outside support.

In addition to this organization which aims at both the fire and its direct consequences, Electricité de France has developed an approach that allows the safe operation of the unit. This approach makes the assumption that the fire remains confined in a fire compartment, but all the electric equipment within this compartment is liable to be damaged and is subject to spurious faults. In order to mitigate these faults, all actuators are de-energised, and these de-energisations are programmed in such a way that untimely actions are avoided. The analysis determines which operational functions are unavailable due to this de-energisation. The list of unavailabilities allows the selection of the correct operating procedure. Each procedure is structured according to the function to be ensured (core cooling, water level or anti-reactivity margin) and for each function, according to the systems necessary or their possible substitutes .

This approach implies that one can prove that whatever compartment is on fire, at least one substitute system for each safety function is available. The electrical distribution gives great importance to the A-train as only the redundant protection and safeguard systems are supplied by the B-train. For the control room operator to have sufficient means to bring the plant to a safe shut-down state in case of a total loss of the A-train, it is preferable to ensure the availability of some minimal operating systems in addition to the B-train ones. In pratice, this imposes the protection of a few control and power cables from non-redundant systems necessary for operator information or for long term operation.

NUCLEAR PLANT SAFETY ENHANCEMENT BY EARLY IDENTIFICATION OF SLOW-DEVELOPING ABNORMAL PROCESSES.

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The primary focus for the owner of a nuclear power station is to operate the plant in a safe manner. The early identification of slow developing abnormal processes plays a vital role in the plant performance increasing.

Slow proceeding abnormal processes which may take place during NPP operation are as a rule connected with an ageing and a wear of equipment. Such processes may result in the serious malfunctions and incidents. Therefore it is clearly desirable to have a method by which the abnormal process may be identified.

Anomalies are always accompanied by the thermohydraulic parameters trends. Even its small deviation from the base values may serve as an indicator of unfavorable tendency beginnings. Depending on the nature of proceeding process there is a wide variety of possible combinations of the parameters behavior and its quantitative changes.

The reactor operator is not able to identify the beginning of the abnormal process because in most cases it is necessary to evaluate a great amount of statistic data, with the values of each separate parameter, as a rule, being within normal range.

The condition of equipment is characterized by the set of N readings of M sensors. Therefore it may be presented as a point in M-dimensional space. This "space of conditions" includes all possible conditions of equipment, for example "normal", "pre-alarm", and "alarm".

The beginning of slow-proceeding abnormal process may be identified using parameters diagnostics algorithms. Actually the problem of identification of equipment condition on the basis of thermohydraulic parameters trends is the problem of classification. The main items of this problem are as follows:
- division of "equipment condition space" into regions (or subspaces) for example "normal", "pre-alarm", and "alarm";
- identification of a region which the observed equipment condition belongs;
- prognostication of the equipment condition developing trend and the determination of the time limit before a transition into anohter region (subspace).

The success of a classification depends on the algorithm which was chosen for the processing of the measured parameters. In our view one of the most suitable algorithm is the "method of eigenvectors". This method was used to develop a computer code called "SODI". The code has been successfully probed at the Yuhnoukrainskaya NPP. The obtained results confirm the possibility of identification of slow developing abnormal processes at the basis of parameters diagnostics methods.

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