IAEA-TECDOC-1650

Good Practices in Heavy Water Reactor Operation

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International Atomic Energy Agency

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FOREWORD

The value and importance of organizations in the nuclear industry engaged in the collection and analysis of operating experience and best practices has been clearly identified in various IAEA publications and exercises. Both facility safety and operational efficiency can benefit from such information sharing. Such sharing also benefits organizations engaged in the development of new nuclear power plants, as it provides information to assist in optimizing designs to deliver improved safety and power generation performance. In cooperation with Atomic Energy of Canada, Ltd, the IAEA organized the workshop on best practices in Heavy Water Reactor Operation in Toronto, Canada from 16 to 19 September 2008, to assist interested Member States in sharing best practices and to provide a forum for the exchange of information among participating nuclear professionals. This workshop was organized under Technical Cooperation Project INT/4/141, on Status and Prospects of Development for and Applications of Innovative Reactor Concepts for Developing Countries.

The workshop participants were experts actively engaged in various aspects of heavy water reactor operation. Participants presented information on activities and practices deemed by them to be best practices in a particular area for consideration by the workshop participants. Presentations by the participants covered a broad range of operational practices, including regulatory aspects, the reduction of occupational dose, performance improvements, and reducing operating and maintenance costs.

This publication summarizes the material presented at the workshop, and includes session summaries prepared by the chair of each session and papers submitted by the presenters. The IAEA thanks all the experts for their contributions to the workshop through presentation of their work, detailed discussions on various aspects of the topic, and the manuscript to this publication. The special contributions of P. Allsop of AECL, who worked extensively with the IAEA to edit this document, is gratefully acknowledged. The IAEA is also thankful to B. Shalaby and R. Zemdegs for local organization of the workshop, and to L. Biro, A. Dell, D. Mullin and R. Urjan for chairing technical sessions. The IAEA officer responsible for this publication was J.-H. Choi of the Division of Nuclear Power.

EDITORIAL NOTE

CONTENTS

SUMMARY

Heavy water reactors (HWRs) are the second most common type of nuclear power reactor in the world, and have accumulated more than 600 reactor years of operating experience. Member States operating and developing HWRs are interested in exchanging operational information to improve the safety and efficiency of their own reactors. Such sharing also benefits organizations engaged in the development of new nuclear power plants as it provides information to assist in optimizing designs to deliver improved safety and power generation performance.

On the recommendation of the IAEA Nuclear Energy Department's Technical Working Group on Advanced Technologies for Heavy Water Reactors, the IAEA, in cooperation with Atomic Energy of Canada, Ltd, organized a workshop on best practices in Heavy Water Reactor Operation in Toronto, Canada from 16 to 19 September 2008. The objective of the workshop was to share best practices on the operational experience of Heavy Water Reactors, including the following areas:

- Regulatory aspects;
- Reduction in occupational dose exposure;
- Improvements in performance;
- Reduction in operational and maintenance costs.

The workshop participants were experts actively engaged in various aspects of HWR operation, regulation or design. Participants presented information on operational activities and practices deemed by themselves and/or their organizations to be best practices in a particular area, and workshop participants discussed their applicability to other plants based on differences in plant design, regulatory requirements or operating philosophy. This publication summarizes the material presented at the workshop, and includes session summaries prepared by the chair of each session and papers submitted by the presenters.

While the material presented and the detailed lessons spanned a wide range of activities and practices, several common themes emerged during the workshop. These consisted of more generic best practices that were the foundation of several different experiences. These overall observations/best practices were as follows.

Best practice: Challenge the status quo

Improved technology and understanding have created opportunities for improvement that did not exist when some plants were built and may not have existed even a few years ago. While the use of proven methods remains of great value and importance, this should not preclude considering or adopting newer technologies and methodologies where they have benefits.

Best Practice: Share information and benchmark performance

Good ideas originate everywhere, in organizations and groups at all performance levels within the industry. Organizations and groups within organizations should actively pursue opportunities to share experiences and benchmark performance.

Best Practice: Engage the workers and use multi-disciplinary teams

Operational excellence requires an integrated approach. Workers directly involved in operating, maintaining, modifying and designing processes and equipment have a wealth of knowledge that should be tapped to improve performance.

Best Practice: There is value in analysing performance

We must identify and understand the challenge before we can solve it. Both on-going and event-driven analyses of performance can provide valuable information needed to improve performance.

Best Practice: Set targets and assign monetary values

Developing the business case for change is part of the improvement process, and can be the most challenging part. Having agreed performance targets and associating those with monetary values simplifies the approval process and helps individuals, groups and organizations prioritise activities.

Best Practice: Training should never stop

Human performance is critical to good operational performance. Training programmes should go beyond initial-task and remedial training, to encompass effective refresher training.

As a final measure of the workshop's value and effectiveness, it was recommended that the IAEA consider repeating this workshop annually. The workshop organizers and session chairs indicated that the participation of working-level staff (e.g. first and second-level managers) actively involved in operating power reactors was of particular value, and suggested that consideration be given to hosting future workshops at or very close to operating power stations.

Summary of Session I: Regulatory aspects

The session was devoted to the specific regulatory assessment aspects of HWR safety performance and the development of risk-informed regulatory positions on HWR safety issues in Canada, and the evolution of regulatory requirements for HWRs in Romania. Experts from the Canadian Nuclear Safety Commission (CNSC) and from the National Commission for Nuclear Activities Control of Romania (CNCAN) presented the material.

P. Corcoran from CNSC presented the first paper. The paper focussed on the regulatory process used in Canada to assess the safety performance of nuclear power plants (NPPs). The main objective of such assessments is to ensure that the NPP utilities have programmes in place to meet their licensing requirements. It was emphasised that assessments by the CNSC staff are based on legal requirements in the Nuclear Safety and Control Act and its associated regulations, on conditions in operating licences, and on applicable standards. The assessments are further supported through information gathered through staff inspections, on-site staff presence, corrective-action follow-ups, document reviews, event reviews and performance indicators.

The best practice is that the results of the CNSC staff assessments are documented annually in a report entitled The Annual CNSC Staff Report on the Safety Performance of the Canadian Nuclear Power Industry (the NPP Report). The NPP Report is a public pronouncement on the safety performance of power-reactor licensees in Canada, and serves to demonstrate to stakeholders how the CNSC fulfils its mandate to ensure that NPP operation poses no unreasonable risk to the health, safety and security of Canadians and their environment, and respects international obligations on the peaceful use of nuclear energy. Public information sessions to promote the report in NPP-host communities were first introduced in 2007. The objective of these sessions was to engage the public and inform the communities about the findings and results of the 2006 NPP Report. Overall feedback from session participants was positive.

The NPP Report contains station-by-station assessments of safety performance in eight safety areas, S-99 performance indicator data,[1] and highlights industry trends across the safety areas. Licensee programmes are grouped into nine safety areas encompassing eighteen programmes. Collectively, these represent the licensee programmes necessary for the safe operation of an NPP and are consistent with the criteria found in IAEA Safety Standards Series No. NS-R-2, Safety of Nuclear Power Plant Operation. Only eight of these safety areas are included in the NPP Report.

The CNSC rating system uses a five-level grading scheme:

A – exceeds requirements,
B – meets requirements,
C – below requirements,
D – significantly below requirements, and
E – unacceptable.

The annual report card is a summary of the programme and implementation grades for all the NPP stations. It is provided at the end of the NPP Report. The paper also presents the report card showing programme and implementation grades for all safety areas and programmes in 2007.

The annual process of assessing and rating each programme within the safety areas also helps CNSC staff focus on those areas that will require increased regulatory oversight.

The paper presents a number of improvements that are currently being considered, such as:

- risk ranking the safety areas and applying risk-informed decision making to the rating process for a more transparent and methodical means of grading the facilities;

- decreasing the number of ratings by evolving the grades for ‘programme’ and ‘implementation’ into a single grade for ‘safety performance’;

- decreasing the size of the report by using a standard template and eliminating repetition; and

- using plain language to facilitate understanding by the public.

Lucian Biro from the CNCAN presented the second paper of the session. The paper focussed on the main features of the licensing process developed by the CNCAN for Cernavoda NPP, Units 1 & 2, and on the experience gained by the regulatory body that will be applied for the next units (3 & 4).

In evolving the Romanian regulatory environment, the CNCAN has taken into consideration several important aspects that arise in the context of the post-accession process in the European Union, after 1st January 2007. The first category of regulatory requirements originated from the necessity to comply with the requirements imposed by the high-level safety standards planned for 2014-2015 in Europe. The second category of regulatory requirements was developed by the CNCAN, as a member of Western European Nuclear Regulators' Association (WENRA), in recognition of the reactor-regulations harmonisation process envisaged to be completed by 2010. The third group of regulatory requirements was based on the experiences and practices discussed during meetings of the CANDU® Seniors Regulators Group since 1997. Also, the paper presented the CNCAN approach of using the Convention on Nuclear Safety, IAEA IRRS Missions and the IAEA Technical Cooperation projects as driving forces for assisting in the development of high standards, regulations and regulatory practices for nuclear safety, the commissioning process and operation of the Cernavoda NPP.

The evolution of HWR regulatory requirements in Romania shows a positive trend. The CNCAN, as safety authority of Romania, established an appropriate strategy for the integration of Canadian

[1] Regulatory Standard S-99, Reporting Requirements for Operating Nuclear Power Plants. CNSC, 2003.
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regulatory experience and practices into the Romanian licensing process for the Cernavoda NPP. The regulatory approach for units 3 & 4 of the Cernavoda NPP will continue in the same manner as for the previous units 1 & 2. Future amendments to the licensing process will take into consideration compliance with the safety standards and guides, which are anticipated for 2014-2015 at the international level.

G. Ishack from the CNSC presented the third paper of this session. The paper was devoted to the development of risk-informed regulatory positions on CANDU safety issues and was divided in two parts. The subject of the first part was the methodology development for risk estimation and evaluation, and the authors were A. Bujor, G. Rzentkowski and D. Miller from the CNSC. The subject of the second part was the application of risk-informed decision making for categorizing safety issues and the authors were D. Miller, G. Rzentkowski and A. Bujor. The paper presented the risk-informed decision-making process used by the CNSC as a management tool to support decisions related to licensing, compliance and planning/resource allocation. By using this decision-making process the CNSC obtains information on the risks related to a certain decision, produces recommendations for controlling the risk, and provides for implementing risk-control measures and monitoring the impact on risk following a decision. The paper described the methodology developed for estimating and evaluating risks for input into the risk-informed decision making process. The methodology described allows all risks to be considered in a consistent manner, and recognizes the differences between the risks in the absence and presence of the safety issue. The methodology refers to safety related risks, although similar considerations could be made for other types of risks. Also, the paper described the approach taken by CNSC staff to identify the list of outstanding safety issues for Canadian CANDU reactors, and the application of Risk-Informed Decision Making for developing risk-informed regulatory positions, including the path forward for resolution of each safety issue in view of currently operating reactors, life extension of existing reactors and new reactors. A general categorization of the issues was outlined, and two specific examples were presented for the risk identification, estimation, and evaluation with the use of the methodology. Measures to control the risk were also recommended.

Summary of Session II: Reduction in occupational dose exposure

The materials presented in this session dealt with specific experience on occupational dose reduction. The best practice overall identified during this session was that developing the business case for change is part of the improvement process, operational excellence requires an integrated approach between workers directly involved in operating, maintaining, modifying and designing processes, and training should never stop.

Pickering B, Ontario Power Generation of Canada (OPG), presented a paper documenting significant dose savings during a single fuel channel replacement. The dose savings were achieved through the use of a new reactor-face shielding tool. A multi-disciplined team was involved in the development of this reactor-face tool. The tool cut reactor-face radiation fields in half and resulted in the work being completed for 15% less than target.

GE Hitachi has developed a new detritiation technology at its light isotope centre of excellence in Peterborough, Ontario, Canada. This technology uses a diffusion-based isotope separation process and has the potential to achieve very low levels of tritium in primary heat transport (PHT) and moderator systems. Low levels of tritium in PHT systems can lead to significantly reduced airborne tritium during outages, reduced requirement for air supplied plastic suits, and ultimately lower doses.

The environment section at Pickering B, OPG, Canada presented a paper on their response to an incident where tritium emissions increased and were approaching unacceptable (to station management) levels. A multi-disciplined team was formed to develop an action plan. The plan included cataloguing all leaks, improving dryer performance, reducing PHT and moderator tritium concentrations, and implementing real time tritium monitors for surveillance. The plan was implemented in 2004 and resulted in significant reductions in both tritium emissions and internal dose.

The Nuclear Power Corporation of India Limited (NPCIL) took major steps and used a holistic approach to reduce collective doses in their nuclear power plants. Corporate policy was revised to strengthen the radiation protection programme. All aspects of dose reduction were addressed including procedure changes, design changes, training, performance expectations and management oversight. The efforts resulted in significant reductions in collective dose over the period 2002 to 2007.

The Korea Hydro and Nuclear Power Company undertook an initiative at Wolsung 1 in 2005 to reduce PHT filter pore size in stages from 6 µm to 0.45 µm. Between 2005 and 2007, a reduction in radionuclides in crud samples was observed. Co 58, Co 60, and Zr/Nb 95 had all decreased. A 14% decrease in steam generator dose during this period was also observed. In January 2008, a 0.1 µm filter was installed and performance will continue to be monitored. KHNP also have plans to reduce pore size in their fuelling machine system filters at Wolsung 1.

GE Hitachi presented a paper on the use of rotary type portable vapour recovery dryers. The presentation included a description of how the system works and potential uses such as drying calandria vaults, drying PHT filters prior to disposal and drying spills. Portable dryers are in use at a number of CANDU stations, where they are primarily used to augment existing vapour recovery dryers and have proven to be very useful in lowering airborne tritium concentrations.

The Darlington nuclear power station from Canada delivered a presentation on improvements in its ALARA programme that were key contributors in receiving the 2007 ISOE World Class ALARA Performance Award. The topics covered were leadership, creativity and innovation, internal dose reduction, and human performance improvement.

- A bold vision of the future was established with action plans to achieve less than one (1) MPC[1] in containment by 2011, and a 25% reduction in gamma dose rate by 2011.

- Innovative monitoring and shielding techniques were discussed including remote reactor face scans, use of water walls and water bags for shielding, and flexible reactor face shielding blocks. Analysis and characterization of source term, dose and dose rates has led to the development of improved shielding strategies.

Human Performance was also identified as an area that can contribute to dose reduction through effective communication strategies, recognition and performance-management programmes.

Summary of Session III: Performance improvement

The materials presented at this session dealt with various performance-improvement initiatives. The best overall practice identified during this session was that there is value in analysing performance and striving to gain efficiency and reliability in plant processes and performance as part of an overall effort in continuous improvement.

The first on the agenda, S. Milley from OPG presented the use of a human-performance simulator at Pickering NGS. The key message from this presentation was that human performance is a major contributor to events involving reduction in safety and loss of productivity. The human-performance simulator was developed to train already qualified staff in the application of the Event Prevention Framework (observations and coaching), to identify and practice real-life situations in an error-tolerant environment, and to practice as a supervised work team. The best practice identified from this presentation was that training never stops and the use of ‘real-life’ training simulations improves worker performance and understanding of work conditions.

D. Mullin from New Brunswick Power delivered the second presentation of the session. This highlighted the unique aspects of the Level 2 Probabilistic Safety Assessment (PSA) developed for Point Lepreau Generating station, and how it used integrated models and a high degree of operational feedback in the model data including site-specific failure data, access and repair times, and

1 Maximum permissible concentration in air.

surveillance intervals. The PSA has led to many improvements in design, testing and maintenance to cater to both design basis and severe accidents. The best practice identified from this presentation was that modern information tools applied to an existing plant can bring benefits in both safety and performance, and that such tools must reflect the realities of how a plant is designed, operated and maintained.

G.A. Urrutia from CNEA[1] delivered the third presentation of this session, which described improvements in power measurement and plant efficiency that has been achieved at Atucha 1 Nuclear Power Plant. An approach was detailed on how feedwater flow and process temperature measurement precision and accuracy can be improved to realize gains in overall plant efficiency. The best practice identified from this presentation was that it is important to understand what you are actually measuring versus what you believe you are measuring; data reconciliation is an important tool.

W.G. Park from KHNP[2] delivered the fourth presentation of the session. The presentation described the evolution of the fuelling machine D_2O pressure control system for Wolsong Nuclear Power Plants. The presentation highlighted fuelling machine magazine and C-ram control and pressure problems that were experienced at the Wolsong units, and the solutions that were implemented to prevent loss of safety function, to prevent D_2O spills, forced shutdowns for fuelling machine maintenance, and human error. The solution involved a redesign, moving away from analog controllers to digital modular programming controllers. The best practice identified from this presentation was that when modernizing Instrumentation and Control equipment or components, consider opportunities for improvements beyond like-for-like replacement.

S.G. Hada from CNE-PROD delivered the fifth paper of this session. The paper addressed the implementation of the PLATON (PLAnt Tailored InformatiON system) and SPV (Single Point Vulnerability) programmes at the Cernavoda Nuclear Power Plant.

The implementation of PLATON, which allows for real-time monitoring of plant systems, resulted in an improvement in the quality of technical work, manpower savings, and increased knowledge of the plant and its systems. The system provided a crucial tool for transient analysis, and a unique facility for fast capturing and reporting of real system behavior during an event. It also facilitated optimization and cost reductions between the two Cernavoda units. The benefits were large enough that the plant realized a full return on investment before the system was completely deployed.

The main goal of the SPV initiative was described as being to reduce as much as possible the vulnerability of the plant to events caused by failures of equipment that represent a single point of vulnerability (SPV). Screening for SPV items at Unit 1 and, in part, Unit 2, at Cernavoda resulted in 1420 pieces of equipment that required further examination of their failure mechanisms, identification of the necessary preventative maintenance tasks to prevent those failure mechanisms from occurring, and optimization of the maintenance programme. The next steps described included actions such as examining critical spares inventory to ensure that the maintenance programme is adequately supported.

The best practices identified through the fifth paper were (1) modern information tools applied to an existing plant can bring benefits in both safety and performance, and (2) examining and understanding Single Pont Vulnerabilities that can cause problems with safety, power production or economics can yield large financial benefits to utilities through a reduction of forced loss rates.

M. Reid of the CANDU Owners Group (COG) presented the sixth and final paper of the session. The paper provided an overall review of CANDU plant performance. The presentation examined three-year average load factors, causal factors and various COG initiatives that are underway to assist utilities with improving plant performance. The review of CANDU performance confirms that CANDU reactors can deliver excellent performance, but utilities need to focus on Forced Loss Factors

[1] Comisión Nacional De Energía Atómica, República Argentina.
[2] Korea Hydro and Nuclear Power, Limited.

(FLR) and Unplanned Capacity Loss Factors (UCLF). The best practice identified from this presentation was the importance of sharing information, benchmarking and being more proactive in improving key areas (e.g. preventative maintenance, human factors, etc.).

Summary of Session IV: Reduction in operation and maintenance cost

The materials presented in this session dealt with specific improvements in station performance. The best practice overall identified during this session was that there is always potential for reductions in costs, for optimization of outage duration, and for integrating large-scale projects by leveraging the innovation of staff, networking between utilities and cooperation between stations and their regulator.

M. Brown from OPG made the first presentation, which described the cleanup and recovery of high TOC (total organic carbon) D_2O at Pickering NGS. The key message from this presentation was that the most powerful resource in the nuclear business (as well as in any other business) is the people. Pickering managed to focus the creativity of Common Services staff to solve a long-standing problem. A large quantity of downgraded D_2O had accumulated over years and was stored in drums. The quantity of stored D_2O was sufficiently large (250 Mg of TOC contaminated D_2O and over 2100 drums) that housekeeping and normal access into some areas inside the station were affected.

A team comprised of engineers, operators and maintenance staff designed and built a process using plant equipment and commercially available parts. The initial cleanup of a small amount of TOC-contaminated D_2O was attempted and found to be successful. A larger-scale process was then built. The system has been added to and modified for maximum efficiency, and has been in use for approximately two years. The Pickering units are now free of drums and housekeeping has improved immensely. The best practice identified from this presentation was to challenge the status quo, use multi-disciplinary teams and let the staff innovate.

P. Lafreniere from the CANDU Owners Group (COG) gave the second presentation of the session. It provided an overview of the Fuel-Handling Benchmarking project, which was coordinated by COG and it was an excellent example of international cooperation between the CANDU utilities.

Since May 2007, when this project was launched, a benchmarking team comprised of fuel-handling experts performed twelve station visits around the world with the goal of identifying the strengths and areas for improvements related to fuel-handling activities, such that all utilities could take advantage from this information database. The motivation behind this initiative was the observation that “the station name did not really matter…fuel handling was often a bad news file”, and that the advantages of CANDU on-line fuelling were not being realized due to poor equipment reliability. In April 2008, the benchmarking visits were complete and by September 2008, the final report of the project was issued.

In the benchmarking report, fifteen best practices and eighteen common issues were identified and documented. An active fuel handling peer network was developed and cooperation continues with the exchange of procedures, post-project exchanges visits, emergency spare parts requests, etc. The best practice identified by this project was the power of sharing experience: fuel-handling issues exist at every station, but the problems are different because someone has solved every problem somewhere.

W.G. Park from KHNP made the third presentation of the session. This described the development of the consolidated spent fuel dry storage system at Wolsong NPP. At Wolsong site there are four CANDU 6 units that discharge about 20 000 bundles of spent fuel annually into the spent fuel bay. After a cooling period these bundles are transferred to the dry storage facility, which uses concrete canisters for fuel storage. This dry storage facility will reach its capacity limit by the end of 2009.

To prepare for the extension of the dry storage facility, KHNP evaluated storage options in depth to select the best system on the basis of nuclear and radiation safety, technological maturity, economics and space requirements. KHNP has selected AECL as a partner for the joint development of the MACSTOR/KN-400 storage module. The MACSTOR/KN-400 is an enlarged version of the

MACSTOR-200 that reuses its proven features while doubling its capacity. The storage density of the new modules will be approximately 88 bundles per square metre, which is three times the density of the existing concrete canisters.

It was concluded that the MACSTOR/KN-400 module satisfies requirements for the safe storage of CANDU spent fuels: it can store CANDU 6 fuel baskets containing reference fuel bundles, and passively dissipate heat generated by the stored fuel to maintain fuel bundles and the storage module at acceptable temperatures. It can provide sufficient shielding to attenuate gamma and neutron radiation below acceptable values; it provides confinement for the storage basket; it provides adequate structural integrity during construction, normal and abnormal operation and during Design Basis Events; it provides capability for periodic sampling of each storage cylinder cavity; it provides a basic intrusion resistance against removal of fissile material and receptacles for installation of Safeguards monitoring equipment by the IAEA. The best practice identified by this presentation was that cooperation and new technologies offer improved solutions to existing challenges.

R. Urjan from OPG presented the fourth and final paper of the session. This paper dealt with the rod-based guaranteed shutdown state (RBGSS) implementation at Pickering B. RBGSS offers significant advantages over the traditional method of achieving the guaranteed shutdown state (GSS).

Historically, the GSS on CANDU stations was typically achieved by injecting poison (Gadolinium Nitrate) into the main moderator circuit with a physical separation of the moderator-purification system from the main circuit (for example, removal of a spool piece) to prevent accidental or intentional poison removal. This method was considered labour intensive, time consuming and entailed radioactive dose to the personnel.

In comparison to liquid-poison GSS, RBGSS was viewed as easier to apply, with low dose to personnel and savings of up to 48 h on outages duration. Another advantage was that the reactor remains under regulating system control at all times. The RBGSS method employed by Pickering B achieves sub-criticality by the insertion of the reactivity mechanisms in core; i.e. by inserting all the shutoff rods, adjuster absorbers and control absorbers. Criticality can be restored by simply withdrawing the shutoff rods and control absorbers from core.

Pickering B started the technical analysis of RBGSS and discussions with the regulator (the CNSC) several years ago. The fact that RBGSS was approved (for one time use at the time of the presentation) was identified as a very good example of cooperation between the Utility, the Regulator and the scientific world. A trial of the concept was performed in 2007 during the Unit 6 planned outage, when poison and RBGSS were used in parallel. The CNSC gave one-time approval to use RBGSS during the Unit 7 outage in October 2008. The next step is to obtain approval for permanent use. The best practice from this project was that an alternate method to ensure reactor guaranteed shutdown state has been developed, and that it offers significant improvements in operational performance.

SESSION I. REGULATORY ASPECTS

REGULATORY ASSESSMENT OF NUCLEAR POWER PLANT SAFETY PERFORMANCE IN CANADA

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Abstract

The NPP Report (The Annual CNSC Staff Report on the Safety Performance of the Canadian Nuclear Power Industry) is a public pronouncement on the safety performance of power reactor licensees in Canada and serves to demonstrate to stakeholders how the CNSC fulfils its mandate of ensuring that NPP operation poses no unreasonable risk to the health, safety and security of Canadians and their environment and respects international obligations on the peaceful use of nuclear energy. The goal of the CNSC is to produce a report that is transparent, clear, concise and timely. In the spirit of continuous improvement, the CNSC is constantly looking for ways to improve the NPP Report as a communication product.

1. Introduction

Regulatory assessment of the safety performance of power reactor licensees in Canada is conducted to ensure that the nuclear power plant (NPP) utilities have programmes in place to meet their licensing requirements. Canadian Nuclear Safety Commission (CNSC) staff base their assessments on the legal requirements in the Nuclear Safety and Control Act, on the regulations, on conditions of operating licences and on applicable standards. The assessments are further supported through information gathered through staff inspections, on-site staff presence, corrective action follow-up, document reviews, event reviews and performance indicators.

Results of the CNSC staff assessments are documented annually in a report entitled The Annual CNSC Staff Report on the Safety Performance of the Canadian Nuclear Power Industry (the NPP Report). The NPP Report is a public pronouncement on the safety performance of power reactor licensees in Canada and serves to demonstrate to stakeholders how the CNSC fulfils its mandate of ensuring that NPP operation poses no unreasonable risk to the health, safety and security of Canadians and their environment and respects international obligations on the peaceful use of nuclear energy.

Public information sessions to promote the report in NPP-host communities were first introduced in 2007. The objective of these sessions was to engage the public and inform the communities about the findings and results of the 2006 NPP Report. Overall feedback from session participants was positive. Public information sessions for the 2007 NPP Report are planned for Fall 2008. NPP Reports from 2001 to 2007 are available on the CNSC web site at www.nuclearsafety.gc.ca

In addition to its role as an important reporting tool, the NPP Report also serves to inform and guide the CNSC compliance programme for power reactor licensees. The report provides individual station assessments, tracks performance indicators over time, and summarizes industry trends, which in turn enables CNSC staff to focus their attention and to follow-up on areas of particular concern.

2. Report content

The NPP Report is comprised of two sections and a number of appendices. Section 1 of the report contains station-by-station assessments of safety performance in 8 safety areas, while Section 2 presents S-99[1] performance indicator data and highlights industry trends across the safety areas. Typically, the report includes approximately six appendices containing supplementary information such as definitions and performance objectives, a glossary, an explanation of the rating system, etc. Also included is an appendix summarizing significant developments at each station for the calendar year as well as an appendix describing outstanding Generic Action Items and CANDU safety issues.

[1] Regulatory Standard S-99 Reporting Requirements for Operating Nuclear Power Plants. CNSC, 2003.

Licensee programmes are grouped into nine safety areas encompassing eighteen programmes (see Table 1). Collectively these represent the licensee programmes necessary for the safe operation of NPP and are consistent with the criteria found in IAEA Safety Standards No. NS-R-2, Safety of Nuclear Power Plant Operation. Only eight of these safety areas are included in the NPP Report. station assessments for the Site Security safety area are treated as classified information and are addressed through a separate report.

Table 1. Safety areas and programmes

Safety Area	Programme
Operating Performance	Organization and Plant Management
	Operations
	Occupational Health and Safety (non-radiological)
Performance Assurance	Quality Management
	Human Factors
	Training, Examination and Certification
Design and Analysis	Safety Analysis
	Safety Issues
	Design
Equipment Fitness for Service	Maintenance
	Structural Integrity
	Reliability
	Equipment Qualification
Emergency Preparedness	Emergency Preparedness
Environmental Protection	Environmental Protection
Radiation Protection	Radiation Protection
Site Security	Site Security
Safeguards	Safeguards

3. Rating process and rating system

CNSC staff assessments are based on applicable legal requirements, staff inspections, on-site staff presence, corrective action follow-up, document reviews, event reviews and performance indicators. The assessments culminate in a ‘programme’ (i.e. programme design) rating and an ‘implementation’ rating. Programme and implementation ratings are assigned for each safety area and for each programme within the safety areas.

CNSC staff use the same rating system when making recommendations to the Commission for operating licence renewals. The rating system uses a five-level grading scheme as follows:

A – Exceeds requirements: Assessment topics or programmes meet and consistently exceed applicable CNSC requirements and performance expectations. Performance is stable or improving. Any problems or issues that arise are promptly addressed, such that they do not pose an unreasonable risk to the maintenance of health, safety, security, environmental protection, or conformance with international obligations to which Canada has agreed.

B – Meets requirements: Assessment topics or programmes meet the intent or objectives of CNSC requirements and performance expectations. There is only minor deviation from requirements or the expectations for the design and/or execution of the programmes, but these deviations do not represent an unreasonable risk to the maintenance of health, safety, security, environmental protection, or conformance with international obligations to which Canada has agreed. That is, there is some slippage with respect to the requirements and expectations for programme design and execution. However those issues are considered to pose a low risk to the achievement of regulatory performance requirements and expectations of the CNSC.

C – Below requirements: Performance deteriorates and falls below expectations, or assessment topics or programmes deviate from the intent or objectives of CNSC requirements, to the extent that there is a moderate risk that the programmes will ultimately fail to achieve expectations for the maintenance of health, safety, security, environmental protection, or conformance with international obligations to which Canada has agreed. Although the risk of failing to meet regulatory requirements in the short term remains low, improvements in performance or programmes are required to address identified weaknesses. The licensee or applicant has taken, or is taking appropriate action.

D – Significantly below requirements: Assessment topics or programmes are significantly below requirements, or there is evidence of continued poor performance, to the extent that whole programmes are undermined. This area is compromised. Without corrective action, there is a high probability that the deficiencies will lead to an unreasonable risk to the maintenance of health, safety, security, environmental protection, or conformance with international obligations to which Canada has agreed. Issues are not being addressed effectively by the licensee or applicant. The licensee or applicant has neither taken appropriate compensating measures nor provided an alternative plan of action.

E – Unacceptable: Evidence of an absence, total inadequacy, breakdown, or loss of control of an assessment topic or a programme. There is a very high probability of an unreasonable risk to the maintenance of health, safety, security, environmental protection, or conformance with international obligations to which Canada has agreed. An appropriate regulatory response, such as an order or restrictive licensing action has been or is being implemented to rectify the situation.

Since the introduction of this rating scheme in 2002, no station has received a grade below ‘C’.

4. Report card

The annual report card is a summary of the programme and implementation grades for all the NPP stations. It is provided at the end of the NPP Report in addition to being published separately on the CNSC web site. The 2007 Report Card is provided in Table 2.

As indicated in Table 2, the CNSC ranked a number of safety areas and programmes at the ‘A’ level in 2007, including implementation of Occupational Health and Safety (Station 1, Station 2), Reliability programme design (Station 5), Emergency Preparedness (all stations for programme and Station 1, Station 2 and Station 3 for implementation) and Radiation Protection implementation (Station 2).

The majority of the programmes and their implementation were rated as ‘B’ in 2007. This is typical of most years. Some programmes, however, such as implementation of the Safety Analysis Programme at Station 3, were noted by CNSC staff to be ‘deteriorating’. As such, these areas will likely receive increased scrutiny, despite being currently rated as ‘B’.

Across the industry, thirteen safety and programme areas received ‘C’ grades in 2007, all of them for implementation of the particular safety area or programme in question. Detailed explanations for all the ratings are provided in the 2007 NPP Report.

In some cases a particular event was a major factor in a number of the ‘C’ grades for that station. For example, in 2007 there was an event involving the shutdown of all Station 3A units in order to restore functionality to a key electrical system. This event, and the findings of the subsequent investigation into the event, contributed to the ‘C’ ratings the station received in Organization and Plant Management, Quality Management, and Design.

Other ‘C’ grades are carried over from previous years, for example, implementation of the maintenance programme at Station 1A and implementation of the Equipment Qualification at Station 2. Licensees develop corrective action plans to address the issues, however it may take several years to fully implement the plans. It is understood by the licensee that the programme will continue to receive a ‘C’ grade until the corrective actions are complete. It’s essential that there is clear communication between the CNSC and licensee to ensure that appropriate actions are being taken to fully meet the objectives of CNSC requirements and performance expectations in these areas.

Safety area grades from the previous three years are provided in the report to indicate how a particular station is performing over time. Individual programme ratings for previous years are not listed; however, acknowledgement is given to those stations that have shown an improvement in a certain area. For example, in the 2007 report the Performance Assurance safety area is identified as an area of improvement for Station 1. The licensee had made improvements to their management system through a process and documents enhancement project. It was based on the achievements of this project that CNSC staff upgraded the documented Quality Management programme to ‘B’ for Station 1A and 1B for 2007. Implementation of the Quality Management programme at Station 1A was also upgraded to a ‘B’.

Similarly, the Human Factors programme at Station 5 was identified as another area of improvement with the grade for this programme increasing from a ‘C’ in 2006 to a ‘B’ in 2007.

5. Informing Canadians and CNSC activities

The NPP Report serves a number of purposes. Presented first to the Commission as a Commission Member Document and later published as an INFO Document, the NPP Report demonstrates to stakeholders how the CNSC is discharging its responsibilities to ensure that licensees have programmes in place to meet their licensing requirements. For the Commission, it also serves as a mid-term report for stations in the middle of their licensing period. As such, the assessment write-ups for these stations will also report on progress against commitments made by the licensee at time of licence renewal.

The annual process of assessing and rating each programme within the safety areas also helps CNSC staff focus on those areas that will require increased regulatory oversight. The CNSC has developed a baseline compliance plan, which is a set of compliance activities that must be undertaken by staff over the licence lifecycle (typically 5 years). Information is gathered through these activities to provide the CNSC with confidence that a station is operating safely and continues to be in compliance with regulatory requirements. A ‘C’ grade in a given programme indicates to staff that additional regulatory oversight in required in that area. The compliance work plan for that station would be adjusted accordingly.

6. Continuous improvement

In the spirit of continuous improvement, the CNSC is constantly looking for ways to improve the NPP Report as a communication product. A number of improvements are currently being considered, such as:

- Risk ranking the safety areas and applying risk-informed decision making to the rating process for a more transparent and methodical means of grading the facilities

- Decreasing the number of ratings by evolving the grades for ‘programme’ and ‘implementation’ into a single grade for ‘safety performance’

- Decreasing the size of the report by using a standard template and eliminating repetition

- Using plain language to facilitate understanding by the public.

The goal of the CNSC is to produce a report that is transparent, clear, concise and timely. It is anticipated that many of these enhancements will be internally reviewed and approved in time for the development of the 2008 NPP Report.

Table 2. Report card showing programme and implementation grades for all safety areas and programmes in 2007

Safety Area/Programme	P or I	Station 1 A	Station 1 B	Station 2	Station 3 A	Station 3 B	Station 4	Station 5
Operating	P	B	B	B	B	B	B	B
	I	B	B	B	C	B	B	B
Organization and Plant Management	P	B	B	B	B	B	B	B
	I	B	B	B	C	B	B	B
Operations	P	B	B	B	B	B	B	B
	I	B	B	B	C	B	B	B
Occupational Health and Safety (non-radiological)	P	B	B	B	B	B	B	B
	I	A	A	A	B	B	B	B
Performance	P	B	B	B	B	B	B	B
	I	B	B	B	C	B	B	B
Quality Management	P	B	B	B	B	B	B	B
	I	B	B	B	C	B	C	B
Human Factors	P	B	B	B	B	B	B	B
	I	B	B	B	C	B	B	C
Training, Examination, and Certification	P	B	B	B	B	B	B	B
	I	C	B	B	B	B	B	B
Design and Analysis	P	B	B	B	B	B	B	B
	I	B	B	B	B	B	B	B
Safety Analysis	P	B	B	B	B	B	B	B
	I	B	B	B	B	B	B	B
Safety Issues	P	B	B	B	B	B	B	B
	I	B	B	B	B	B	B	B
Design	P	B	B	B	B	B	B	B
	I	C	B	B	C	B	B	B
Equipment Fitness for Service	P	B	B	B	B	B	B	B
	I	B	B	B	B	B	B	B
Maintenance	P	B	B	B	B	B	B	B
	I	C	B	B	B	B	B	B
Structural Integrity	P	B	B	B	B	B	B	B
	I	B	B	B	B	B	B	B
Reliability	P	B	B	B	B	B	B	A
	I	B	B	B	B	B	B	B
Equipment Qualification	P	B	B	B	B	B	B	B
	I	B	B	C	B	B	B	B
Emergency Preparedness	P	A	A	A	A	A	A	A
	I	A	A	A	A	A	B	B
Environmental Protection	P	B	B	B	B	B	B	B
	I	B	B	B	B	B	B	B
Radiation Protection	P	B	B	B	B	B	B	B
	I	B	B	A	B	B	B	B
Site Security	P	Secret						
	I	Secret						
Safeguards	P	B	B	B	B	B	B	B
	I	B	B	B	B	B	B	B

P= programme I= implementation

EVOLUTION OF REGULATORY REQUIREMENTS FOR HWRS IN ROMANIA

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Abstract

The CNCAN, as safety authority of Romania, established an appropriate strategy for the integration of Canadian regulatory experience and practices into the Romanian licensing process for Cernavoda NPPs. Some important factors were taken into consideration by CNCAN in evolving the Romanian regulatory environment in the context of the post-accession process in the European Union, after 1st January 2007. The regulatory approach for the units 3 & 4 of the Cernavoda NPP will continue in the same manner as for the previous units 1 & 2. Future amendments to the licensing process will consider the compliance with the safety standards and guides, which are anticipated for 2014-2015 at the international level.

1. Introduction

Following from the success story of very good nuclear safety records of Cernavoda NPP, Units 1 & 2, the Romanian Government decided to continue the project by completing Units 3 & 4. These are important sources of energy needed cover national economic needs starting in 2014-2015. The Nuclear Safety Authority of Romania has already issued regulatory requirements for the licensing process, which is anticipated to start in 2009-2010. The main features of the licensing process developed by CNCAN for Cernavoda NPP, Units 1 & 2 and the experience gained by the regulatory body to will be applied to Units 3 & 4.

Some important factors were taken into consideration by CNCAN in evolving the Romanian regulatory environment in the context of the post-accession process in the European Union, after 1st January 2007. First category of regulatory requirements originates from the necessity to comply with the requirements imposed by the high level of safety standards planned for the 2014-2015 in Europe. The second category of regulatory requirements was developed by CNCAN — as a member of Western European Nuclear Regulators' Association (WENRA) — in the reactor regulations harmonisation process, which is envisaged to be completed by 2010. The third group of regulatory requirements was based on the experiences and practices discussed since 1997 during the CANDU Seniors Regulators Group meetings.

2. Legislative framework and evolution of the regulatory body

In 1969, by Decree no. 870/1969 of the State Council, which was approved by Law no. 7/1970, the State Committee for Nuclear Energy (CSEN) was established as the central body of State administration responsible for implementing the State's policy for the nuclear field. These responsibilities were later modified by the Decree of the State Council (DCS 282/1972), the role of CSEN in the regulation, licensing and control of nuclear activities being strengthened. For a long period of time, up to 1989, nuclear activities in areas such as promotion, development, and nuclear installation commissioning, operation and regulation were handled by CSEN. While CSEN had the overall responsibility for licensing and regulation, the ISCAN division within CSEN was responsible for performing inspection activities.

In 1974 Law 61/1974 on the deployment of nuclear activities in Romania was issued, which appointed the State Committee for Nuclear Energy as the central body of the State administration ensuring the implementation of the State's policy in the nuclear field. This strengthened and expanded the attributions of CSEN-ISCAN for the regulation, licensing and control of nuclear activities.

Although the responsibilities of ISCAN were formally consistent with international practices, in reality the authority of the regulatory organisation suffered because it was acting as a division of CSEN, which was responsible for both promotion/operation and regulation of nuclear activities. Despite these difficulties, the regulatory organization started to issue nuclear safety regulations based on Law No. 61/1974. These regulations were essentially based on the IAEA-NUSS series, and the provisions of the US 10CFR. Moreover, following a prescriptive approach, the Romanian regulatory organisation was fully involved in the control of all-important phases of the national nuclear programme.

In 1982 Law no. 6/1982 regarding quality assurance for nuclear units and installations was adopted, stating in part that "the State Committee for Nuclear Energy controls and is responsible for the accomplishment of the quality assurance requirements in design, manufacture, construction and operation of the nuclear units and installations, and of the execution of products and supply of services for these units and installations". Law 61/1974 and Law 6/1982 were replaced by Law no. 111/1996.

The State Committee for Nuclear Energy was dissolved by Decree 6/1990, and its responsibilities in the nuclear energy field were transferred to the Ministry of Electric Energy.

As the new institutional framework had created conditions to separate promotion/operation of nuclear energy from regulation activities, the National Commission for Nuclear Activities Control was established by Decree no. 29/1990 on the 8th of January, 1990, taking over the mandate and responsibilities of State Inspectorate for the control of nuclear activities and of quality assurance in the nuclear field.

The responsibilities and rules of operation of the National Commission for Nuclear Activities Control were established by Decree no. 221/1990, CNCAN becoming the central body of the State administration having national authority for the licensing and control of all the activities related to the peaceful use of nuclear energy. CNCAN had as its main duties the licensing of the siting, construction and operation of nuclear installations, and the control of the measures taken by licensees to ensure the protection of personnel, population and the environment from the harmful effects of ionizing radiation.

By Government Decision 983/1990, CNCAN reported to the Ministry of Environment. In 1996, Law 111 on the safe deployment of nuclear activities came into force. The national competent authority in the nuclear field — which was empowered to exercise the attributions for regulation, licensing and control in accordance with the Law 111/1996 — was the Ministry of Waters, Forests and Environment Protection, through the National Commission for Nuclear Activities Control. In 1997 by Government Decision no. 249/1997, the organisation and operating rules of the CNCAN were approved.

Law no. 16/1998 brought some amendments in relation to the observance of principles stipulated in the Convention on Nuclear Safety, ratified by Romania through Law no. 43/1995. Thus, the CNCAN was transferred from the Ministry of Waters, Forests and Environment Protection to directly report to the Government.

In 2004 CNCAN was separated from the Ministry of Waters and Environmental Protection, according to the Rule approved by the Governmental Decision 1627/2003 on the organisation and functioning of CNCAN.

3. Evolution of HWR regulatory requirements in Romania

The regulatory requirements developed by CNCAN for HWRs in Romania accounted practically for three phases of development. These stages took into consideration the national strategy regarding NPP to be built in Romania as follows:

- To achieve a high level of nuclear safety;
- To use, as much as possible, the national human resources and technical capabilities;
- To achieve a high degree of HWR technology integration in the Romanian nuclear industry;
- To reduce, as much as possible, the dependence by the outside energy sources;

- To implement standards and practices internationally recognized for nuclear energy.

In this context, since 1967 CNCAN taken into consideration, as references, regulations and practices from the IAEA, US NRC and CNSC-Canada. After 2001, a number of regulations were developed by CNCAN to harmonize the Romanian regulatory framework with the EU countries regulatory framework. The regulations developed in Romania are listed below:

- Nuclear Safety Requirements (NSR) — Nuclear Reactors and Nuclear Power Plants (1975), which contains provisions concerning licensing-basis documentation, site-evaluation criteria and design criteria for NPPs.

- Requirements for prevention and extinction of fires, applicable in the nuclear activities (1976);

- Nuclear Safety Requirements on Emergency Plans, Preparedness and Intervention for Nuclear Accidents and Radiological Emergencies (1993);

- Regulation on granting permits to operating, management and specific training personnel of Nuclear Power Plants, Research Reactors and other Nuclear Installations (2004);

- The set of regulations on Quality Management Systems for nuclear installations (NMC series, 2003), which contain provisions related to the quality assurance and safety of operation, maintenance, in-service inspection, testing, modifications, training of personnel, procurement activities, etc.

- Technical Prescriptions for Design, Execution, Assembling, Repair, Verification and Operation of Pipes under Pressure and of Elements of Pipes from Nuclear Plants and Facilities (NC2-83) issued by the State Inspectorate for Boilers, Pressure Vessels and Hoisting Installations (ISCIR).

Since the completion of the benchmarking, CNCAN has published the following regulations:

- Requirements on Containment Systems for CANDU Nuclear Power Plants (2005);

- Requirements on Shutdown Systems for CANDU Nuclear Power Plants (2005);

- Requirements on Emergency Core Cooling Systems for CANDU Nuclear Power Plants (2006);

- Requirements on Fire Protection in Nuclear Power Plants (2006).

- Requirements on Periodic Safety Review for nuclear power plants (2006).

- Requirements on Probabilistic Safety Assessment for nuclear power plants (2006).

The CNCAN requirements for special safety systems in HWRs (containment system, shutdown systems and emergency core cooling system) endorse the Regulatory Documents issued by the CNSC.

The main features and phases of the development process of Romanian regulatory requirements are presented in Table 1.

4. HWR licensing practices in Romania

The current licensing practice for Cernavoda NPP is based on the provisions of the Law and of the regulations issued by CNCAN. The requirements specified in the Law and the regulations are rather general and therefore a number of mechanisms are in place to ensure effective management of the licensing process. The detailed regulatory requirements, as well as the assessment and inspection criteria used by CNCAN in the licensing process, are derived from a number of sources, such as:

- Romanian regulations;

- Limits and Conditions specified in the different licences;

- IAEA Safety Standards and Guides;
- ICRP recommendations;
- Regulatory documents developed by CNSC and US NRC;
- Applicable Standards and Codes (CSA, ANSI, ASME, IEEE, etc.);
- Safety related documentation produced by the licensee and approved or accepted by CNCAN (e.g. Safety Analysis Reports, Safety Design Guides, Design Manuals, reference documents, station instructions, operating manuals, technical basis documents, etc.).

Table 1. Phases features in the development of regulatory requirements for HWR in Romania

Phase	Phase Period	Phase Features
Phase 1	1967-1989	• Prescriptive approach for regulatory processes; • Implementation of safety regulations based on IAEA safety guides and US NRC 10-CFR-50 regulations; • Implementation of quality assurance regulations based on IAEA safety guides and Canadian similar regulations; • 'System by system' approach for the licensing process of Cernavoda NPP; • Significant conflict of interest: Regulatory Body and Utility belongs the same Governmental Organization (CSEN);
Phase 2	1990-2006	• Prescriptive approach for regulatory processes; • Implementation of safety regulations based on IAEA safety guides, Canadian regulations, EU countries and the regulations and practices in place in the countries with HWR in operation; • 'Milestone' approach for the licensing process of Cernavoda NPP; • No conflict of interest between Regulatory Body and Utility;
Phase 3	2007-2015	• Implementation of EU countries experiences and practices for regulatory processes; • Continuation of the implementation of safety regulations based on IAEA safety guides, Canadian regulations, EU countries and the regulations and practices in place in the countries with HWR in operation; • 'Milestone' approach for the licensing process of Cernavoda NPP; • No conflict of interest between Regulatory Body and Utility; • Harmonisation process is in place within WENRA countries for the reactor regulations harmonisation; • Licensing process for Cernavoda NPP, Units 3 & 4.

The regulatory pyramid used by CNCAN in the licensing process is presented in Fig. 1. Apart from the formally issued (published) regulations, the requirements established by CNCAN in the licensing process are imposed through regulatory letters. Requirements and dispositions are also stated in inspection reports. The licensing submissions include, as the main document, a safety analysis report having content in accordance with the specifications established by CNCAN for each stage of the licensing process. In addition to the safety analysis reports, various supporting documents are submitted by the applicants to demonstrate the safety of the nuclear installation and the fulfilment of all the relevant legislative and regulatory requirements.

The review process performed by CNCAN is documented by one of the following means: evaluation reports, regulatory letters, inspection reports, containing findings and dispositions, written minutes as result of the licensing meetings (common meetings between CNCAN staff and the representatives of the licence holder or applicant). If the review concludes that the applicant has met all requirements, a licence is issue by CNCAN for a limited period of time (usually 2 years). All the limits and conditions derived for each specific case are clearly stated in the licence, which includes sections devoted to quality management, emergency preparedness, radiation protection, reporting requirements, compliance with licensing basis documents, the hierarchy of documents of the licensee, etc.

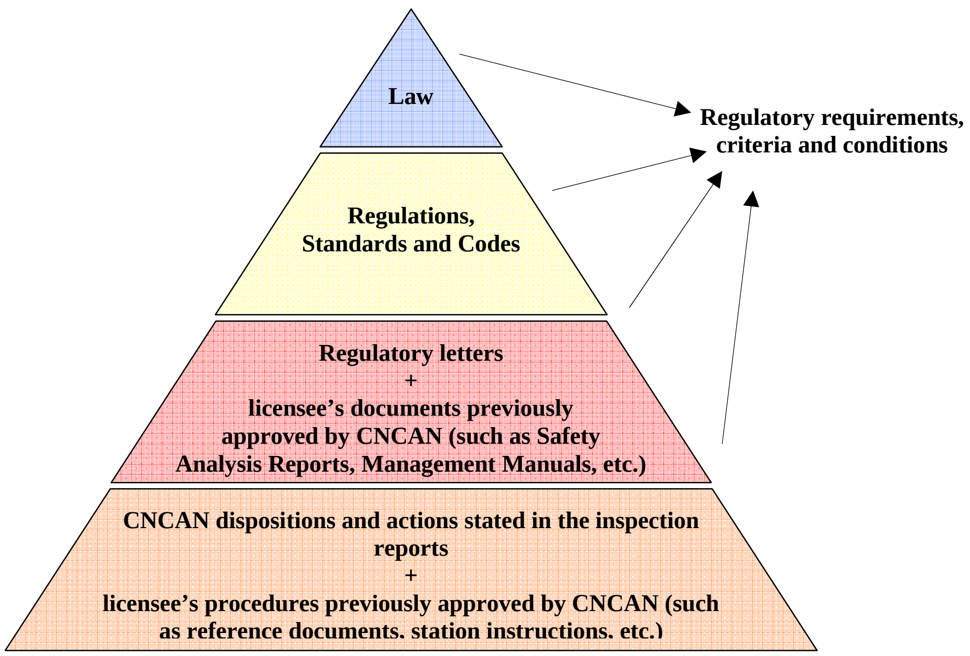

FIG. 1. Regulatory pyramid used by CNCAN in the licensing process.

CNCAN requirements for the Level II Licensing Schedule for CNE Cernavoda Unit 1, can be summarized as follows:

- CNCAN Requirements for FSAR to be in Compliance with the Licensing Level 2 Schedule;
- Commissioning Phase A 'Fuel Load' Prerequisites;
- Commissioning Phase B 'Criticality' Prerequisites;
- Commissioning Phase C 'Low to High Power Operation' Prerequisites;
- Commissioning Phase D 'Full Power Operation' Prerequisites;
- Construction Completion Assurance List;

- Commissioning Completion Assurance List;
- General CNCAN Requirements for the Cernavoda NPP 1 Licensing Level 2 Schedule;
- Safety Compliance Assurance — Basic Requirements;
- Application Contents for Milestones:
- Receive, Store & Use Radioactive Sources;
- Receive & Store Heavy Water;
- Receive & Store Fuel;
- Load Heavy Water Into Moderator System;
- Commissioning Licence;
- Load Fuel;
- Load Heavy Water into PHT System;
- First Criticality;
- Power Increase from Low to High Power;
- On Power Refuelling Demonstration;
- Operating Licence.

The licensing documents explicitly depict the way in which the correspondence between Canadian and Romanian standard requirements must be assured. Meantime, the documents demonstrate that these approaches are not in contradiction. The licensing documents are expected to:

- ensure the completeness of FSAR as a license document;
- ensure the general compliance with the Romanian regulations;
- ensure the compatibility with the Canadian and Romanian operating practices;
- ensure an appropriate Regulatory/Utility interface during the licensing process.

In order to comply with legal requirements and strengthen the connection to reality in the field, and in conformance with the provisions of Law 111/1996, a procedure named: 'Safety and Compliance Assurance procedure' (SCA) had to be issued by the Utility. The SCA basic principles were as follows:

- SCA is to demonstrate compliance with the Romanian regulations during construction, commissioning and operation of the plant. SCA will be sent to CNCAN after finalizing the turnover (T/O) process for each system;
- SCA has to demonstrate that the safety requirements stated in the FSAR are actually met. The revision process should be developed simultaneously with the preparation of the Design Completion Assurance (DCA), Construction Compliance Assurance (CtCA) and/or CCA type documents. In this respect, a document called 'Outstanding Issues List' will be prepared for each system.

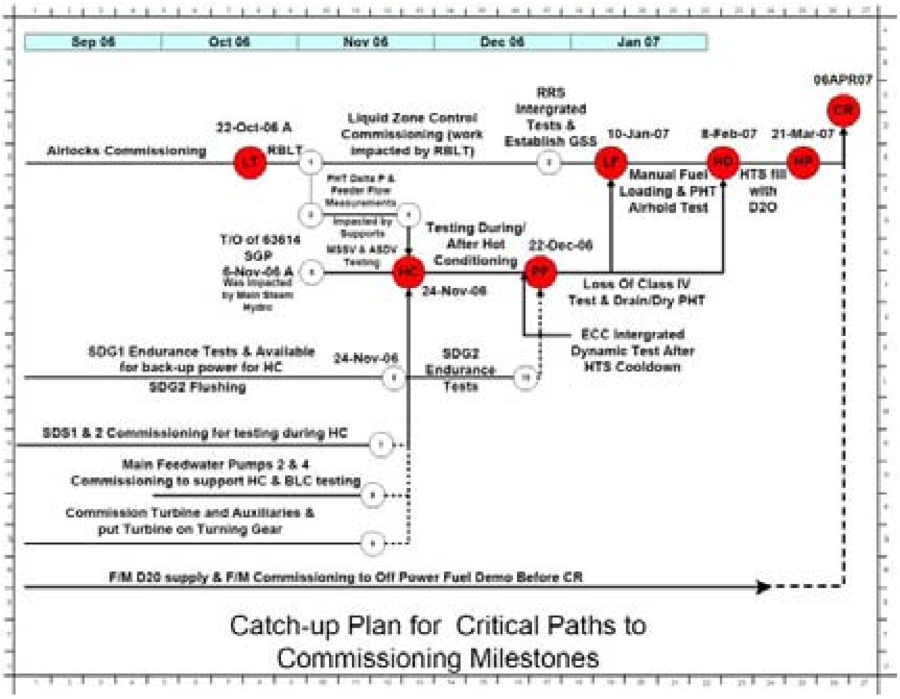

FIG. 2. Critical path to the Unit 2 commissioning milestones status, assessed by regulatory body and utility in the monthly licensing meetings.

5. Conclusions

The evolution of regulatory requirements for HWRs in Romania has a very positive trend. The CNCAN, as safety authority of Romania, established an appropriate strategy for the integration of Canadian regulatory experience and practices into the Romanian licensing process for Cernavoda NPP. The regulatory approach for the units 3 & 4 of the Cernavoda NPP will continue in the same manner as for the previous units 1 & 2. Future amendments to the licensing process will take into consideration compliance with the safety standards and guides, which are anticipated for 2014-2015 at the international level.

DEVELOPMENT OF RISK-INFORMED REGULATORY POSITIONS ON CANDU SAFETY ISSUES, *PART I: METHODOLOGY DEVELOPMENT FOR RISK ESTIMATION AND EVALUATION*

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Abstract

Risk informed decision making is used by the Canadian Nuclear Safety Commission as a management tool to support decisions related to licensing, compliance and planning/resource allocation. The decision-making process provides information on the risks related to a certain decision, produces recommendations for controlling the risk, and provides for implementation of the risk-control measures and for monitoring the impact on risk following a decision. This paper describes the methodology developed for the estimation and evaluation of the risks used as an input into the risk-informed decision making process. The methodology allows for consideration of all risks in a consistent manner; it recognizes the differences between the risks in the absence of the safety issue and the risks when the safety issue is present. The methodology refers to safety related risks, although similar considerations can be made for other types of risks.

1. Introduction

The Canadian Nuclear Safety Commission (CNSC) regulatory approach considers risk in both licensing requirements and in decision making in a manner consistent with the recommendations of the Treasury Board of Canada. This approach includes the Risk-Informed Decision Making (RIDM) process that provides the decision makers with information on the risk environment, and recommends risk control measures. The process takes into account all the relevant risks associated with an issue to support decisions in areas of licensing, compliance, and planning and resource allocation. It thus follows that the safety risks (i.e. radiological effects on public for design basis accidents and risks associated with severe accidents) are not the only risks considered. Other sources of risks, related to the CNSC mandate and objectives, including environmental and organizational risks, are also accounted for in the decision-making process.

There is no one single methodology universally applied for consideration of risk in regulatory activities. Significant differences between the approaches in various jurisdictions exist [1-25]. These differences typically originate from specifics of the national regulatory environment and practices. For example, the RIDM process developed by CNSC staff is based on the Canadian Standard Association document on risk management, CSA Q-850 [26]. In this process, risk-control measures are recommended based on risk estimation and evaluation. Risk management is then accomplished by implementation of measures to address these risks and monitoring the impact of these measures. Provisions for adequate documentation, transparency, consultation with stakeholders and consideration of feedbacks, are included.

The RIDM process is briefly described in Section 2. The methodology for the estimation and evaluation of the risks uses risk matrices: it involves establishing for each risk area categories of consequences and likelihoods, and evaluation of the significance of the risk. The methodology, as applied to safety issues, is described and discussed in Section 3.

An application of the RIDM process and tools to outstanding CANDU safety issues is discussed in detail in Part 2 of this paper [27]. The overall objective of this work was to identify, estimate and evaluate the risks associated with each safety issue and to recommend measures to control the risks.

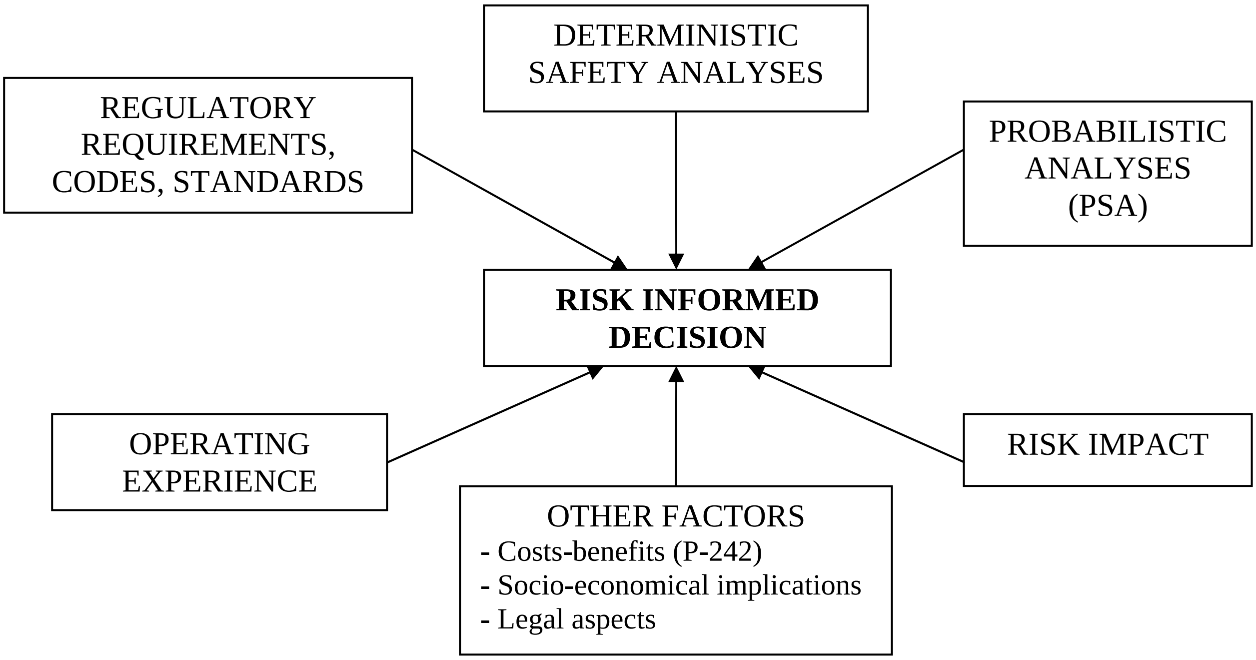

FIG. 1. Inputs for risk informed decision. 2. Risk-informed decision making process.

2. Risk-informed decision making process

The RIDM provides the decision makers with information on the risk environment, and recommends risk-control measures. Of particular interest for the risk-informed regulatory approach are safety issues (including but not limited to design and analysis issues, operational occurrences, inspection findings, new knowledge) that may result in increased risks related to power-plant operation. It must be realized, however, that the risk associated with safety issues is not the only input to be considered by the decision maker, as depicted in Figure 1. Other inputs (e.g. deterministic and probabilistic insights, past practice, operational experience, etc.) may be equally or more important, depending on the risk environment. For example, for the life extension of power plants, economical aspects play an important role in developing an overall safety improvement plan.

The RIDM process is shown schematically in Figure 2. It can be seen that, prior to the decision, the main activities include:

- Set up the team, identify constraints (i.e. time and resources available), define the issue, identify stakeholders (Step 1);

- Performing initial analysis: identify hazards and risks, assess whether immediate measures need to be taken (Step 2);

- Estimation of the identified risks that is, assessment of the magnitude of the consequences, identification of the risk scenarios leading to those consequences, and assessment of the likelihood of the risk scenarios/consequences (Step 3);

- Evaluation of the risks, that is, determining the significance level of the risks (Step 4); and

- Recommendation of measures to control the risks (Step 5).

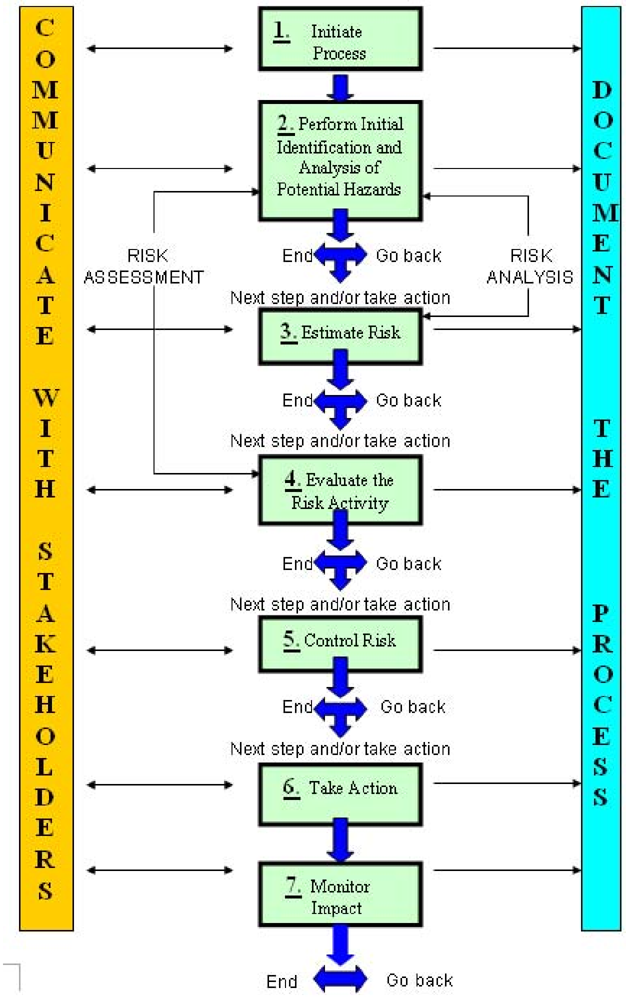

FIG. 2. CNSC risk informed decision making process.

The documentation of all these activities, including the recommendations on how to address the risks, is then provided to the decision maker. Following the decision, the RIDM process provides for:

- Implementation of actions to reduce the risks (Step 6);
- Monitoring the impact of these actions (Step 7).

Throughout the process, the work should be documented and stakeholder communication carried out. Feedback should be considered as appropriate, including return to previous steps as needed.

3. Methodology for risk estimation and risk evaluation

Referring to Figure 2, the risk estimation and evaluation (RE&E) methodology is applied as part of the Step 3 ‘Risk Estimation’ and Step 4 ‘Risk Evaluation’ of the RIDM process. This activity is performed by the team established in Step 1, but consultation with other specialists (i.e. subject matter experts) should be arranged if needed.

The methodology described here uses risk matrices; it involves establishing consequences and likelihoods categories for each risk area, and evaluation of the significance of the risk. Estimation and evaluation of risks involves qualitative judgments although numerical data can be used if readily available.

3.1. Underlying principles

In the development of the risk-informed decision-making process, special attention was given to ensuring that the RE&E methodology is clearly aligned with the CNSC regulatory framework. This was done by:

- Ensuring that the risks identified in Step 2 of the RIDM process are consistent with the CNSC logic model, objectives and mandate;

- Use of the CNSC regulatory requirements (such as frequency classification of events and dose limits for anticipated operational occurrences, limits for Safety Goals, reliability of special safety systems, dose limits for workers, environmental release limits) for estimation of risks, and as the basis for defining the risk significance levels on the tolerability scale;

- Considering the CNSC licensing ratings;

- Use of concepts specific to the CNSC regulatory approach and integration of the previous CNSC work on determining safety significance and risk significance.

It was also recognized that the RE&E methodology needed to be generally applicable to regulatory assessments of risks, including:

- Risks associated with outstanding safety issues (e.g. radiological risks to public in design basis conditions, severe accident risks, and health and safety risks to workers, environment risks due to radioactive releases and releases of hazardous substances);

- Risks associated with licensing or compliance decisions (e.g. introduction of new requirements); and

- Non-safety risks such as organizational risks, legal risks, risks related to meeting international obligations, safeguard and security risks, etc.

The approach recognizes the differences between the risks in the absence of the safety issue and the risks when the safety issue is present (consequently, it allows for consideration of all risks in a consistent manner). For safety related risks it is assumed that the CNSC safety requirements are equally important and that the risks associated to the CNSC requirements are equally tolerable. Estimation of the consequences and of their probabilities is done, as far as possible, following a realistic best-judgment approach rather than a conservative worst-case-scenario approach. As comprehensive risk analyses may be prohibitive in terms of resources needed, qualitative estimates are used when applicable data are not readily available.

3.2. Risk tolerability scale and risk significance levels

The RE&E methodology is based on using the Risk Tolerability Scale (RTS) [5], which was selected as the unique risk matrix for risk evaluation. It involves assigning significance levels to the risks identified and estimated based on their tolerability levels, regardless of the nature of those risks. Use of the risk tolerability as a unique scale has the following advantages:

- it ensures consistency in determining the risk significance;

- it permits comparing risks in different areas via their significance level/tolerability and avoids assigning weighting factors for risks in different risk areas; and

- it allows taking into consideration the subjective perception of the risk by experts in diverse risk areas, without affecting consistency in determining the risk significance levels for risks in diverse risk areas.

The following three regions are defined in Figure 3: acceptable risk, tolerable risk, and intolerable (unacceptable) risk. Activities that have risks in the unacceptable region should not be permitted unless risk control measures are implemented to lower the level of risk into the tolerable or acceptable range. Within both the tolerable and acceptable ranges, additional risk control measures should be taken if it is reasonably practical to do so. Generally, if the risk is judged to be negligible, additional efforts for risk reduction may not be justified.

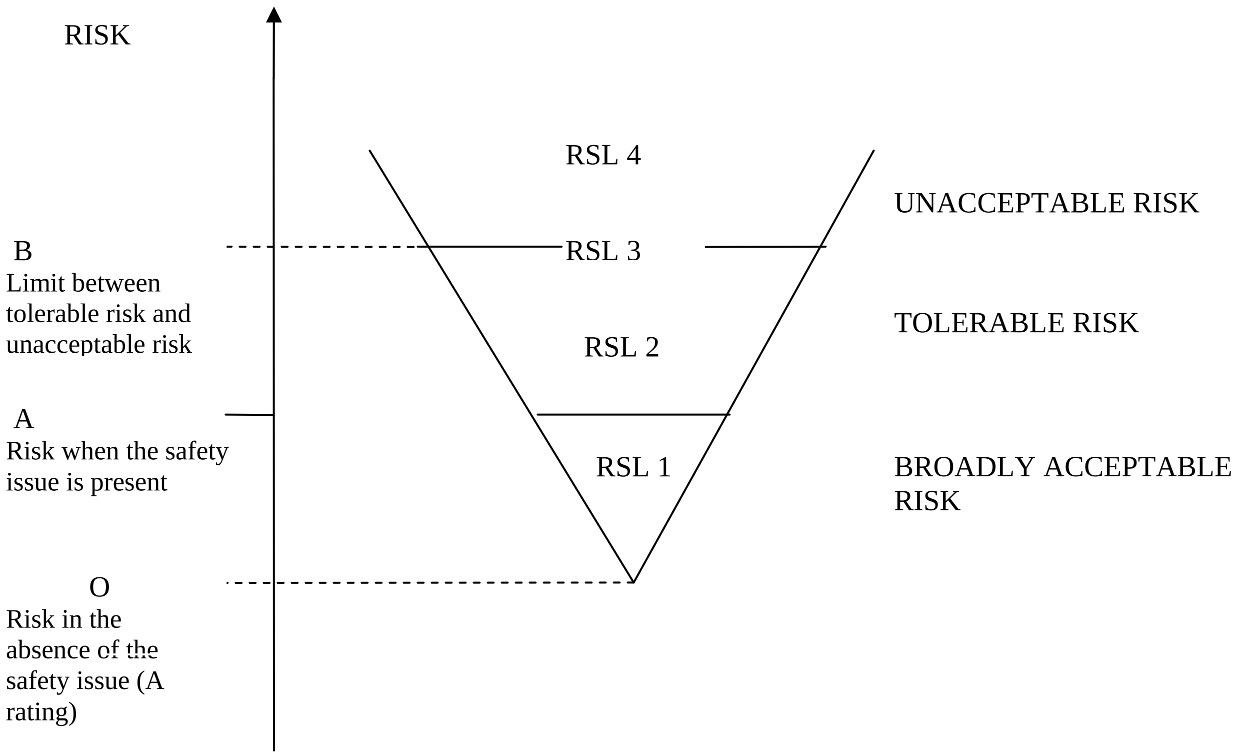

FIG.3. Risk tolerability scale and risk significance levels.

It is assumed here that the CNSC relevant requirements are determining the limit separating tolerable risks from unacceptable risks (point B in Figure 3). The point O on the Figure 3 represents the case when the safety issue is absent. From the risk point of view, an 'A' licensing rating can be awarded as there is sufficient margin (OB) to the limit to unacceptable risk (in CNSC regulatory approach, 'A' is the highest rating where no special compliance activities are needed).

When a safety issue occurs, the risk associated with a safety issue contributes to the increase of the total risk (i.e. point A in Figure 3). The RE&E recognizes that the total risk (i.e. point A in Figure 3) has two components: the risk due to the existence of the plant in the absence of the safety issue (i.e. point O in Figure 3) and the incremental risk associated to the safety issue. The significance of the risk in the presence of the safety issue is given on the tolerability scale by the distance on the Risk Axis between A and O or between A and B.

Within each risk area, the RE&E is based on using risk matrices for defining the Risk Significance Level (RSL) of the risks. The RSL is given by the magnitude of the impact on the risk. The correspondence between the risk significance levels on the Risk Tolerability scale is represented in Figure 3. Four risk significance levels are employed:

- RSL 1 — a safety issue brings no additional risk or a negligible one;
- RSL 2 — the risk associated with the safety issue increases, but it is still well within the tolerable region;
- RSL 3 — the risk evaluated to be at the border between the tolerable and the unacceptable regions; and
- RSL 4 — the risk evaluated to be not acceptable.

It is important to note that the consequence criteria, the likelihood criteria and the RSL for the elements of the risk matrices are independent from the issue under study; the same matrix and criteria for consequences/likelihoods have to be used where applicable. This ensures that risks within a risk area for various issues/applications will receive consistent evaluation. There is full flexibility on the size of the risk matrices and on the level of detail for defining the consequence and likelihood criteria.

It is important also to note that introduction of new regulatory requirements changes the perspective for evaluation of the plant risks and it may raise compliance issues, and hence licensability risks. The main elements for defining the RSL on the tolerability scale are represented in Figure 4.

The principal impacts of the new requirements are changing the location of the limit between Tolerable Risk and the Unacceptable Risk, and the conditions that would guarantee an A licensing rating. For the existing requirements, these are represented in Figure 4 by the point B and respectively O.

If the new requirements are more demanding than the old (or existing) ones, B will have a lower position (i.e. B') as a lower risk will be considered acceptable. While the change of the requirements will not induce additional risks in the plant (assumed at the point O), the margin to the unacceptable risk is reduced (OB' < OB), showing difficulties to comply with the new requirements. Although the risk from the plant is unchanged, the licensing rating will be affected proportionately with the reduction of the margin to unacceptable risk, leading to increased RSL for licensability risks.

If the new requirements are more relaxed than the old (or existing) ones, then the margin from O to the unacceptable risk increases (OB' > OB in Figure 4). Although initially the risk from the plant will be unaffected, the extra-margin will likely be permitted to erode during plant operation and the previous margin will be re-established; the risk from the plant will likely increase (from O to O' in Figure 4). All the additional risks associated to the differences between the old (existing) and the new requirements should be considered. At an extreme, complete 'Risk informing the new regulatory requirements' may be needed.

For RDIM purposes, the risks associated with the impact of the change of the requirements have to be identified, estimated and evaluated; on this basis, appropriate control measures may be recommended.

3.3. Risk matrices

The risk matrices are defined in the risk plane (Consequences, Likelihood) by quantifying the continuum range of consequences and likelihoods into a finite number of categories defined by appropriate criteria. A risk matrix has to be used for the estimation of the risk and evaluation of the significance level for each Risk Area identified in Step 2 (Figure 2) of the RIDM process.

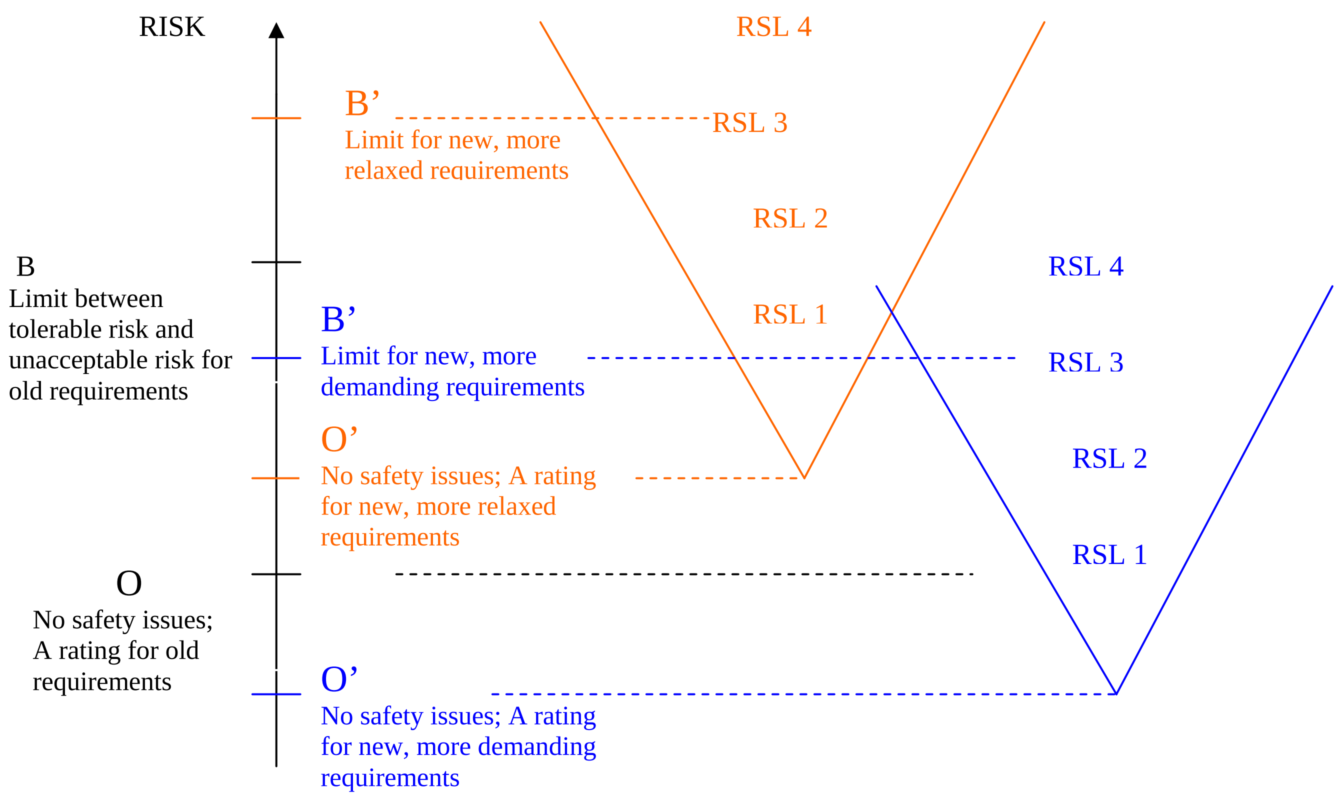

FIG. 4. Risk tolerability scale for introduction of new regulatory requirements.

The development of a risk matrix involves the following steps.

- Identification, for each risk area, of the potential consequences in the range of interest. Divide this range of consequences into a manageable number of consequence categories and define the criteria for defining these categories. Usually three categories suffice, but there is no formal restriction.

- Identification, for each risk area, of the range of likelihoods for the consequences, and define the criteria for likelihood categories. The likelihood can be expressed in terms of probability, frequency events per reactor year, fraction of time when the negative outcome may occur, chance, etc. As for the consequence categories, three likelihood categories would normally be sufficient but there is no formal restriction...

In the (consequence, likelihood) coordinates, the criteria for consequence categories and the criteria for the likelihood categories define the risk matrix. A risk significance level is assigned to each element of the matrix according to the indications given in Section 3.2, and to the tolerability of the risk given by the particular consequence/likelihood criteria on the tolerability scale. The risk matrices are specific to each risk area, but independent of the issue under study. once developed, the risk matrices may be used, as applicable, for risk evaluations in that risk area. The risk matrices are used to determine the risk significance level of the appropriate risks, as described below:

- Identify the available risk matrices (including the criteria for the consequences and respectively the likelihood categories) to be used. For risk areas where no risk matrix is already available, or for situations when a better description is desired, new risk matrices, and/or specific definitions for the criteria to define new consequences categories or likelihood categories will have to be developed.

- Using the results of the risk estimations (i.e. the assessment of the consequences and of their likelihood), the applicable consequence criteria and likelihood criteria and thus appropriate location in the risk matrix are determined. The risk significance level is given by the RSL of that element of the risk matrix.

3.4. Risk matrices for risk estimation and evaluation in selected risk areas

The consequence and likelihood ranking criteria, and risk matrices for selected risk areas are described below for illustrative purposes.

- Risk of increased off-site radiological consequences to public, Tables 1 to 3. The ranking criteria for Risk to severe accidents are given in Table 4. This is a rather unique case, where the PSA assessments provide directly risk indicators (i.e. safety goals) such that separate criteria for consequence categories and for likelihood categories are not needed.

- For the situations when the above matrices may not be readily applicable (i.e. it is difficult to establish a direct link between the safety issue and radiological consequences or it is difficult to demonstrate that all risk scenarios were considered), qualitative consequence criteria, likelihood criteria and risk matrix to assess risk of negative impact of the issue on safety are also given (Tables 5, 6, 7).

- Risks to environment due to radioactive releases and spills of hazardous substances, and health and safety risk to workers, Tables 8, 9 and 10. In these tables, existing requirements and experience with the criteria for safety significance level were directly considered.

- The criteria in Tables 11, 12 and 13 are an example for handling non-technical risks, in this case organizational risks for the regulator (reputation loss, not meeting its objectives and obligations). These criteria could be used for activities such as regulatory responses to industry initiatives, new initiatives and programmes originated in the CNSC, and, in general, any other project needed to in accordance to CNSC objectives and obligations.

- The following should be noted here:

- It may happen that, for a given safety issue, rankings of risk in various risk areas will result in different RSLs. In this case, the final significance level for the issue will be established using expert judgment.

- The RSL for a safety issue can be affected by other, unsolved, concurrent issues (i.e. the issues are affecting simultaneously active for the same unit). This can happen when different safety issues are enhancing each other's impact on risk. A possible approach to deal with this type of situation is to re-define the safety issue at a higher level of generality to include the related sub-issues; the risks associated to the generic issue will then be estimated and evaluated.

Table 1. Criteria for consequence categories for radiological risk to public at DBA

Consequence Category	Criteria
C1	- No significant additional radioactive releases would occur, such that the public doses calculated in the existing Safety Report are expected to be bounding.
C2	- The radioactive releases would lead to public doses greater than those determined in the Safety Report, but still less the limits for the applicable class of the accident.
C3	- The radioactive releases would lead to public doses may exceed the limits for the applicable class of the accident. The releases would not trigger off-site protection measures.
C4	- The radioactive releases would require initiation of off site protection measures.

Table 2. Criteria for likelihood categories for radiological risk to public at DBA

Likelihood Category	Criteria
L1	- Frequency of accident scenario is greater than 10^{-7} /year but less than the DBA frequency limit; the accident is beyond design basis.
L2	- Frequency is not significantly different from, or is the same as that, originally assigned in the Safety Report; re-classification of the event is not necessary.
L3	- Frequency of the accident scenario is significantly greater than that considered in the Safety Report; the accident sequence may have to be re-classified into a higher frequency category (example: from Class 3 to Class 2 in C-6, or from DBA to AOO in S-310).

Table 3. Risk matrix for radiological risk to public at DBA

CONSEQUENCE				
	C4	3	4	4
	C3	1	3	4
	C2	1	2	3
	C1	1	1	2
		L1	L2	L3
		LIKELIHOOD		

Table 4. Risk matrix for severe accidents risks

Risk Level	Criteria
1	- The increase of the Safety Goals[1] is negligible. - The Safety Goals remain below the accepted targets.
2	- All or some Safety Goals increase, but remain less than the accepted limits.
3	- All or some Safety Goals may exceed the accepted limits.
4	- All or some Safety Goals significantly exceed accepted limits

Table 5. Qualitative criteria for consequence categories for risk of negative impact on safety

Consequence Category	Criteria
C3	- Defense in depth is insufficient or unacceptable (one or more levels of protection are lost, the safety function is disabled) - Impossibility (i.e. lack of knowledge, data, tools) to assess conditions relevant for safety when compliance verification is impossible - Continuous deterioration of plant safety - Excessive increase of the time at risk of plant operation
C2	- Defense in depth is degraded (one or more levels of protection are affected, the safety function is impaired) - Difficulty (i.e. insufficient information, data, tools) to assess conditions relevant for safety when compliance verification is impossible - Incomplete restoration of safety - Significant increase of the time at risk of plant operation
C1	- Levels of protection/safety functions are affected but not significantly - Inadequate confidence in accuracy of data, models and code predictions - Non-sustainable long term safe operation - Increase of the time at risk of plant operation

[1] For the purpose of this table, the Safety Goals are quantitative risk indicators specific to severe accident conditions, calculated in PSA. The quantitative Safety Goals defined in CNSC document RD-337 are Core Damage Frequency, Large Release Frequency and Small Release Frequency

Table 6. Criteria for likelihood categories for risk of negative impact on safety

Likelihood Category	Criteria
L3	- The consequences will very likely occur (> 75% chance)
L2	- The consequences will likely occur (25%–75% chance)
L1	- The occurrence of consequences is unlikely (< 25% chance)

Table 7. Risk matrix for risk of negative impact on safety

CONSEQUENCES				
	C3	3	4	4
	C2	2	3	3
	C1	1	2	3
		L1	L2	L3

LIKELIHOOD

Table 8. Criteria for consequence categories for health and safety risks to workers and for risks to environment due to radioactive releases and spills of hazardous substances

Consequence Category	Health and Safety Risks to Workers	Risks to Environment due to Radioactive Releases and Spills of Hazardous substances
C3	• Severe health effects or death of a worker	• Releases of nuclear substances significantly higher than the Derived Release Limits (DRL) requiring implementation of off-site protection measures. • Releases of hazardous substances requiring implementation of off-site protection measures.
C2	• Exposure of a person, organ or tissue to radiation may exceed the applicable radiation dose limits prescribed by the Radiation Protection Regulation • Doses to workers may be greater than the limits specified in CNSC regulations (unplanned dose of above 1 mSv to an individual, or a collective dose of above 5 mSv).	• Releases of nuclear substances may exceed DRL • Uncontrolled release of nuclear substances (rate, amount, name), or releases through an unauthorized point, without exceeding DRL. • Releases of hazardous substances reportable immediately to CNSC.
C1	• Lost time injury • Undue staff radioactive exposure or contamination	• Releases of hazardous substances that have the potential to harm the environment (not reportable immediately to CNSC). • Event (such as fire, explosion) leading to releases of hazardous substances.

Table 9. Criteria for likelihood categories for health and safety risks to workers and for risks to environment due to radioactive releases and spills of hazardous substances

Likelihood Category	Criteria
L3	Frequent occurrences expected — several times during the life of the plant.
L2	It can happen once or very few times during the life of the plant.
L1	Unlikely during the life of the plant.

Table 10. Risk matrix for health and safety risks to workers and for risks to environment due to radioactive releases and spills of hazardous substances

CONSEQUENCE			
C3	3	4	4
C2	2	3	3
C1	1	2	3
	L1	L2	L3

LIKELIHOOD

Table 11. Criteria for consequence categories for organizational risks

Consequence Category	Criteria
C3	• loss of public trust in organization's capability to deliver its mandate • strong criticism by review agencies
C2	• perception of organization's inability to ensure safety • perception of conflicts of interest • strong negative media attention • criticism by review agencies
C1	• some unfavourable media attention • some unfavourable observations by review agencies • perception of excessive regulation (excessive regulatory risks)

Table 12. Criteria for likelihood categories for organizational risks

Likelihood Category	Criteria
L3	• The consequences are expected to occur in most circumstances (> 75% chance)
L2	• The consequences should occur sometimes (25%–75% chance)
L1	• The occurrence of consequences is unlikely (< 25% chance)

Table 13. Risk matrix for organizational risks

CONSEQUENCES			
C3	3	4	4
C2	2	3	3
C1	1	2	3
	L1	L2	L3
		LIKELIHOOD	

4. Summary

The paper discusses the main results of the work leading to the development of the methodology for estimation and evaluation of the significance of the risks. The estimation of the risk is qualitative; it is based on a best-estimate judgment, using risk matrices for consequences and likelihood of their occurrence. The risk tolerability scale was selected as unique risk metric to ensure consistency in evaluation of the risks of diverse nature. Four risk significance levels defined according to their tolerability. As an example, risk matrices for selected risk areas are presented.

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REFERENCES

[1] RAHBAR, E.K., SEWELL, R.T., ERIKSON, H., “A new approach to development of a Risk Based Safety Performance Monitoring System for Nuclear Power Plants”, Specialist Meeting on Safety Performance Indicators, Madrid, Spain, 17-19 October 2000.

[2] SANYASI RAO, V.V.S., BABAR, A.K., SARAF, R.K., “Probabilistic Safety Criteria: An Approach for Indian Nuclear Power Plants”.

[3] RAHBAR, M.K., “Use of Probabilistic Safety Criteria within the Framework of Nuclear Power Risk Management”, Proceedings of ESA Workshop on Risk Management (1998), European Space Agency (ESTEC), The Netherlands.

[4] “Regulatory control of nuclear safety in Finland”, STUK-B-YTO 233, July 2004.

[5] “Reducing Risk Protecting People”, Health and Safety Executive’s (HSE’s) decision making process, 2001.

[6] “Safety Assessment Principles for Nuclear Plants”, Health and Safety Executive1992
[7] Convention on Nuclear Safety National Report of Japan for the Third Review Meeting (Provisional Translation), August, 2004.
[8] Performance Indicators, Appendix E, INPO AP-928.
[9] D’EER, A.M., “Identifying measures to balance the risk profile of the Tihange 2 NPP”, INIS-FR-809.
[10] VRBANIC, I., *et.al.*, “The profile of quantitative risk indicators in Krsko NPP”, 5th International Conference on Nuclear Options in Countries with Small and Medium Electricity Grids, Dubrovnik, Croatia, May 16-20, 2004.
[11] CHAKRABORTY, S., *et.al.*, “Risk-based Safety Performance Indicators for Nuclear Power Plants”, Transactions of the 17th International Conference on Structural Mechanics in Reactor Technology (SMiRT 17), Prague, Czech Republic, August 17-22, 2003.
[12] VERSTEEG, M.F., ANDREWS, R.M., “Consideration of Probabilistic Safety Objectives in OECD/NEA Member Countries; Short Overview and Update”, IAEA—J4-TC-803.3, Technical Committee Meeting on Use of Probabilistic Safety Assessment in the Regulatory Process, Vienna, Austria, 5-8 Dec. 1994.
[13] HOLMBERG, J.E., “A Utility Theoretic View on Probabilistic Safety Criteria”, Helsinki University of Technology Systems Analysis Laboratory Research Reports, A64, March 1997.
[14] FARMER, F.R., “Siting Criteria — A New Approach”, Proceedings of the IAEA Symposium on the Containment and Siting of Nuclear power Reactors”, Viena, April 1967, IAEA SM-89/31.
[15] BUJOR, A., “Discussion of the Concept of Safety Indicators from the Point of View of TfUX2 Accident Sequence for Forsmark 3”, RISOE-M-2926, January 2001.
[16] BRIGHTON, P.W.M., “Risk Concepts in UK Nuclear Safety Decision-Making”, IAEA-CN-82/18.
[17] OFFICE FEDERAL DE L’ENVIRONNEMENT, DES FORETS ET DU PAYSAGE (OFEFP), Manuel I de l’ordonnance sur les accidents majeurs OPAM, CH, Juin 1991, CNSC Library N100038, HV 551.5.S94 M36, 1991.
[18] Validation of Risk-Based Performance Indicators: Safety System Function Trends, NUREG/CR-5323, BNL-NUREG-52186, 1989.
[19] EMRIT, R., RIGGS, R., MILSTEAD, W., PITTMAN, J., “A Prioritization of Generic Safety Issues”, US NRC NUREG-0933, April 1999.
[20] INTERNATIONAL ATOMIC ENERGY AGENCY, Ranking of Safety Issues for WWER-440 Model 230 Nuclear Power Plants, IAEA-TECDOC-640, February 1992.
[21] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Issues and their Ranking for WWER 440 model 213 Nuclear Power Plants, IAEA, EBP-WWER-03, WWER-SC-108, August, 1995.
[22] INTERNATIONAL ATOMIC ENERGY AGENCY, Generic Safety Issues for Nuclear Power Plants with Light Water Reactors and Measures for their Resolutions, IAEA-TECDOC-1044, September 1988.
[23] VESELY, W.E., *et.al.*, Measures of Risk Importance and their Application, NUREG/CR-3385, BMI-2103, July 1983.
[24] CHAKRABORTY, S., GONEN, Y.G., VERSTEEG, M.F., “Technical Note: Quantitative Safety Guidelines (QSGs) and probabilistic safety Assessment (PSA) in the OECD Countries”, Nuclear Safety, Vol.32, No.2, April-June 1991.
[25] INTERNATIONAL ATOMIC ENERGY AGENCY, Risk Informed Regulation for Nuclear Facilities: Overview of the Current Status, IAEA-TECDOC-1436, 2005.
[26] “Risk Management: Guideline for Decision Makers”, National Standard of Canada CAN/CSA Q-850-97, September 1997.
[27] MILLER, D., RZENTKOWSKI, G., BUJOR, A., “Development of Risk-Informed Regulatory Position on CANDU Safety Issues. Part II: Application of Risk-Informed Decision Making for Assessment of Safety Issues”, IAEA Topical Meeting on Advanced Safety Assessment Methods for Nuclear Reactors, Daejon, Republic of Korea, October 30 — November 2, 2007.

DEVELOPMENT OF RISK-INFORMED REGULATORY POSITIONS ON CANDU SAFETY ISSUES, *PART II: APPLICATION OF RISK-INFORMED DECISION MAKING FOR CATEGORIZATION OF SAFETY ISSUES*

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Abstract

The paper describes the approach taken by staff of the Canadian Nuclear Safety Commission to identify the list of outstanding safety issues for Canadian CANDU reactors, and the application of the Risk-Informed Decision Making for developing risk-informed regulatory positions, including the path forward for resolution of each safety issue in view of currently operating reactors, life extension of existing reactors and new reactors. A general categorization of the issues is outlined here, and two specific examples presented for the risk identification, estimation, and evaluation with the use of the methodology described in Part I of this paper. Measures to control the risk are also recommended. It has to be realized that the issues identified in this paper, and associated risks, should not be viewed as questioning the safety of operating reactors, which have attained a very high operational-safety record, but rather as areas where uncertainty in knowledge exists, where the safety assessment has been based on conservative assumptions, and where regulatory decisions are needed, or will need to be confirmed. It is also important to note that some of the safety issues identified here are common to other reactor types as well.

1. Introduction

The regulatory oversight of currently operating reactors, as well as the occurrence and recurrence of events, deviations from current international practice in design and operation, and results of probabilistic safety assessment (PSA) studies provide information on weaknesses in plant safety, and provide insights into corrective measures to resolve them. As such, these weaknesses, or ‘safety issues’ are used as a reference to facilitate the development of plant-specific safety improvement programmes. These programmes are based not only on deterministic and probabilistic considerations, but also risk insights and cost-benefit arguments.

This document describes the approach taken by staff of the Canadian Nuclear Safety Commission (CNSC) to identify the list of key outstanding safety issues for Canadian CANDU reactors, and the application of the risk-informed decision making (RIDM) for developing risk-informed regulatory positions, including the path forward for resolution of each safety issue in view of currently operating reactors, life extension of existing reactors and new reactors. The approach used for identification of safety issues, and their initial categorization and determination of the risk significance is described in Section 2. An application of the RIDM process to assess the adequacy of the emergency core cooling system (ECCS) sump screen design and moderator temperature predictions is presented, as an example, in Section 3. The RIDM process leads to the identification of risk-control measures based on risk estimation and evaluation. The risk management is then accomplished by implementation of measures to address these risks and monitoring the impact of these measures. This is described in more detail in Part I of this paper.

It has to be realized that the issues identified in this paper should not be viewed as questioning the safety of operating reactors, which have attained a very high operational-safety record, but rather as areas where uncertainty in knowledge exists, where the safety assessment has been based on conservative assumptions, and where regulatory decisions are needed, or will need to be confirmed. Further work, including experimental research, may be required to more accurately determine the overall effect of an issue on the safe operation of the facility, and to confirm that station operation is acceptable, as there remain adequate safety margins.

It is important to note that some of the safety issues identified in this paper for CANDU reactors are common to other reactor types as well.

2. Identification of safety issues, and the process for development of the path forward for resolution of issues

2.1. Issue identification

An initial list of issues was developed using the IAEA-TECDOC on Generic Safety Issues for Nuclear Power Plants with Pressurized Heavy Water Reactors and Measures for their Resolution [1]. All design and safety issues listed in this IAEA-TECDOC were considered. In addition, issues were identified through:

- CNSC staff's regulatory oversight of currently operating reactors;
- results of life extension assessments;
- safety issues identified in pre-licensing reviews of new CANDU designs; and
- feedback from CNSC management following their review of the preliminary list of safety issues.

For the purposes of this work, the following definition of an issue was used: "A safety issue is an issue related to the design or analysis of NPPs that has been identified as potentially challenging to safety functions, safety barriers, or both." This definition is in-line with that provided in IAEA-TECDOC-1044 [2] and IAEA-TECDOC-640 [3].

The generic safety issues listed in references 2 and 3 were identified from operational experience or events, deviations from current standards and practices, and potential weakness identified by analysis. It is important to note that throughout the IAEA-TECDOCs on safety issues for light water reactors [2, 3], the focus is on the impact of the issue on safety functions and barriers.

2.2. An approach for initial categorization of safety issues

The first step in the development of the path forward for the resolution of outstanding safety issues is categorizing the issues into three broad categories, as described below.

Category 1: Not an issue in Canada, drop from the list

Issues may be dropped from the list if the issue does not meet the definition of an issue, or if the issue does not exist in Canada.

Category 2: The issue is a concern in Canada — appropriate measures are in place to maintain safety margins.

The licensees have appropriate control measures in place to address the issue and to maintain safety margins. It is important to recognize that different control measures may be implemented for life extension and new reactors. The CNSC continues to monitor licensees' management of the issue.

Category 3: The issue is a concern in Canada — measures are in place to maintain safety margins, but the adequacy of these measures needs to be confirmed.

The licensees have some control measures in place to maintain safety margins, but further experiments and/or analysis are required to improve knowledge and understanding of the issue, and to confirm the adequacy of safety margins. Additional measures may be needed such as design improvements, supplementary administrative and operational controls, additional surveillance and/or inspections. The RIDM process is applied and the path forward for resolution of the issue for operating reactors, life extension of operating reactors and new reactors is developed. The CNSC will continue to monitor licensees' management of the issue to ensure its timely and effective resolution.

Table 1 lists all issues by category while the basic approach used to categorize the issues is illustrated in Figure 1. Multiple flow paths were needed to reflect the fact that, although an issue may be under control for operating reactors, it is prudent to determine whether other actions could be taken to address the issue at the time of life extension, or for new reactors (i.e. undertake design changes rather than addressing the issue through operational or administrative measures). This pertains to Category 2 issues.

2.3. Process for determining the risk significance of safety issues

The risk significance of Category 3 issues was determined using the methodology for risk evaluation and estimation, which is described in Part I of this paper. The methodology employs the CNSC's RDIM process [4], which is based on the CSA Q-850 standard [5]. It allows that risk control measures are recommended based on risk estimation and evaluation; risk management is accomplished by implementing measures and monitoring the impact of those measures. Provisions for transparency, consultation with stakeholders, and consideration of feedback are included.

The RIDM process recognizes that the total risk has two components: the risk due to the existence of the plant in the absence of the safety issue and the incremental risk associated to the safety issue. The Risk Significance Level (RSL) reflects the impact of the safety issue on the risk. Four risk significance levels are employed:

- RSL 1 — a safety issue brings no additional risks or a negligible one;
- RSL 2 — the risk associated with the safety issue increases, but it is still well within the tolerable region;
- RSL 3 — the risk evaluated to be at the border between the tolerable and the unacceptable regions; and
- RSL 4 – to risk evaluated to be not acceptable.

Reference [6] provides more details on the definition of the risk significance levels.

3. Application of the RDIM process to safety issues for pressurized heavy water reactors

The RDIM process has been applied to several Category 3 safety issues. As an example, the results for the assessment of Emergency Core Cooling System (ECCS) sump screen adequacy (SS 1 in [1]) and moderator temperature predictions (Issues AA 8 and SS 8 in [1]) are presented here.

3.1. Adequacy of ECCS sump screen

3.1.1. Issue description

Containment is equipped with sumps to collect the water lost from the primary circuit after a LOCA in order to recirculate the water in the ECC recovery phase of the accident. The sumps are covered with a screen that that is intended to prevent debris penetration to the suction of the ECCS pumps.

The thermal insulation used inside the containment and the total area of the screen above the sump together with dust and debris that occur in containments form a combination that raises safety concerns regarding the possibility of maintaining ECCS circulation after a medium or large LOCA. Operational experience based on events in Sweden and in the USA has demonstrated that even a relatively small amount of similar fibres can effectively block a large portion of the screen area. Sump screens must be designed and installed to ensure that the screening function is maintained.

Table 1. Categorization of issues

Category 1 – not an issue in Canada	**Category 2** – an issue, but appropriate measures are in place to maintain safety margins	**Category 3** – an issue, apply RIDM to develop path for resolution
GL 1 Classification of components	GL 2 Environmental qualification of equipment and structures	GL 3 Ageing of equipment and structures
GL 5 Need for performance of plant-specific probabilistic safety assessments (PSA)	RC 1 Inadvertent dilution or precipitation of poison under low power and shutdown conditions	GL 4 Inadequacy of reliability data
RC 2 Fuel cladding corrosion and fretting	CI 3 SG tube integrity	CI 1 Fuel channel integrity and effect on core internals
SS 2 Potential problems in ECCS switchover to recirculation	CI 4 Loads not specified in the original design	CI 2 Deterioration of core internals
SS 4 Leakage from systems penetrating containment or confinement during an accident	CI 5 Steam and feedwater piping degradation	SS 1 ECCS sump screen adequacy
SS 7 Assurance of ultimate heat sink	PC1 Overpressure protection of the primary circuit and connected systems	SS 5 Hydrogen control measures during accidents
ES 1 Reliability of off-site power supply	PC 2 Safety valve and relief valve reliability	SS 8 Availability of the moderator as a heat sink (AA7, AA8 addressed through SS 8)
ES 5 Reliability of instrument air systems	PC 3 Water hammer in feedwater and steam lines	IH 6 Need for systematic assessment of high energy line break effects
ES 6 Solenoid valve reliability	SS 3 Severe core damage accident management measures	AA 3 Computer code and plant model validation
IC 1 Inadequate electrical isolation of safety from non-safety-related equipment	SS 6 Reliability of motor-operated and check valves	AA 9 Analysis for void reactivity coefficient
IC 2 I&C component reliability	ES 2 Diesel generator reliability	PSA 3 Open design of the balance of plant — steam protection
IC 3 Lack of on-line testability of protection systems	ES 3 Reliability of emergency DC supplies	PF 9 Fuel behaviour in high temperature transients

Category 1 – not an issue in Canada	**Category 2** – an issue, but appropriate measures are in place to maintain safety margins	**Category 3** – an issue, apply RIDM to develop path for resolution
IC 4 Reliability and safety basis for digital I&C conversions	ES 4 Control room habitability	PF 10 Fuel behaviour in power pulse transients
IC 5 Reliable ventilation of control room cabinets	IC 7 Availability and adequacy of accident monitoring instrumentation	PF 12 GAI 00G01 Channel voiding during a Large LOCA
IC 6 Need for a safety parameter display system	IC 9 Establishment and surveillance of setpoints in instrumentation	PF 14 Positive reactivity feedback
IC 8 Water chemistry control and monitoring equipment (primary and secondary)	CS 1 Containment integrity	PF 15 GAI 95G01: Molten fuel/moderator interaction
IH 5 Need for systematic internal flooding assessment including backflow through floor drains	IH 1 Need for systematic fire hazards assessment	PF 18 Fuel bundle/element behaviour under post dryout conditions
IH 7 Need for assessment of dropping heavy loads	IH 2 Adequacy of fire prevention and fire barriers	PF 19 Impact of ageing on safe plant Operation
IH 8 Need for assessment of turbine missile hazard	IH 3 Adequacy of fire detection and extinguishing	PF 20 Analysis methodology for NOP/ROP trips
AA 2 Adequacy of plant data used in accident analyses	IH 4 Adequacy of the mitigation of the secondary effects of fire and fire protection systems on plant safety	
AA 4 Need for analysis of accidents under low power and shutdown conditions	EH 1 Need for systematic assessment of seismic effects (EH 2, Need for assessment of seismic interaction of structures or equipment on safety functions, is a sub-set)	
AA 6 Need for analysis of total loss of AC power	EH 3 Need for assessment of plant-specific natural external conditions	
MA 5 Degraded and non-conforming conditions and operability determinations	EH 4 Need for assessment of plant-specific man induced external events	

Category 1 – not an issue in Canada	**Category 2** – an issue, but appropriate measures are in place to maintain safety margins	**Category 3** – an issue, apply RIDM to develop path for resolution
OP 1 Operating experience feedback	AA 1 Adequacy of scope and methodology of design basis accident analysis	
	AA 5 Need for severe accident analysis	
	MA 13 Availability of R&D, technical and analysis capabilities for each NPP	
	PSA 2 Equipment qualification	
	PSA 4 PHT relief	

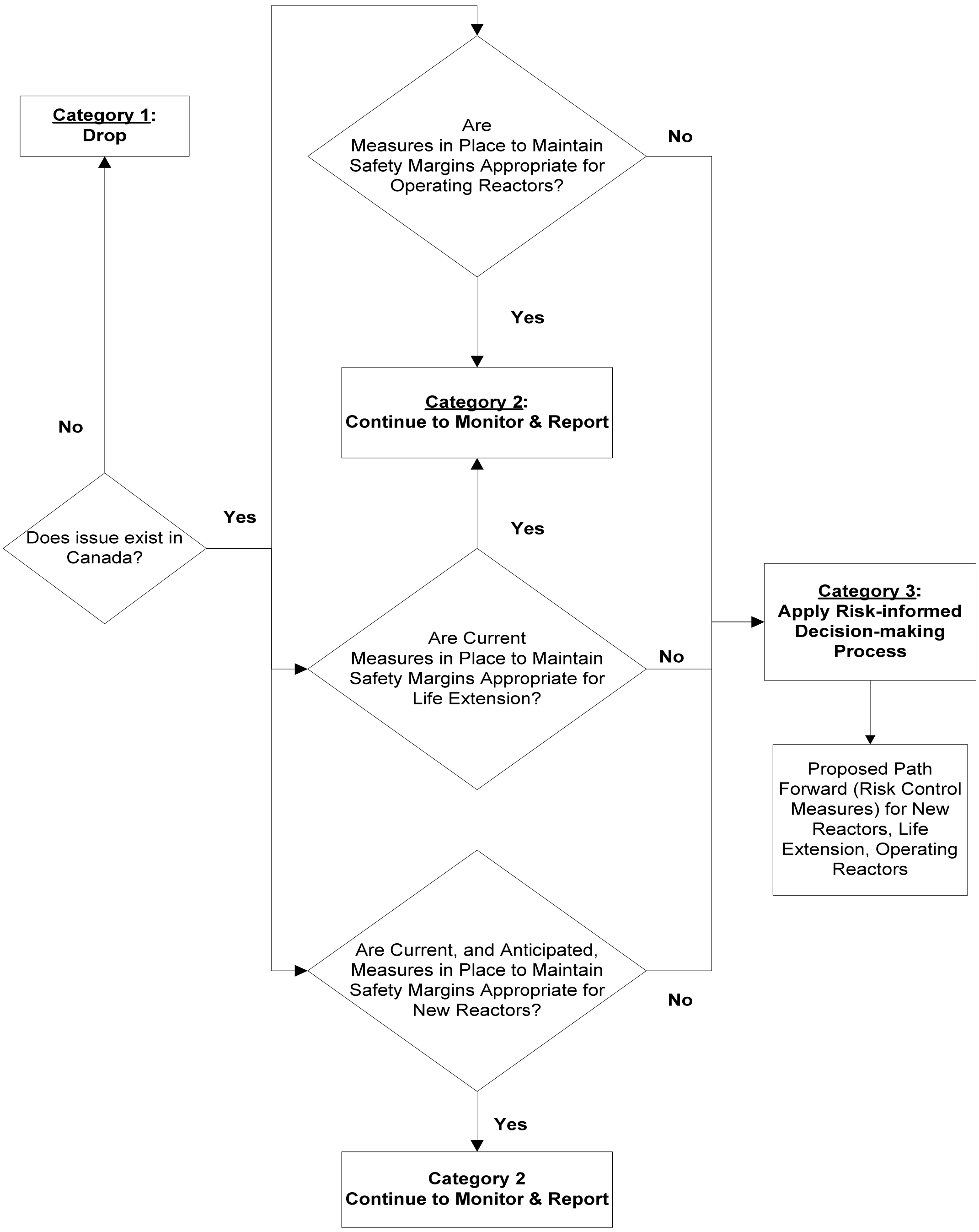

FIG. 1. Process for categorization of outstanding safety issues pertaining to operating reactors, life extension and new reactors.

Break-up of thermal insulation around equipment and pipes inside the containment can, under LOCA conditions, lead to an impairment of ECCS recirculation by clogging the sump screens and/or the ECCS heat exchangers. The ECCS function can also be affected by inadequately screened debris.

A postulated Loss of Coolant Accident (LOCA) would dislodge significant quantities of insulation material, both fibrous and particulate. Much of this debris is expected to be transported to the reactor building sump with the coolant lost from the reactor through the break. ECCS recirculation recovers water from the sump, cools it and returns it to the reactor to cool the core. The ECCS strainers are located in the sump and protect the ECCS recirculation flow path by preventing the debris from entering the ECC system. As a result, a layer builds up over the strainer surface. The strainers should be designed with sufficient surface area that the debris bed does not impede flow.

In addition, preliminary research findings of the Integrated Chemical Effects Test (ICET) programme in the United States have raised concerns about the formation of deposits on ECCS strainers. The ICET programme assessed the impact of reactor building sump chemistry following a LOCA and possible implications for ECCS strainers during recirculation following a LOCA. In some of the ICET tests a gelatinous deposit was discovered on the fibre samples in the tank. There is a concern that such chemical deposits could lead to a partial blockage of the strainer thereby impairing the ECCS recirculation.

3.1.2. Status of issue in Canada

Upon learning of the incident of ECCS strainer blockage at Barseback, Sweden, the CNSC took the following measures:

- A comprehensive study was done in 1995-1996 and concluded that licensees needed to evaluate properly the quantity and characteristics of the debris that could be generated, that fine as well as large pieces should be considered, that existing strainers in some stations were inadequately sized, and that strainers may be susceptible to significant mechanical loads due to pressure differentials; and

- Licensees were asked to consider design changes if necessary.

The licensees undertook the following actions:

- A comprehensive programme was carried out to evaluate debris generation, transport and accumulation;

- An experimental programme was initiated under the CANDU Owners Group (COG) to study the pressure drop characteristics, the type of insulation, the effect of particulates, and the long-term behaviour of the debris bed;

- AECL developed a fine type strainer to provide a larger surface area; and

- Methods and guidelines have been developed for assessing ECCS sump strainers for individual NPPs to fulfill the requirements of:

    - The maximum allowable pressure drop across the strainer at the expected flow rate and temperature;
    - Assessing the debris type, flow path assessment, water hold-up and quantities of debris transported; and
    - Larger replacement strainers are installed at Darlington, Pickering A & B, Point Lepreau and Gentilly-2. Old strainers have been enlarged at Bruce A and old strainers have been determined to be sufficient at Bruce B.

The issue as described in the Generic Safety Issues document [1] has been closed. However, as discussed above, a related issue has been identified in US research into chemical effects in sump water– the ICET tests. CNSC raised Generic Action Item 06G01 ‘ECC Strainer Deposits’ to address this concern.

Industry was advised of CNSC staff concerns and immediately established a CANDU Owners Group research programme to address it. CNSC staff raised GAI 06G01 to track the issue. It has the following closure criteria.

1) Licensees are to evaluate the ICET tests and demonstrate that CANDU ECCS strainers are not vulnerable to deposits such as those identified in the ICET tests.
2) If closure criterion 1 cannot be achieved or if additional supporting information is needed, licensees are to perform appropriate research to identify what deposits may form in CANDU reactors and show their effects on ECCS performance are acceptable.
3) If closure criterion 2 cannot be achieved, licensees are to propose appropriate mitigating measures to ensure that ECCS remains effective, in the presence of debris and any deposits that may form in the sump environment.
4) Licensees are to identify the physical phenomena that are important to ECCS recirculation and use this information to demonstrate that existing designs are adequate.

Licensees have submitted information giving confidence that the chemical environment in CANDU reactors does not include the features that led to possibly harmful deposits in the ICET tests. In particular, the study showed that addition of tri-sodium phosphate (TSP) to the water in the ICET tests led to accelerated aluminium corrosion and the formation of the deposits. CANDU reactors do not make use of TSP to raise sump pH after a LOCA. CNSC staff accepted the conclusions of the study and agreed that Closure Criterion 1 has been met for all licensees.

However, licensees could not completely exclude chemical effects under CANDU sump conditions. Therefore an experimental programme was established to close this gap in knowledge. CNSC has been consulted on the test plan and methods and staff's views have been taken into account. The programme to address closure criterion 2 was scheduled to be completed in 2007. Early results are encouraging.

Progress by industry in addressing this generic action item has been excellent. The research programme was established quickly and the work is proceeding on schedule.

3.1.3. Risk control measures

The risk assessment for this safety issues is provided in Annex 1. Recommended risk control measures are as follows:

- Operating Reactors: Address GAI 06G01 closure criteria, which include performing the planned chemical effects tests to improve knowledge understanding of the potential chemical effects.

- Life Extension: Address GAI 06G01 closure criteria, which includes performing the planned chemical effects tests to improve knowledge understanding of the potential chemical effects. Consider implementing practicable design changes

- New Build: It is expected that the GAI will be closed via improved design.

3.1.4. Risk assessment summary

This safety issue has the potential to severely impair a special safety system (ECCS recirculation) when the system is needed (LOCA). While the frequency of LOCA will not be changed, the off-site doses to the public can be significantly higher than those estimated in the Safety Report. Licensees have demonstrated that serious chemical effects that have been identified for other reactor designs do not occur with CANDU reactors. However, at this time, the possibility of other chemical effects specific to CANDU has not been eliminated; therefore, there is uncertainty in assessing the likelihood of this impairment. To address this risk, we recommend that licensees complete the current R&D programme. When this R&D programme is completed, we expect to have sufficient information to estimate the risks to the public with a higher confidence and recommend design changes, if needed.

On this basis, the overall Risk Significance Level was judged to be a ‘3’, although it could be 2 for plants with large area strainers.

3.2. Moderator temperature predictions

3.2.1. Issue description

During some loss-of-coolant accidents, fuel channel integrity depends on the capability of the moderator to be the ‘ultimate heat sink’. Fuel channel integrity is assured if the calandria tubes do not fail after contact with the pressure tube. In turn the calandria tube temperature depends on the local moderator subcooling. Analysis is performed to show that there is no prolonged film boiling on the outside of the calandria tube. Such calculations depend on a number of computer codes, which must be validated.

An unreliable ultimate heat sink constitutes a threat to fuel channel, and hence fuel integrity under accident conditions. Validated codes are therefore required to provide confidence in the predictions of fuel channel integrity in accidents where the moderator acts as a heat sink.

3.2.2. Status of issue in Canada

CNSC staff considers that moderator temperature predictions have not been validated adequately, given the tight safety margins that exist currently. Licensees were therefore required to perform three-dimensional experiments to validate the moderator temperature predictions. These experiments take place in a test facility at AECL’s Chalk River Laboratories. They have been completed for the CANDU 9 configuration and the results are in agreement with the predictions of the computer codes. Tests for other designs are underway.

CNSC staff raised generic Action Item 95G05 ‘Moderator Temperature Predictions’ in 1995 to address this issue. CNSC staff also identified a potential scenario whereby a pressure tube failure could lead to loss of moderator. A summary of the CNSC position statements addressing those issues is given below.

In some Large LOCA events, the integrity of fuel channels depends on the capability of the moderator to act as the ultimate heat sink. As fuel channels heat up, pressure tubes balloon and come into contact with the calandria tubes. Fuel channels remain intact upon contact if the moderator fluid outside the calandria tubes is cold enough to provide good heat removal capability. Channels may fail, however, if the moderator temperature is too high to prevent the outside of the calandria tubes from drying out following contact on the inside with the pressure tubes.

In view of the severe consequences of channel failures, and the small safety margins that currently exist with respect to moderator temperature (or moderator subcooling) requirements, CNSC staff has requested the validation of the computer code used to calculate the moderator temperature distribution against three-dimensional (3D) experimental data representative of reactor conditions. A 3D test was completed in 2001 to the satisfaction of CNSC staff, and the validation of the computer code MODTURC-CLAS was performed against both separate-effect testing and the results of the 3D integral test. This work was carried out by an industry team representing all Canadian utilities. The team made a final submission on code validation to the CNSC in December 2005 with a request to closure this GAI.

CNSC staff has developed a plan to review the industry submission in detail, and to identify factors that would lead to acceptance or rejection of the request for GAI closure. The review has started in 2006 and was scheduled to continue to the end of 2007 in view of the large size of the submission that includes seventeen individual assessment reports.

3.2.3. Risk control measures

The risk assessment for this safety issues is provided in Annex 2.

For operating reactors, life extension and new build, licensees are expected to address the GAI closure criteria. However, given that licensees have made a final submission to the CNSC in December 2005 with a request to closure this GAI, the CNSC should address this closure request in a timely manner (As noted above, CNSC staff has developed and implemented a plan to address the matter.)

It is noted closure of the GAI is being considered; however, station-specific action items related to this issue will be raised. Licensees are expected to address the station-specific issues.

3.2.4. Risk assessment summary

Inadequate moderator sub-cooling has the potential to lead to a severe accident by failing more than one channel following a LOCA. For these DBA events, the moderator is credited as a heat sink following pressure tube ballooning into contact with the calandria tube for some of the high-power channels. Channel failure depends on the available local moderator subcooling which in turn depends on local moderator temperature.

Uncertainties in moderator temperature prediction raise concerns that the moderator may not able to ensure removal of decay heat; the fuel-cooling function may be impaired causing additional fuel failures and consequential failure of fuel channels leading to severe core damage. Hence risks of increased public doses and of increased frequency of severe accidents.

To address these risks the licensees need to address the closure criteria of GAI 95G05, to increase the accuracy of the computer codes assessing moderator subcooling, and thus to increase the confidence in the adequacy of the moderator as a heat sink in a LOCA.

The overall Risk Significance Level was judged to be a ‘3’ for this safety issue.

4. Conclusions

The RIDM process has been applied to outstanding CANDU safety issues. The risk associated with each safety issue was identified, estimated and evaluated to recommend measures to control these risks.

The application of the process has led to the development of risk-informed regulatory positions, including the path forward for resolution of each safety issue in view of currently operating reactors, life extension of existing reactors, and new reactors. It provides the decision makers with information on the risk environment and recommends risk control measures. These insights are of particular importance to facilitate the development of plant-specific safety improvement programmes or the reviews of new reactor designs.

ACKNOWLEDGMENTS

The authors wish to acknowledge the contribution of staff of the Directorate of Assessment and Analysis and the Directorate of Power Reactor Regulation who took part in the work leading to the identification of the outstanding CANDU safety issues, and the assessment of the risk associated with each safety issue described in this paper.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, "Generic Safety Issues for Nuclear Power Plants with Pressurized Heavy Water Reactors and Measures for their Resolution", IAEA-TECDOC–1554, June 2007.

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, "Generic Safety Issues for Nuclear Power Plants with Light Water Reactors and Measures Taken for their Resolution", IAEA-TECDOC-1044, 1998.

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, "Ranking of Safety Issues for WWER-440 Model 230 Nuclear Power Plants, Report of the IAEA Extrabudgetary Programme on the Safety of WWER-440 Model 230 Nuclear Power Plants", IAEA-TECDOC-640, February 1992.

[4] "Risk-Informing CNSC Planning, Licensing and Compliance Activities", Detailed Guidance Document, Revision 3, February, 2006.

[5] "Risk Management — Guidelines for Decision Makers", CAN/CSA Q-850-97, A National Standard of Canada. 27.

[6] BUJOR A., RZENTKOWSKI G., MILLER A., "Development of Risk-Informed regulatory Position on CANDU Safety Issues. Part I: Methodology Development for Risk Estimation and Evaluation", IAEA Topical Meeting on Advanced Safety Assessment Methods for Nuclear Reactors, Daejon, Republic of Korea, October 30–November 2, 2007.

Annex 1. Application of the risk-informed decision-making process to ECC strainer effectiveness

Risk Area	Scenario	Likelihood		Consequences		Risk Significance Level
		Category	*Comments*	*Category*	*Comments*	*Category*
Impairment of Special Safety System/Safety Function	Strainer partially blocked due to fouling/chemical effects and subsequent pump cavitation results in recirculation pumps not being able to inject water into core to cool the fuel	Medium	There are still uncertainties with regards to the effect of chemical effects on strainer fouling and subsequent blockage Based on current knowledge of chemical effects , the likelihood for impairment due to chemical effects is estimated to be 25 to 50% (the range in strainer area is quite large amongst the Canadian plants) The results of testing may reduce uncertainties in estimating the likelihood of strainer blockage	Medium	Defence in depth is degraded and the safety function of cooling is impaired	3

Risk Area	**Scenario**	**Likelihood**		**Consequences**		**Risk Significance Level**
		Category	*Comments*	*Category*	*Comments*	*Category*
Increased Doses to Public	LOCA + consequential loss of recirculation	L2	Frequency of LOCA (10^{-2} (small LOCA) to 10^{-4} (Large LOCA)) as a DBA is not significantly changed by a ~ 25% to 50% likelihood of ECC impairment with loss of long-term recirculation	C2	Consequential loss of ECC recirculation will lead to additional fuel failures For Darlington, public doses for this scenario is less than dose limits for LOCA (LOCA dose is well below the C6R0 (0.3% of the Class 3 (LLOCA)) dose limit), but with loss of ECC recirculation, the dose is ~11mSv (~40% of the Class 3 dose limit of 30).	2

Risk Area	**Scenario**	**Likelihood**		**Consequences**		**Risk Significance Level**
		Category	*Comments*	*Category*	*Comments*	*Category*
	LOCA + Consequential Loss of recirculation + Failure of containment	L2	Frequency of LOCA + Loss of Containment as a DBA is not changed significantly if there is strainer blockage	C3	For Darlington, the LOCA with consequential loss of recirculation and with containment failure could be as much as 12 times the Class 5 C6R0 dose limit. However, this estimate is based on early impairment of ECC. If the impairment is later, the consequences will be less. The severity depends on the degree of strainer fouling, and the time at which ECC begins to be impaired.	3

Risk Area	Scenario	Risk Significance Level	Comments
Severe Accidents Risks	Strainer blockage leads to increased probability of failure of ECC recirculation in comparison with that determined in the PSA. Therefore, we expect that CDF and other Safety Goals will be greater than previously determined in the PSA.	3, but could be 2	The ranking will depend on the risk significance of ECC recirculation for specific plants

Annex 2. Application of the risk-informed decision-making process to moderator temperature predictions

Risk Area	Scenario	Likelihood		Consequences		Risk Significance Level
		Category	*Comments*	*Category*	*Comments*	*Category*
Impairment of Safety Function – Fuel Channel and Fuel Cooling	Confidence in prediction of moderator sub-cooling and of adequacy of sub-cooing to ensure integrity of the fuel channel	High	We are confident that there are analysis issues (number and importance of issues; refer to the Issue Description and GAI position statement) As a result we don't know how well the code predicts	Low	Inadequate confidence in safety analysis results due to analysis issues	3*
	Failure of several Fuel Channels due to Inadequate Cooling under a LOCA, or LOCA + LOECC	Medium	As we don't know how well the code predicts, we consider that there is a 50% chance that the moderator is not effective as credited	Medium	Cooling safety function impaired and barriers impaired (both additional fuel failures and fuel channel failures resulting in release of tritium from moderator to containment)	3
Increased Doses to Public	Consequential CT/PT Failure because of inadequate sub-cooling during a LOCA, or LOCA	L2	Inadequate sub-cooling has no impact on LOCA frequency This is where the	C3	If there is inadequate sub-cooling, may get PT/CT failure, then additional releases due to fuel failures and due to release of tritium from moderator to	3

Risk Area	Scenario	Likelihood		Consequences		Risk Significance Level
		Category	*Comments*	*Category*	*Comments*	*Category*
	+ LOECC		uncertainty in the adequacy of the code vis-à-vis predictions comes into consideration, if the code significantly underpredicts sub-cooling , then greater likelihood of entering SA regime		containment We do not have calculations for public doses as containment performance needs to be considered; we assume that they will be greater than for a LOCA, or a LOCA+LOECC	

* this rating may change following the completion of the review of licensees submissions

Risk Area	Scenario	Risk Significance Level	Comments
Severe Accidents Risks	Large LOCA where there is a need to rely on moderator for fuel cooling	3	If moderator fails to do its function, reactor enters SA regime. There are uncertainties regarding the number of channel failures, and progression to severe core damage

SESSION II. REDUCTION IN OCCUPATIONAL DOSE EXPOSURE

PICKERING B NUCLEAR DOSE REDUCTION THROUGH INNOVATIVE SHIELDING AND MOCK-UP TRAINING

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Abstract

An innovative temporary shielding tool design has made a significant contribution to the success of the P761 Single Fuel Channel Replacement (SFCR). Significantly reducing collective radiation dose and project expenses, the tool (referred to as REACTORshield) will lower production costs and has proven to be an effective tool that can be implemented industry-wide in the future. Aligning around the nuclear values of teamwork, respect, integrity and commitment, the team responsible for this success overcame administrative obstacles and time constraints to get the REACTORshield pieces in place before the first project evolution. Prior to unit shutdown, the team worked diligently to source the right product provider and ensure just-in-time delivery of the tool. Contributing to this phase of the success story was the solid support of line management, who worked side by side with the team. As the project gained momentum, technical and management staff from across Ontario Power Generation (OPG) assisted in the design, development and procurement processes. Compared to previous campaigns at various CANDU stations, the collective dose rates received during the Pickering B SFCR were well-below industry norms. Previous campaigns at both Darlington and the Bruce site resulted in 39 and 38 person-rem (0.39 and 0.38 p-Sv), respectively. The collective dose for this project was 26.5 person-rem (0.265 p-Sv) versus a target of 31.3 person-rem (0.313 p-Sv). This translates approximately into a five person-rem (0.05 p-Sv) and $200 000 cost savings. The capital expense for acquiring these REACTORshield shielding tools was $150 000. The gain is obvious and will be significant when future reactor face projects are taken into consideration. For instance, this tool was further utilized for the Feeder Replacement project during the Pickering A P711 outage. In summary, the tool has consistently demonstrated a reduction of general gamma fields by 40 to 60 per cent.

1. Reactor face dose rate challenge

Completing a single fuel channel replacement (SFCR) during outage P761 was not only a part of the outage plan, but also a mandatory requirement of the CNSC.

The duration of this work (the maintenance window) at the face was estimated to be more than 1000 person-hours. The average working distance dose rate at reactor face was predicted to be 150 mrem/h to 200 mrem/h,[1] therefore the collective dose was targeted at 31.3 rem. Outage management concluded if dose rate was not lowered; the target collective dose could easily be exceeded.

Operating experience (OPEX) from both Darlington and Bruce Power was reviewed. One of the major findings stemming from the OPEX was the necessity of a temporary shielding cabinet tool to reduce working area dose rates.

With imminent outage commitments and milestones taking priority, there was lack of person-power within the outage department to provide oversight of the shielding cabinet project. In the truest sense of teamwork, ALARA (As Low As Reasonably Acceptable) and Field Section Managers stepped up to provide a dedicated team to review project work plans and procedures.

In short time, a project to develop a shielding cabinet was incorporated in the plan as a means to reduce the worker dose. Early into the planning stage it was clear this project was not progressing in a timely fashion. Additionally, the price tag of this cabinet was estimated more than $3 million. In the end, a prudent decision was made to investigate an alternate shielding solution.

2. The employment of radiation protection measures

2.1. ALARA oversight

With a commitment of collective dose of 31.3 person-rem, the ALARA Oversight team continued to address the shielding issue. The Pickering B Shielding SPOC was contacted to provide insight on available shielding tools for the SFCR. There were limited numbers of options.

[1] 100 rem = 1 Sv.

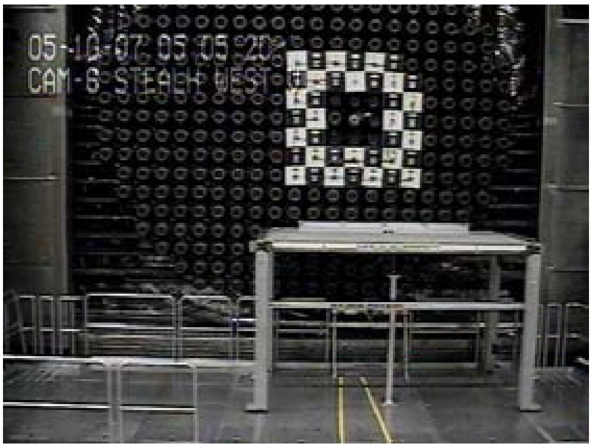


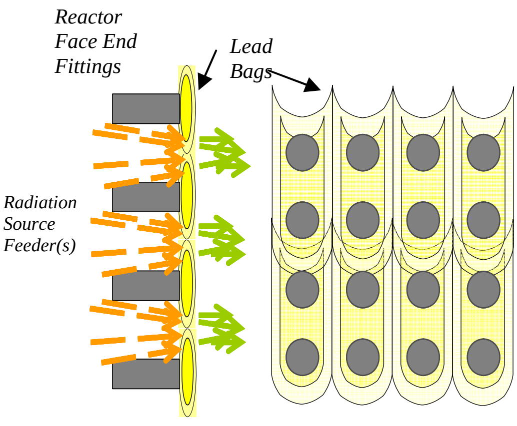
FIG. 1. REACTORshield tool applied.

FIG. 2. REACTORshield pieces.

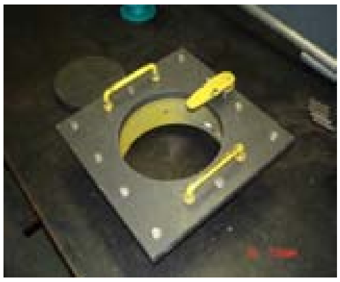
2.2. Innovative shielding

Lead blankets have been the traditional method for temporarily shielding hot spots and reactor face. Due to the tight dose target, use of the blankets proved to be an inadequate way to reduce the gamma fields. The SFCR campaign needed a new shielding solution that would eliminate the inherent problem with the traditional method.

Although it was not yet at a usable iteration, the existing design from Darlington (which had been field tested) was considered. In principle it would have been performed in conjunction with a radiation attenuator. However, it was hypothesized the hard outer surfaces of the shield cap could potentially damage the reactor face components if the shield caps were to fall. After a careful review and discussion within the ALARA team, a new and improved design of the shielding tool was developed.



Front View showing traditional lead bag's intrinsic deficiency – 85% of source remains unshielded.



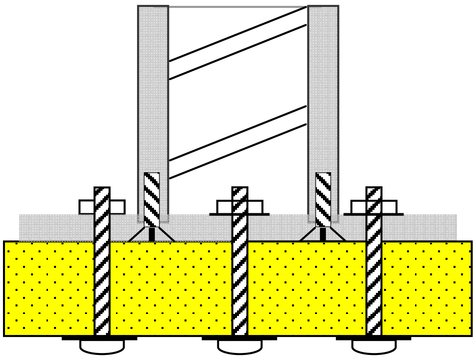
Darlington NGS Reactor Face shielding innovation required further development for use at Pickering NGS.

2.3. Timeline — development and Implementation

On June 6, 2007 (13 weeks prior to the outage), the ALARA shielding Single Point of Contact (SPOC) recognized a very stringent time line for the process of design, prototyping and delivery. The imminent threat to the project was the readiness of the shielding – *would there be enough time to have all pieces made?* Adhering to set OPG procedures, the design, material selection and the determination of a suitable approved supplier had to be finalized before the outage start. Meeting all of these requirements was quite a challenge. Thus, it was critical that all stakeholders were conscious of the time constraints and were prepared to fully support the project.

With problems defined, theoretical design ideas were brainstormed on a whiteboard and a final design was developed. The Shielding SPOC drafted a new sketch and managed to find a facilitator to test the prototypes in order to develop the final version. As always, time was very much a constraint.

First Prototype Sketch

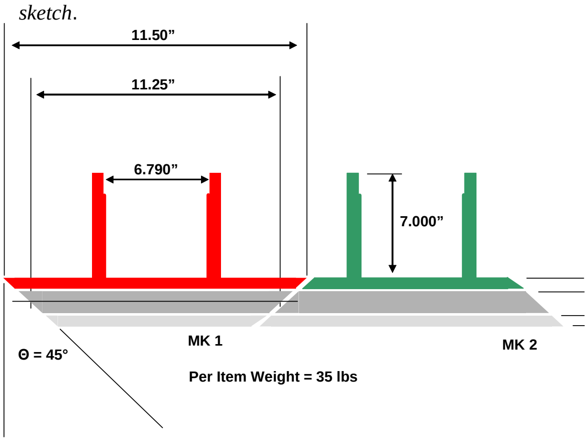


First Prototype

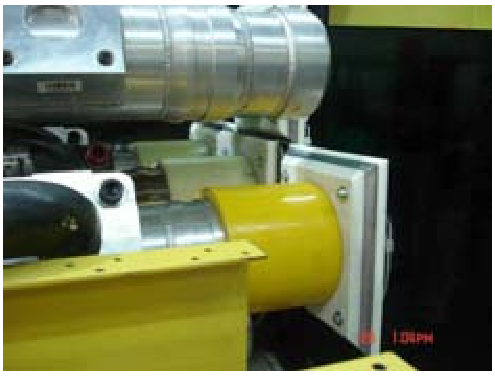
At this point, there were only 60 days left before the outage.

After going through many inquiries, the ALARA Shielding SPOC finally found a flexible, competent facilitator that could supply the required materials and manufacture the first prototype. Once the first prototype was reviewed, it was determined a major change was required to ensure that it would fit on the reactor face. Less than six days later the second prototype was built with all major changes incorporated. It was named REACTORShield.

Second prototype sketch.



Second Prototype



2.4. *Leadership role of radiation protection and IMS management*

After a successful REACTORShield demonstration at the TMB, the IMS Project manager, Radiation Protection Manager, Pickering B Shielding SPOC and other FLMs involved in the project agreed that this new design would work for the SFCR project.

Time was of the essence. Unanticipated hurdles related to the procurement process and funding occurred during this time constraint. Fortunately, the Radiation Protection Manager and IMS Project Manager stepped up, working together to expedite the procurement in a timely fashion. This exhibited strong leadership and commitment within our management team.

Though more than 100 components of the shielding cabinet had to be fabricated, the REACTORShield unit arrived on time.

2.5. Mock-up training — team building

A comprehensive training programme continued over several weeks. From early July to early October of 2007, more than 70 staff members branched into multiple skilled trades' teams. Pickering B Unit 6 P761 Outage SFCR trade teams included: Radiation Protection Crew (12), Pressure Tube Rolling Crew (14), Welding Crew (14), Flasking Crew (16) and a Utilities Crew (16).

With team members rehearsing on a pseudo-reactor face, opportunities for best-practice discovery were immediately available to all team members. The pre-execution learning positioned the P761 SFCR for a commendably lower person-rem dose by campaign completion.

Training Mock-Up Building

Bridge Rehearsals for P/T Swab Evolution

Training Mock-Up Building

Radiation Protection Team Photo

3. Success and results

On Sept. 27, 2007, new temporary shielding pieces finally arrived at Pickering and were quickly installed on the end fittings of the east and west reactor (see below).

Table 1 summarizes the reduction in dose rates, dose expenditure for installing these REACTORshield tool, and associated dollar values.

4. Empowered future performance

The reactor face shielding developed during the P761 SFCR and radiation protection team building initiatives has the potential to provide future savings on all CANDU reactor projects. The shielding provides significant reduction in gamma fields from feeder pipes on a reactor face. Temporary shielding can be used for a variety of projects involving fuel channel work including CIGAR, UDM, realignment, feeder replacement, re-tube and refurbishment. Dose savings for future work can be considered as hundreds of rem or in the millions of dollars.

Table 1. P761 SFCR — reactor face dose rate (Before and after temporary shielding installed)

Dose Rate Reduction	
Working Area Rx Face Dose Rates *(unshielded)*	***140 mrem/h***
Working Area Rx Face Dose Rates *(shielded)*	***40 mrem/h***
Installation Dose	
Traditional Shielding	***250 mrem***
REACTORShield Install Dose	***100 mrem***
Project Dose	
SFCR — Project Dose Target	***31.3 rem***
Project Actual	***26.5 rem***
Dose Saved	***4.8 rem***
Dollar Value Equivalent	
Temporary Shielding Tool Cost	***$147 000***
Dollar Value per person-rem	***$40 000***
Actual Dose Savings/Person-rem value ($)	***4.8 rem/$180 000***
Empowered Future Work — Person-Rem Value	***> $1 000 000***

BEST PRACTICES IN MANAGEMENT OF HEAVY WATER AND TRITIUM

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Abstract

The heavy water inventory of a typical HWR constitutes about 12% of the capital cost of the HWR. The typical tritium production in a single unit HWR is about 2×10^6 Ci/y.[1] Heavy water and tritium control are important aspects of HWR operation, and this involves people, procedures, equipment and heavy water and tritium separation systems. Station personnel are trained to understand the importance of heavy water management and the economics and environmental impact of tritiated heavy water losses. The tritium and heavy water losses from a HWR are both airborne and waterborne in nature. Tritium is of particular concern in the HWR industry given the nature of heavy water reactors to build up high levels of tritium over time. Recent increased interest from regulators and the public has led more HWR utilities to pay increasing attention to occupational safety and environmental emissions of tritium at their power stations. As competing reactor technologies improve, a simple and economic means for tritium removal from heavy water in HWRs is essential for the long-term attractiveness of HWR technology. Tritium safety, occupational and environmental issues are of central importance in HWR licensing and operation. Building upon GE's extensive operational experience in tritium management in HWR reactors and its own tritium handling facility, GE[2] has developed a large-scale diffusion-based isotope separation process as an alternative to conventional cryogenic distillation. Having a tritium inventory an order of magnitude lower than conventional cryogenic distillation, this process is very attractive for heavy water detritiation, applicable to single and multi-unit HWR and research reactors. Additionally, the new process has significant benefits to an operating HWR utility such as reducing environmental emissions and significantly lowering reactor vault tritium MPC(a) levels to a point where station capacity factors can be improved by shorter outages – representing best practice.

1. Introduction

Tritium builds up in a HWR moderator and heat transport systems due to neutron capture by deuterium. This is a characteristic of heavy water reactors, and one of the major safety concerns at HWR stations.

This paper outlines suggested best practices adopted and developed further by GE in its design and delivery for heavy water and tritium processing facilities. GE has developed extensive experience based upon its own tritium handling facilities, the delivery of the Tritium Waste Treatment & Enrichment Facility[1] and long-term experience in system design and reactor service work for HWR power stations.

2. GE's experience

In April 1955, Canadian General Electric created the Civilian Atomic Power Department at its Peterborough, Ontario facility to begin work on design of the prototype nuclear plant: the NPD at Rolphton. GE was the prime contractor for the design, supply, installation and start-up of the Nuclear Power Demonstration Station (NPD) at Rolphton, Ontario. In 1965, GE signed a contract for the turnkey supply of the Pakistan Atomic Energy Commission's 137 MW Kanupp HWR. GE followed the success of the CANDU design with participation in the subsequent design and building of Pickering, Bruce A and B, Darlington, Gentilly, Point Lepreau, Embalse, Wolsong and Cernavoda.

In 1943 a research facility was established in Amersham, UK, later becoming a national centre for radiochemical research under the UK Atomic Energy Authority. After the Second World War, the centre began producing radiochemical labelled compounds with tritium and carbon-14. Today, as GE Healthcare, GE is the world's largest supplier of radiolabelled molecules, supplying around one quarter of all labelled drugs used by the pharmaceutical industry. Through the successful long-term operation of these facilities, GE has gained a great deal of knowledge in the handling of highly tritiated compounds including pure tritiated water. Building on this experience, GE recently designed

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1 1 Ci = 3.7×10^{10} Bq.
2 In this paper GE refers to both GE Healthcare and GE Hitachi Nuclear Energy Canada Inc.

and installed award winning technology for the separation and recovery of tritium at its radiochemical manufacturing site in Cardiff, UK.

The establishment of the Light Isotope Technology Centre of Excellence at GE-Hitachi Canada's headquarters in Peterborough, Ontario has centralised GE's knowledge base of heavy water and tritium handling.

Today, GE's core products and services to the HWR fleet include fuel, fuel-handling machines, inspection and maintenance tooling, computer control systems, reactor field services, heavy water systems, heavy water upgraders and tritium removal systems.

This paper provides GE's assessment of existing HWR operation with respect to management of heavy water and tritium and its impact on operational performance, safety, and environmental discharges.

3. Impact of tritium for an operational PHWR

A CANDU 600 type reactor nominally produces 2×10^6 Ci/y of tritium, with approximately 95% of the tritium formed within the moderator heavy water (D_2O). Tritium in heavy water contributes 30-50% of the annual radiation dose received by operation personnel and represents up to 20% of the radioactivity released from the reactor to the environment[2]. Tritium discharges to the environment from current HWR power stations are up to 20 times higher than from light water reactors[3]. Therefore management of tritium is an important and unique aspect of operating heavy water reactors.

4. Lower vault tritium concentrations

HWR utilities are increasingly recognizing the importance of reducing reactor vault tritium airborne concentrations. Operating with a tritium concentration above the current adopted threshold of 100 MPC(a) [Maximum Permissible Concentration airborne][1] means that outage activities within the vault are conducted within cumbersome air-suits. Working within air-suits increases the time taken to perform a single outage activity by two to three times in comparison to working with respirators. Reducing tritium airborne levels within the vault below the 100 MPC(a) threshold would have a two-fold improvement for an operating company, firstly reducing occupational dose during an outage and secondly reducing the overall duration of a planned outage; this would directly improve the station capacity factor and revenue.

Whilst tritium concentrations grow more rapidly and to higher levels in the moderator circuit of the HWR, it is the tritium in the PHT system that typically leads to chronic airborne concentrations within the vault during an outage. The PHT system is a pressurized high temperature system, that is more prone to small leaks during operation and during outages small leaks occur continually from the pressure tube end fittings.

Current practices in HWR stations involve the use of reactor vault vapour recovery dryers. These systems use adsorption technology to reduce the dew point of the atmosphere within the vault. These dryers collect both tritiated D_2O vapour and light water vapour present in the vault. The collected downgraded D_2O is sent back to the upgraders following regeneration of the dryer beds. This approach has performance limitations as well as practical and economic limitations. Also, over time the adsorbent becomes degraded, reducing its performance. Replacement of the adsorbent is possible but leads to generation of radioactive waste that needs to be appropriately handled for disposal/storage.

A more effective alternative, as adopted by some HWR utilities, is to reduce the tritium source term in the heat transport system. Lower levels of tritium within the heat transport system directly reduce the tritium vault concentrations.

GE has modelled the relationship between tritium vault concentrations, dryer performance and tritium heat transport concentrations.

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[1] 1 MPC(a) = 10 μCi/m^3 HTO.

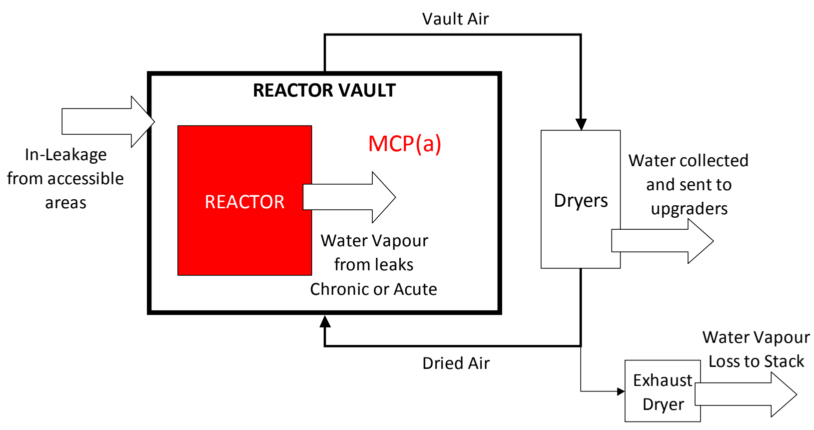

FIG. 1. GE's vault MPC(a) model.

As Figure 1 illustrates, this model accounts for the vault volume, in leakage from accessible areas, losses of water vapour to stack from the exhaust dryer, dryer performance, PHT tritium concentration and both chronic and acute dose scenarios.

The model calculates the vault tritium concentrations as a function of dryer performance.

The results of the model calculations under typical operating conditions in Ontario reactors (nominally 1 Ci/kg in the PHT system) are shown in Figures 2 and 3. Two important observations are:

1) Increasing dryer performance to achieve outlet dew points lower than –30°C has marginal impact on vault tritium MPC(a) levels.

2) Operating at a PHT tritium concentration of 1 Ci/kg does not lower the acute release scenario below the 100 MPC(a) threshold for the use of respirators.

The model was re-run with a lower PHT tritium concentration of 0.3 Ci/kg with the following results:

As Figures 4 and 5 illustrate, operating with a PHT system heavy water tritium concentration of 0.3 Ci/kg would safely ensure that the vault tritium MPC(a) level is below the threshold for safe use of respirators in both chronic and acute release scenarios.

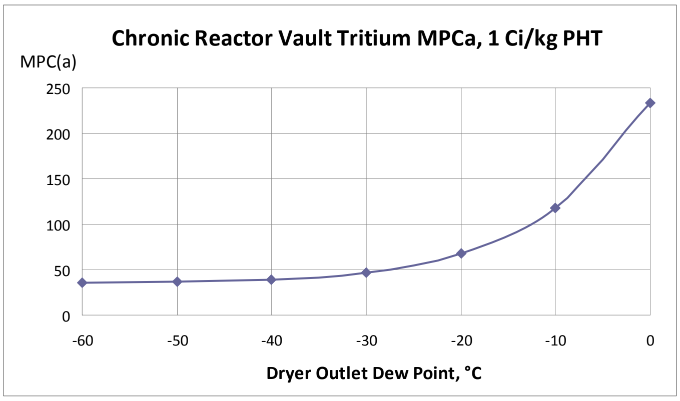

FIG. 2. Vault tritium MPC(a) with 1 Ci/kg PHT – chronic release scenario.

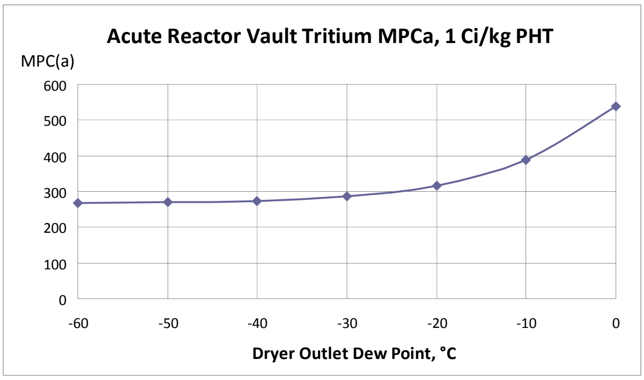

FIG. 3. Vault tritium MPC(a) with 1 Ci/kg PHT – acute release scenario.

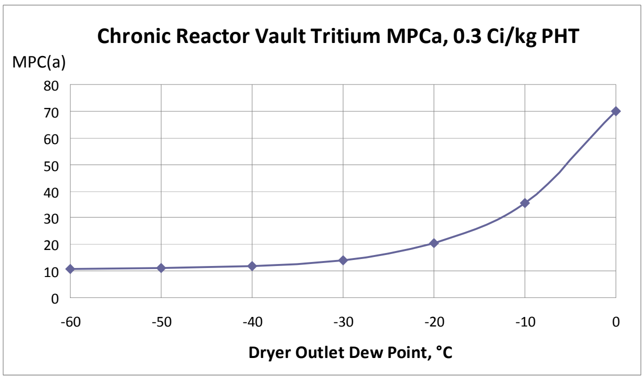

FIG. 4. Vault tritium MPC(a) with 0.3 Ci/kg PHT – chronic release scenario.

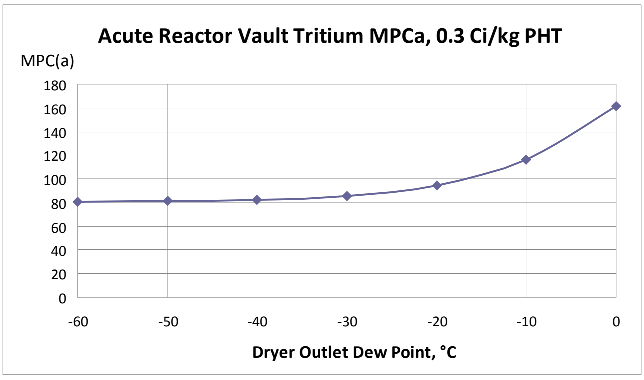

FIG. 5. Vault tritium MPC(a) with 0.3 Ci/kg PHT – acute release scenario.

5. Achieving low PHT heavy water tritium concentrations

In order to achieve this dramatic reduction in vault tritium MPC(a) levels, the PHT heavy water must have significantly low tritium concentration.

As previously discussed, tritium is continuously produced within the PHT system through neutron capture by deuterium. Roughly 12 Ci/h of tritium is generated in an 850 MW CANDU heat transport system. As in any continuous process, in order to maintain low concentrations of an impurity a purge stream is required. In this case, given the high value of virgin heavy water, a tritium separation technology must be employed so that detritiated heavy water can be returned to the reactor.

Current solutions employed for tritium separation at HWR stations have used cryogenic distillation. The use of cryogenic distillation technology for detritiation of heavy water has proven to be complicated, expensive and susceptible to low reliability, mainly due to complications associated with the cryogenic process. Also, given the cost and physical practicalities of the technology, existing facilities have provided limited detritiation of the heat transport system.

GE has developed a new concept: the Tritium Separation Centre (TSC)[4]. This facility houses new, proprietary GE processes and technology capable of extensively detritiating both the moderator and heat transport systems' heavy water.

While the heat transport system is attributed to causing chronic dose of operators and maintainers, it is the moderator that dominates environmental emissions[5] and acute-dose risk when tritium levels are high. The moderator of a 850 MW CANDU continuously produces around 250 Ci/h of tritium and without tritium removal, the moderator tritium concentration will rise to approximately 93 Ci/kg, at which point tritium production is balanced by radioactive decay and losses. Several HWR stations have already seen tritium levels in excess of 65 Ci/kg after years of operation without tritium removal.

The TSC solution is a new and simpler process designed to replace cryogenic distillation, employing a combination of gaseous diffusion and thermal diffusion. This process is a less expensive and safer combination of isotope separation technologies to replace cryogenic distillation. The attractive feature of this technology is simplicity, low inventory, scalability, and no requirement for cryogenic systems with their inherent complexity. The simplified flow schematic for the TSC is as shown in Figure 6.

A rigorous process model with mass and activity balance including the station moderator and PHT has been developed to show how a TSC facility could be integrated into an existing HWR site. The model includes tritium production source terms in both moderator and PHT systems. Figure 7 shows the scheme with molar flows, concentrations and production rates.

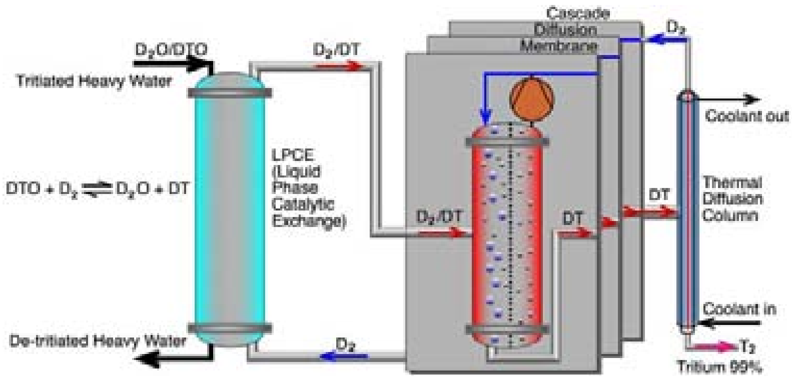

FIG. 6. TSC process block diagram.

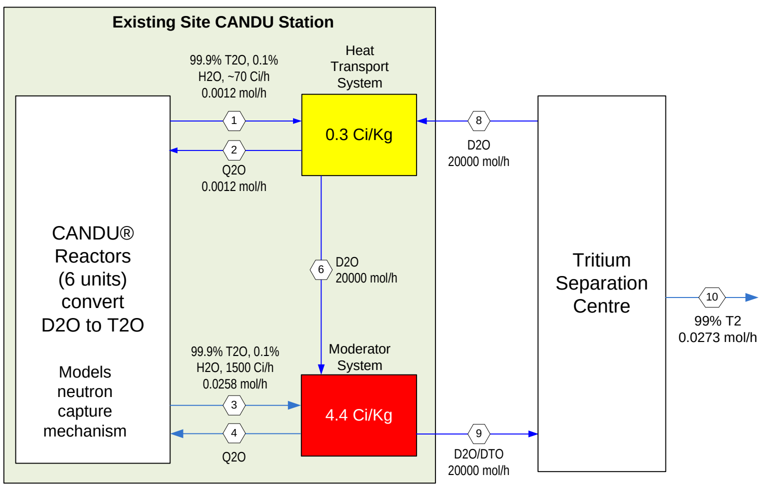

FIG. 7. Tritium separation centre model for a 6 reactor scheme.

The illustrated scheme represents a CANDU station with six 850 MW reactors on site served by the TSC with a throughput capacity of 400 kg/h (20 kmol/h) D_2O. The steady state moderator concentration is maintained below 5 Ci/kg, which exceeds currently best in class requirements, while the heat transport system is maintained to below 0.3 Ci/kg. At this level, the routine vault tritium MPC(a) concentration would be lower than 100 MPC(a) threshold meaning that outage work could be conducted without plastic suits[6] — leading to significantly reduced outage duration while reducing maintainer dose.

It is possible to achieve even further vault tritium MPC(a) reduction for outage work from the same onsite TSC. This is achieved by dedicating the TSC to processing the heat transport system heavy water of the reactor planned for the outage for a 2-3 month campaign prior to the outage. Figure 8 illustrates the impact of such a dedicated campaign on the heat transport system tritium concentration. Achieving levels of around 0.02 Ci/kg PHT tritium concentration would lead to very low vault tritium concentrations, less than 1 MPC(a) for chronic release scenario.

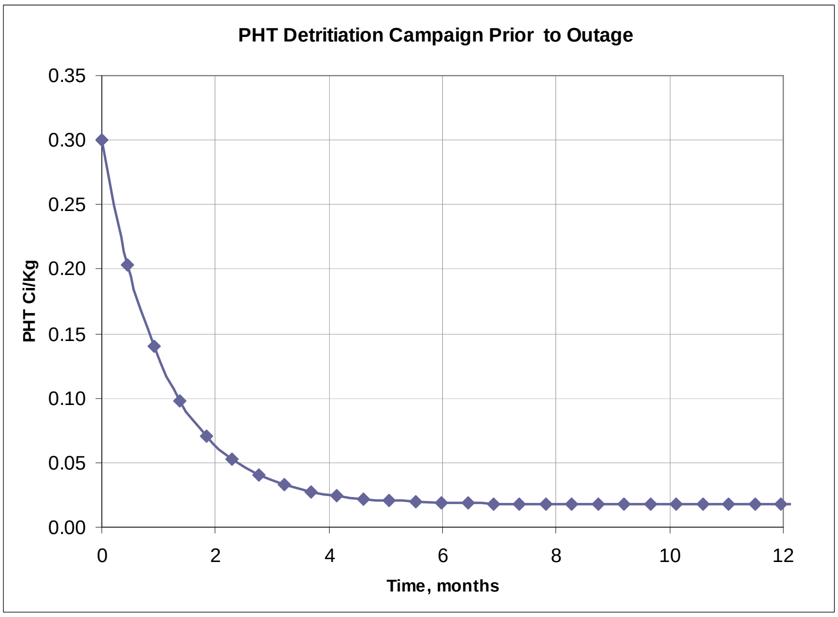

FIG. 8. Dedicated PHT detritiation of one reactor prior to shutdown.

6. Conclusion

GE has a long history of bringing innovative best practice solutions to the HWR industry. The establishment of the Light Isotope Technology Centre of Excellence in Peterborough, Ontario has centralised GE's knowledge base of heavy water and tritium handling. The Centre of Excellence has developed new technology and through analysis of existing HWR operation, provides an opportunity, through GE's innovative Tritium Separation Centre (TSC) product, to extend best practice in management of heavy water and tritium at HWR stations.

GE has already provided a TSC concept design study for the CANDU Owners Group, this provided a description of the technology and project overview for two generic sites – single reactor and multi (4) reactor. The TSC can bring significant benefits to an operating HWR utility such as reducing environmental emissions and significantly lowering reactor vault tritium MPC(a) levels to a point where station capacity factors can be improved by shorter outages – representing best practice.

Utilising GE's technology advancement in heavy water detritiation offers the possibility to introduce best practices for utilities operating HWR in the management of heavy water and tritium.

REFERENCES

[1] BONNETT, I., BUSIGIN, A., LASHFORD, A. & WILLIAMS, H.R.P., “Radioactive Waste Recovery and Enrichment”, 9th International Symposium on the Synthesis and Application of Isotopes and Isotopically Labelled Compounds, Edinburgh, July 16-20, 2006.

[2] BALTEANU, O., STEFAN, J. & RETEVOI C., “Safety System in a Heavy Water Detritiation Plant”, Proceedings of the International Conference Nuclear Energy for New Europe, Portorož, Slovenia, Sept. 8-11, 2003.

[3] UNSCEAR (United Nations Scientific Committee on the Effects of Atomic Radiation), “Sources and Effects of Ionizing Radiation. Report to the UN General Assembly”. United Nations, New York: 1977.

[4] BUSIGIN, A. & BONNETT, I., “Combined Heavy Water Upgrading And Detritiation For Candu Applications”, 28th Annual CNS Conference & 31st CNS/CNA Student Conference, Saint John, New Brunswick, Canada, June 3 – 6 2007.

[5] SON, S.H., SONG, K.M. & CHUNG, Y.G., “Tritium Removal and Applications for Wolsong Heavy Water Reactors”, presented at the 8th International Conference on Tritium Science and Technology, Rochester, NY, September 16-21, 2007.

[6] WONG, K.Y., KHAN, T.A., GUGLIELMI, F, “Canadian Tritium Experience”, Canadian Fusion Fuels Technology Project, Ontario Hydro, 1984.

REDUCTION IN TRITIUM EMISSIONS AND WORKER INTERNAL UPTAKES THROUGH SOURCE TERM REDUCTION, DRYER PERFORMANCE AND ONLINE TRITIUM MONITORING

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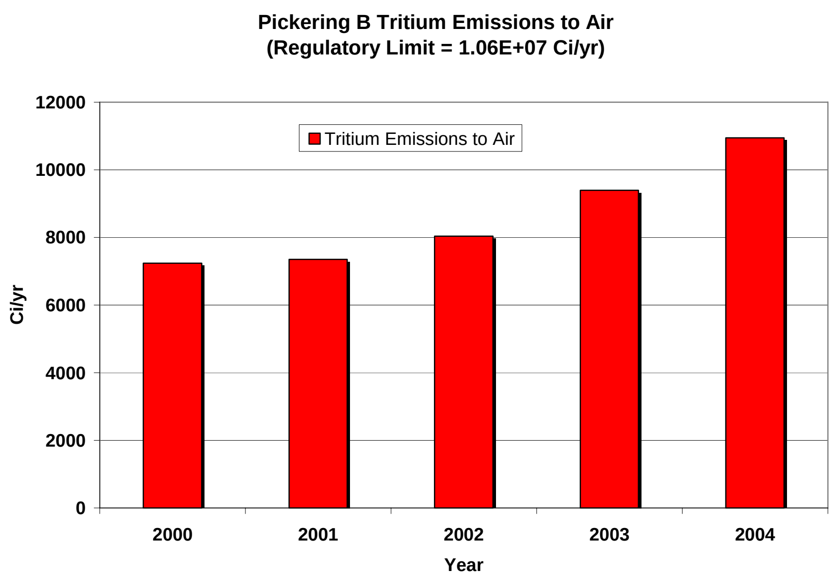
Abstract

In September 2004 a team composed of Radiation Protection, Operations, Maintenance, and Environment and Chemistry was formed under the leadership of the Director of Operations and Maintenance (DOM) to reduce tritium airborne emissions and reduce internal worker dose. Continued focus on the key aspects of the plan brought success. At year-end of 2008, a five-year reduction of over 40% in both tritium emissions and worker dose are expected.

1. Identification of Problem

Although Pickering B airborne tritium emissions to the environment are orders of magnitude lower than the regulatory requirements, over the period 2000-2004 these emissions showed a negative trend.

By August 2004, it was clear that the station internal target for tritium emissions from Pickering B for that year, 8300 Ci,[1] would be surpassed unless unreasonable actions were taken (shutting down units).



2. Action Plan

Consistent with senior management's commitment to 'fix the plant', in September 2004 a team composed of Radiation Protection, Operations, Maintenance, and Environment and Chemistry was formed under the leadership of the Director of Operations and Maintenance (DOM). The team's goal was two pronged: reduce tritium airborne emissions and reduce internal worker dose.

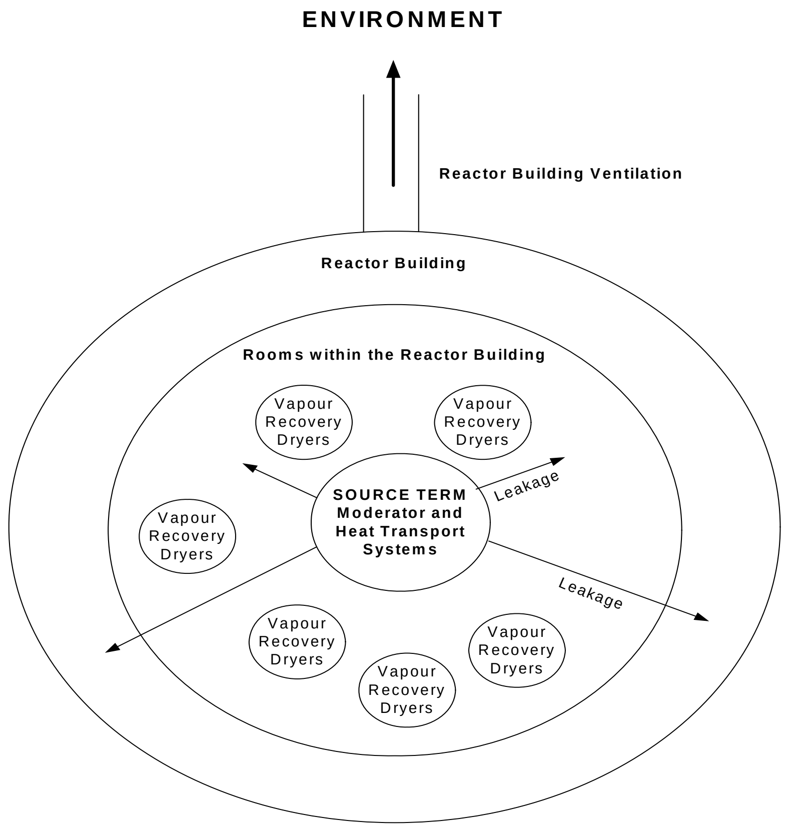
[1] 1 Ci = 3.7×10^{10} Bq.

Tritium emissions to air result from a combination of:

- a build-up of tritium in heat transport and moderator systems (moderator tritium concentrations increase at a rate of 2-4 Ci/kg per year full power operation),

- leaks from tritiated heat transport and moderator systems within containment (the reactor building),

- lack of vapour recovery dryers in specific locations within containment, or containment vapour recovery dryers that are not performing at optimum levels.



A tritium reduction action plan was developed to achieve the following:

- Catalogue all leaks and work down the repairs. These include such things as leaking valves and bellows, leaking Hansen fittings, and Moderator and Heat Transport Systems sample cabinet leaks.

- Improve dryer performance through replacement of: desiccant (the drying agent), cooling coils, relays, temperature switches, breakers and contactors, and modification of sediment traps and 3-way dampers, to improve reliability.

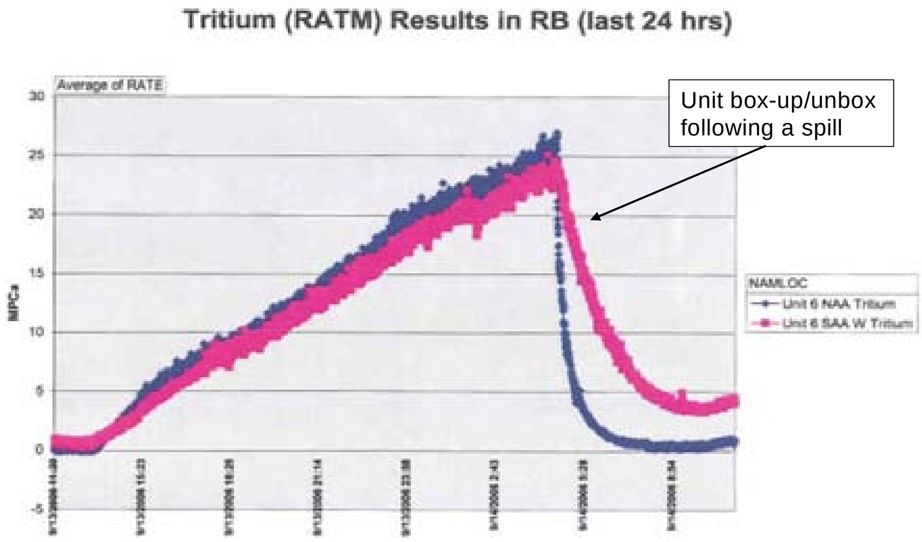
- Reduce the tritium concentration in the Moderator and Heat Transport systems by shipping tritiated heavy water to Darlington's Tritium Removal Facility (TRF) in return for detritiated water.

- Communicate recovery plans, commitments and timelines to plant management to ensure employee engagement.

- Track performance of these key items against expectations through regular meetings between plant staff and the DOM.

In addition, to improve surveillance Real Time Tritium Monitors (RATMs) were purchased and installed at key locations within the reactor building. These devices allowed for better work planning and reduced worker exposure.

Pictured below are rising tritium concentrations within the reactor building following a ventilation box-up as seen from the RATMs.



3. Success and Results

Results of Executing the Tritium Reduction Action Plan:

Leaks Fixed

Year	2005	2006	2007	2008
	100	50	60	60

Dryer Repaired

2005:

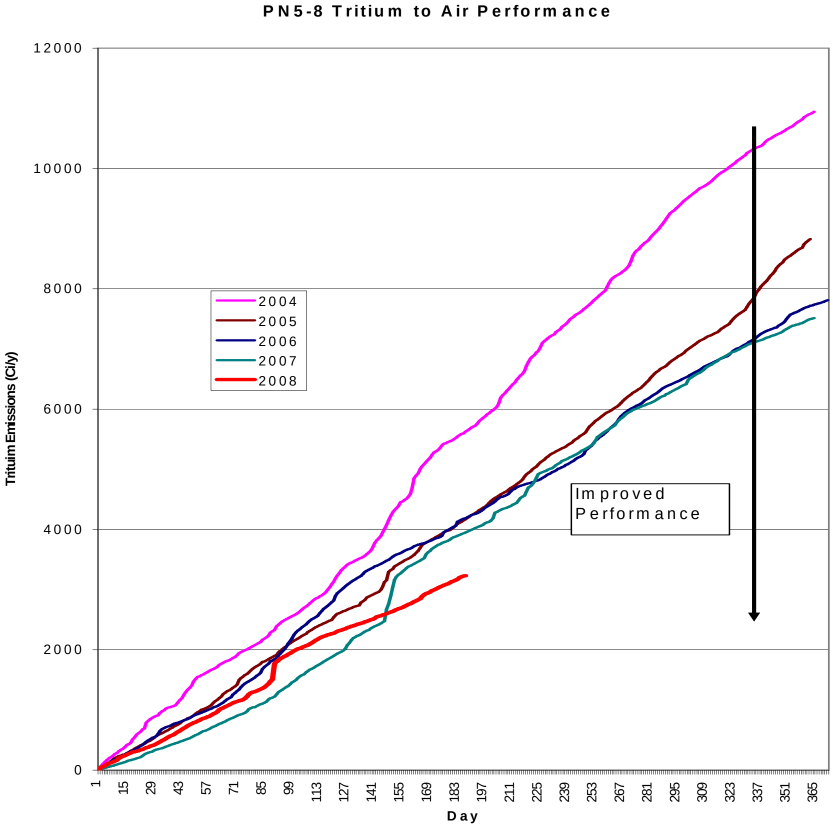
- Heat Exchanger coil replacements - 13
- Desiccant replacements - 18
- Sets of temperature switches - 40
- 3-way damper modifications - 15
- Hygrometers replaced - 20
- Relay upgrades - 3

2006/2007:

- 44 power panels replaced (ensures the maximum # of heaters can be used during the regeneration part of the drying cycle).

Reduction in Tritium Concentrations

Year	2004	2005	2006	2007	Present
Moderator tritium concentration (Ci/kg)	20	18	16.5	12.5	12.9
Heat Transport tritium concentration (Ci/kg)	1.15	1.1	1.05	1.08	1.1



Resulting Performance Improvements – Tritium Emissions to Air

Resulting Performance Improvements – Internal Tritium Dose (person-rem/unit)

Year	2004	2005	2006	2007	Present
	34.38	29.41	26.20	18.8	10.2 YTD

As indicted in the results above, continued focus on the key aspects of the plan brought success. At year-end of 2008, we expect to see a five-year reduction of over 40% in both tritium emissions and worker dose.

In addition to continuing to improve on the key areas described above, some further actions will be considered:

- Ensuring all units are operating consistently and at optimum performance (compare unit by unit performance);
- Using improved metrics such as tracking to a target: moderator tritium concentrations, heavy water shipments to the Darlington TRF, dryer availability/system health, number of outstanding leaks;
- Continue communication of successes and challenges;
- Continue to push for improved performance through teamwork (system health teams);
- Develop and manage a longer term strategic plan.

SYSTEMATIC COLLECTIVE DOSE REDUCTION

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Abstract

It was recognized from the early days that collective doses in Indian NPPs were higher than international values. To adopt a holistic approach, all the players in the country came under one roof: Operators — NPCIL, Regulators — AERB, and the technical support organization — BARC. A workshop was organized to focus the knowledge and resources to review, analyze and define future actions. Setting clear and high expectation from company executives, using a holistic approach to address important issues, reinforcing of training, the continuous evolution of radiation-protection practices, and a management-observation programme have yielded good results and led to a reduction in collective doses in NPPs.

1. Introduction

At present, seventeen nuclear power reactors are being operated in India by the Nuclear Power Corporation of India, Limited (NPCIL). Nuclear power was first produced in India in 1969 with commissioning of Tarapur Atomic Power Station (TAPS), consisting of two boiling water reactors as a turnkey project by General Electric Company (GE) USA. Subsequently, as per a policy decision of the Government of India, several nuclear power plants (NPPs) based on Pressurized Heavy Water Reactors (Pressurised HWRs) were set up starting with the Rajasthan Atomic Power Station (RAPS unit-1&2). Since the inception of the programme, priority has been given to the adoption and maintenance of high safety standards.

In 1987, the NPCIL was formed as a public-sector enterprise wholly owned by the Government of India, under the administrative control of the Department of Atomic Energy (DAE), with the objective of undertaking the design, construction, operation and maintenance of nuclear power stations for generating electricity. The mission of NPCIL is to produce electricity and develop nuclear power technology as a safe, environmentally benign and economically viable source of electrical energy to meet the increasing energy needs of the country.

All Indian NPP sites have an extensive programme in place for monitoring environmental impact. An Environmental Survey Laboratory (ESL) is established at a new site well before commencement of operation of a NPP to facilitate baseline data. The aim of the environmental monitoring and surveillance programme is to assess the radiological impact of NPP operation and to demonstrate compliance with the radiation exposure limits for members of the public set by the Atomic Energy Regulatory Board (AERB), the national regulatory body. The area up to a distance of about 30 km is covered under the environmental survey programme. The estimated doses to the public at the exclusion boundary of the operating NPPs have been a very small fraction of the AERB prescribed dose limit of 1000 μSv per year.

The radiological exposures of all radiation workers are scrupulously monitored and records are maintained. The AERB has prescribed a limit of 20 mSv per year for a radiation worker, averaged over five consecutive years, and a maximum of 30 mSv in any year. The limit for contract workers has been kept at 15 mSv in a year. To enforce these targets, the nuclear power stations have introduced in-house limits for individual dose on a daily, monthly and quarterly basis, for exercising the control and in-house reviews. The AERB committee also carries out an investigation if the exposure of any radiation worker exceeds the limits prescribed by regulator.

For effective implementation of radiological-safety and environmental-surveillance programmes, the Health Physics Units (HPUs) and Environmental Survey Laboratory (ESL) have been independent of station management since beginning of the nuclear power programme. To meet the unlikely situation of an accident, well thought out formal emergency preparedness plans are in place.

2. Safety review of existing nuclear installations

Operating nuclear installations in India are subjected to continuous regulatory appraisal of safety performance as per the established procedures. The operational performance and safety significant events are reviewed and any required corrective measures are implemented.

A comprehensive periodic safety review of operational and safety performance of NPPs is carried out at the time of renewal of authorisation or major refurbishment or for plant-life extension. These reviews include factors like changes in safety standards, ageing, new information, trends in collective doses, etc. Such reviews bring out requirements for modification and safety up-grades, particularly in older plants that were built to earlier safety standards.

Initially, the collective doses in NPPs were relatively high compared to prevailing industry standards. Many of the issues affecting collective dose in the older HWRs were addressed in the new generation of reactors. This was achieved by design improvements and/or improving the working conditions. These improvements have contributed to a reduction in collective dose from 4.0 p-Sv/reactor in RAPS/MAPS (older NPPs) to around 2.0 p-Sv/reactor in KAPS and less than 1.0 p-Sv/reactor in KGS/RAPS 3&4 (latest NPPS).

3. Collective dose reduction programme

It was recognized from the early days that collective doses in Indian NPPs were higher than international values. At the same time, plant performance in terms of capacity factors and availability factors was lower. Traditional efforts to reduce collective doses were not fruitful. Higher collective doses were attributed to frequent reactor shutdowns, modifications and maintenance, etc. Over the years various initiatives were implemented. These resulted in improved performance of NPPs in terms of capacity factors and availability factors, but the collective doses did not decline significantly.

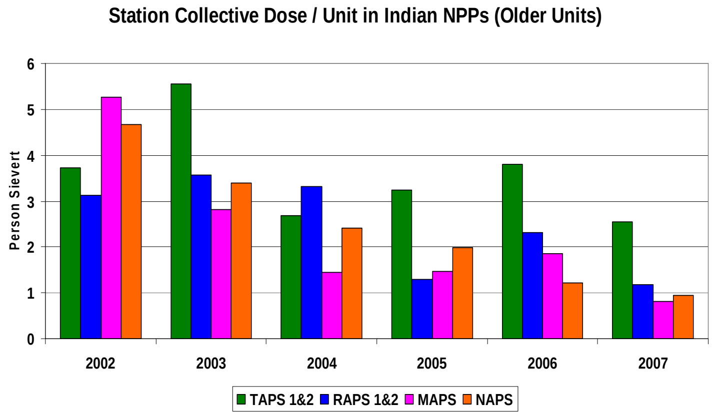
To adopt a holistic approach, all the players in the country came under one roof: Operators – NPCIL, Regulators – AERB, and the technical support organization- BARC. A workshop was organized to focus the knowledge and resources to review, analyze and define future actions. The dose-reduction process was developed based on benchmarking with the WANO performance indicator for collective doses in various NPPs around the world. To achieve tangible results, the NPPs were grouped in two sections depending upon age.

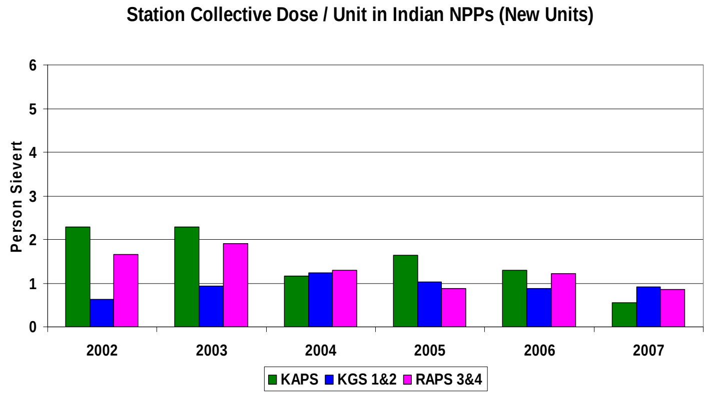
The approach covered all aspects including identification of issues, design changes, improvement in quality assurance plans, procedures and practices, optimisation of deployment of manpower, use of remote tools, extensive training and management-observation programme, etc.

4. Area identification

Areas were identified that were considered leading contributors to increasing collective doses. The identification was achieved by the formation of various focus groups. The result of analysis was as follows:

1) High radiation fields on piping and equipments.
2) Higher specific activity in coolant.
3) Frequent equipment failures, repeat jobs etc.
4) Events of Tritiated heavy water leakages/spillages.
5) Immediate commencement of work in shutdown-accessible areas after shutdown.
6) Large number of persons involved in maintenance and operation activities.
7) Inadequate use of temporary shielding.
8) Manual radiation work-control permits, dose data keeping and review.
9) Manual display of area radiation fields.
10) Shortfall in adhering to radiation-protection procedures.





11) Higher Tritium specific activity in air in access control areas.
12) Increased in-service inspection (ISI) requirements.
13) Coolant channels life management activities.
14) Aging of equipment/plants.
15) Heavier mask air hoses.
16) In adequate radiation hot spot management.
17) Insufficient use of system mockups.
18) Insufficient training programme.

5. Development of corrective actions

A multi-pronged approach was adopted to address the above areas so as to achieve the desired results. Areas covered included administrative measures, enhancement and reinforcement of quality of procedures and practices, and incorporating several design modifications for reduction of external as well internal radiation exposures to plant personnel.

5.1. Administrative measures

1) Setting up targets to reduce collective doses to 70% in next 3 years for old plants (10%/yr).
2) Redefining targets of collective doses for new plants in line with prevailing industry standards.
3) To budget (Plan in advance) the collective dose at the beginning of every calendar year, giving details of works planned, including dose commitment.
4) Introduction of concept of collective dose constraint at 80% of above budgeted dose when plant operator are required to inform regulator.
5) Decontamination or replacement of highly contaminated equipments as a policy.
6) Introduction of penalty clause for contract workers for radiation-protection rule violations.
7) Overview the preparedness prior to commencement of high person-rem intensive jobs.

5.2. *Work practices*

1) Restricting entry to shutdown-accessible areas immediately after reactor shutdown.
2) Keeping coolant under pressure and under purification after shutdown.
3) On line purification of Moderator and PHT systems water even during reactor cold and shutdown state.
4) Drying/decontamination of heat exchangers before taking ISI activities.
5) Use of umbrella concept of isolation and maintenance during shutdowns. (Elimination of repeat isolation and normalization).
6) Revised procedure to keep adjusters/PSS elements inside the core while working in RCM area.
7) Extensive use of temporary shields, Lead aprons, lead fiber mats, pile up shields.
8) Discontinuing practice of slurring the PHT resins for ejection.
9) Optimizing ISI activities, heat exchangers and steam generators.
10) Avoid power ramps and follow a defined power-raise programme for improved fuel performance.
11) Enhancement of OE sharing -periodic meet of concerned departments.
12) Reinforcement in use of maintenance procedures to avoid frequent failures/repeat jobs.
13) Replacement, ejection or recharging of ion exchange columns preferably during reactor shutdown.
14) Use of mixed based resins for chemical control.
15) Chemical decontamination of primary heat transport system, essentially prior to major refurbishment job such as en-masse coolant channel replacement.

5.3. *Design modifications/improvements*

1) Removing shortfalls in fuel fabrication and improving the fuel bundle testing techniques.
2) Adoption of Cobalt free material for reactivity mechanism balls, pump seals.
3) Incorporation of quick disconnects type electrical connectors in MOVs. (time saving).
4) Increased use of remote operated tools for ECT of heat exchanger tubes.
5) Creep measurement with the help of the fuelling machine.
6) Incorporating changes in layout of piping or changes in access control route.
7) Elimination of potential source of light water leaks in shutdown accessible areas (additional valves installation, deletion of redundant valves, installation of caps at open ends and seal welding, use of bellow seal valves etc.)
8) Replacement of hot insulation having metal sheets with quick connectors- reduced work duration in installation /removal.
9) Modified Jigsaw panels, reduced number of panels and reduced fasteners.
10) Shielding of crane operator cabins (save even small contributors).
11) Replaced of pumps having mechanical seals with canned rotor pumps in Moderator system.
12) Replacement of Zircaloy-2 pressure tubes with Zr-Nb 2.5 pressure tubes to reduce work required for creep elongation adjustments, component replacement.

5.4. *Improvements to reduce internal doses*

5.4.1. *Tritium source control*

1) Improving leak tightness of heavy water systems — Routine watch on LC pump out (1/day), LIGs etc, use of D_2O sniffer.

2) In-situ post maintenance leak tightness checking of inter space of double gasket joints, before charging D_2O.
3) Installation of vent condenser in PHT LC system vent line.
4) Incorporation of PHT LC system integrity test during long shut downs.
5) Use only fresh water in ALPAS tanks (change in procedure from using system water having high ${}_1H^3$).
6) Flushing of heat exchangers with fresh (Uncontaminated) D_2O and drying thereafter to bring down Tritium content in air prior to ISI/maintenance.
7) Diaphragms (elastomers) replacement on routine basis as PM.
8) Adopting internal draining of D_2O system and equipment.
9) Maintaining strict segregation between PHT and Moderator water during collection and processing.
10) Swapping of water- during refurbishment of plants PHT water of comparatively low Tritium is drained, up graded and used in Moderator system, thereby reducing Tritium in high pressure & high temperature PHT system).
11) Elimination of potential source of heavy water leaks (additional valves installation, deletion of redundant valves, installation of caps at open ends and seal welding).
12) Use of chloride free insulation for D_2O stainless steel tubes & pipe to avoid stress corrosion cracking such as Delayed Neutron Monitoring tubes, Instrument tubes and LC pipes.
13) Ensuring D_2O draining prior to opening of equipment/system.
14) Ensuring system integrity prior to charging of D_2O
15) Ensuring system configuration prior to D_2O transfer/charging.

5.4.2. Tritium DAC[1] control

1) Improved availability and performance of D_2O vapour recovery dryers, particularly during shutdown. Versatile mode of operation of one dryer.
2) Improving V1-V2 integrity.
3) Ventilation flow balancing.
4) Increased purge &Temporary Supply of fresh air in S/d accessible area during maintenance.
5) Dedicated DAC coordinator during shutdown.
6) Centralized vacuum mopping system to collect spilled heavy water.

5.5. Upgrades in radiological monitoring/protection

7) Introduction of computerized dose management system, dose trending.
8) Introduction of display of area radiation field and Tritium DACs on local computer network.
9) Large-scale use of alarm dosimeters.
10) On-line display of area radiation monitors in shift health physics control room.
11) Computerizations of radiation work permit control.
12) Use of light-weight mask air hoses.

5.6. Reinforcement of other areas

13) Reinforcement of training and retraining of workers, especially crew based training.
14) Introduction of practical demonstration and hands-on practice of using protective equipments.
15) Pre job briefing on radiation-protection aspects.
16) Reinforcement of review meetings to achieve ALARA doses for dose intensive activities.

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[1] Derived air concentration (typically of tritium).

17) Management observation programme by line managers.
18) Corporate review programme based on WANO-Peer Review methodology.

6. Conclusion

Setting clear and high expectation from company executives, using a Holistic approach to address important issues, reinforcing of training, the continuous evolution of radiation-protection practices, and a management-observation programme have yielded good results and led to a reduction in collective doses in NPPs.

DOWNSIZING OF THE PHT PURIFICATION FILTER CARTRIDGE IN WOLSONG UNIT 1

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Abstract

To protect workers and plant equipment from radiation exposure, Wolsong Unit 1 has changed the PHT purification filter cartridge from a rating of 6.0 µm absolute (abs.) to 0.45 µm abs. in stages. The plant evaluated the effect of the adopted fine filter cartridge in 2006. It was found that use of fine mesh cartridge is effective to reduce the effect of radioactive corrosion products.

1. Introduction

As the operating time goes on, radioactivity in the PHT system of Wolsong Unit 1 has increased rapidly. To protect workers and plant equipment from radiation exposure, Wolsong Unit 1 has changed the PHT purification filter cartridge from a rating of 6.0 µm absolute (abs.) to 0.45 µm abs. in stages. The plant evaluated the effect of the adopted fine filter cartridge in 2006 as below.

2. History of changing the PHT purification system filter cartridges

The records of Unit 1 for changing filter cartridge

DATE	System	Filter Size		Note (Unit 2)
		Before	After change	
May, 2000	Purification Filter	5.0 µm Nom.	6.0 µm abs.	Sep, 2000
Oct, 2001	Purification Filter	6.0 µm abs.	2.0 µm abs.	Jan, 2002
Sep, 2005	Purification Filter	2.0 µm abs.	0.45 µm abs.	Mar, 2007
Jan, 2008	**Purification Filter**	**0.45 µm abs.**	**0.1 µm abs.**	**May, 2008**

Filtering Efficiency

Filter Ration (Pore size)	100%	99%	90%
	Particle size (µm)		
5.0µm	5.0	2.5	1.5
2.0µm	2.0	0.8	0.3
1.0µm	1.0	0.6	0.25
0.45µm	0.45	0.2	0.1

3. Evaluation of the changing filter

3.1. A transition of the CRUD[1] concentration in the PHT system

- The Co-58 concentration in the PHT system has decreased by 75% in the 19th plant outage as compared with the 14th (Ref. Table 1 & Fig. 1)

- The concentration of Mn-54 & Nb-95 decreased by 9.8%, 6.2% each in the 19th outage compared with the 18th's. (Ref. Table 2 & Fig. 2)

[1] CRUD = Chalk River Unidentified Deposit.

Table 1. Average concentration of Co-58

Outage	14^{th} O/H	15^{th} O/H	16^{th} O/H	17^{th} O/H	18^{th} O/H	19^{th} O/H
Period	99–00	00–01	01–03	03–04	04–05	05–06
Co-58 (μCi/cc)	1.74E-3	2.54E-3	1.52E-3	9.50E-4	4.78E-4	4.23E-4
Decrement (%)	-	46.2	-12.5	-45.3	-72.5	-75.7

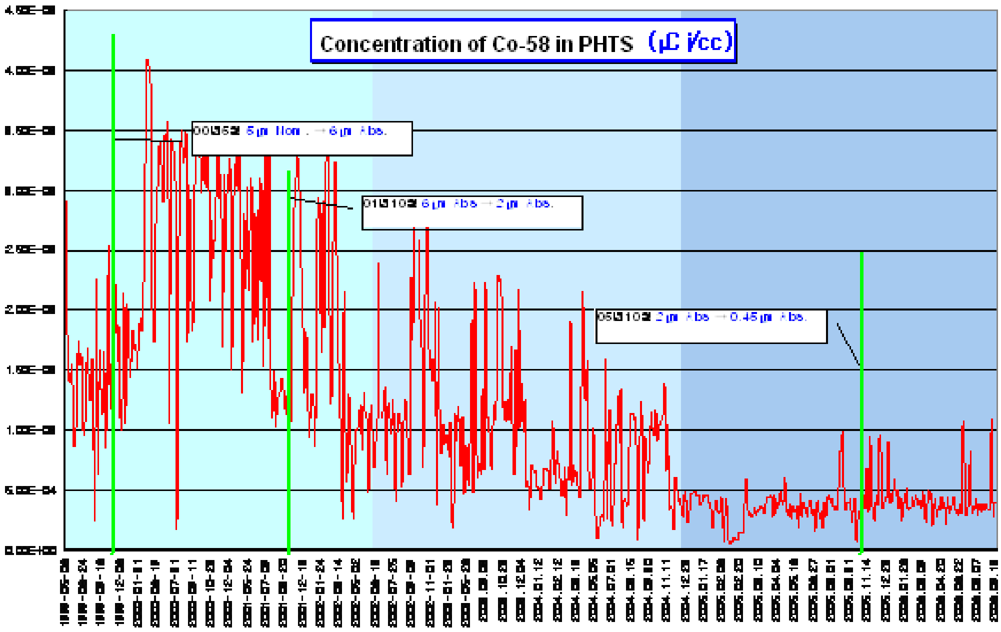

FIG. 1. A transition of Co-58 in PHT system.

Table 2. Average concentration of Mn & Nb

Nuclides	Mn-54	Nb-95
Before 2005	1.32E-4	6.91E-4
After 2005	1.19E-4	6.48E-4
Decrement (%)	9.8	6.2

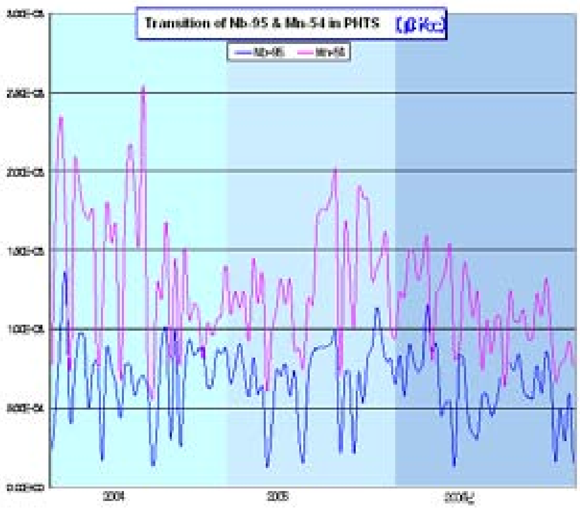

FIG. 2. A transition of Nb-95 & Mn-54 in PHT system.

3.2. Analysis of CRUD constituents in the PHT system

The PHT system CRUD was composed of following radioactive nuclides.

- Niobium & Zirconium were the dominant nuclides of CRUD samples taken from the fuel sheath and the pressure tube material. (80–90%).
- Cobalt generated from the steam generator material was less than 3% of the CRUD.
- Manganese (Mn), the critical ingredient of carbon steel, was less than 1% of the CRUD.

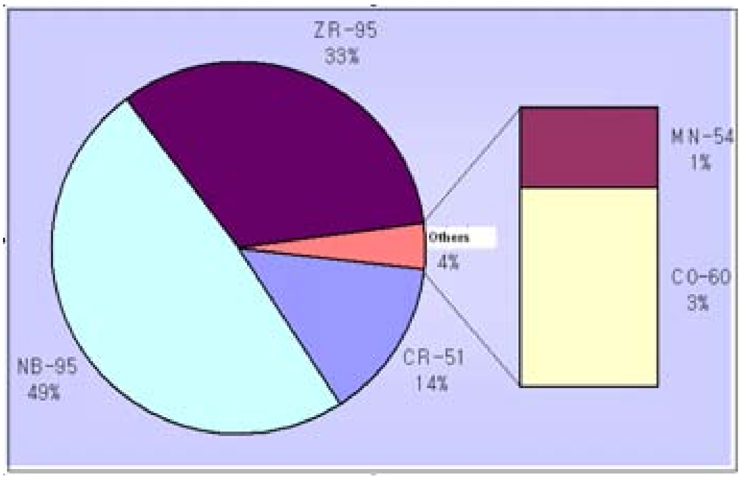

FIG. 3. Wolsong Unit 1 CRUD Ratio in the PHT system.

FIG. 4. Wolsong Unit 2 CRUD Ratio in the PHT system.

- The dominant CRUD nuclides (Nb & Zr) in the Unit 1 PHT system were just 8 percent compared with Unit 2's concentration after changing filter cartridge of 2 μm abs. to 0.45 μm abs.
- The total CRUD concentration in the Wolsong Unit 1 was merely 6.7 percent of Unit 2.

From this, it is concluded that changing the PHT system filter cartridge has proven an effective method of maintaining CRUD nuclides at a low level.

Table 3. Comparison of CRUD concentration Unit 1 and Unit 2, µCi/cc

	Nb-95	Zr-95	Cr-51	Co-60	Mn-54	Tot. Activity
Unit 1	3.81E-3	2.61E-3	1.09E-3	2.14E-4	8.78E-5	6.90E-3
Unit 2	6.91E-2	3.43E-2	2.10E-3	1.43E-3	9.59E-4	1.08E-4
Unit 1/Unit 2	1/18	1/13	1/2	1/7	1/11	1/15

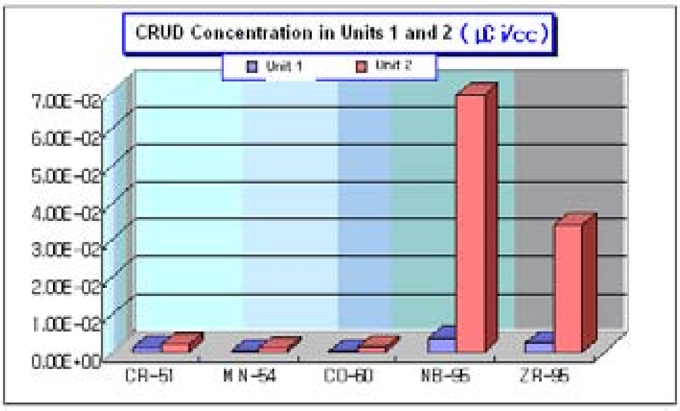

FIG. 5. Comparison of CRUD concentration of Unit 1 with Unit 2 in 2006.

3.3. Evaluation of steam generator worker dose rates

- There was no sound basis for comparing worker dose rates in 2005 and 2006 for work on steam generators, but we did observe a small reduction of 14 percent.

- It seems effective to adopt the fine filter cartridge in the PHT system, so Wolsong NPP plans to monitor results closely.

Table 4. Steam generator worker's dose rate

Year	**The Personnel**	**Man-hour**	**Man-mSv**	**Dose rate/time**
2005	34	306.3	45.26	0.148
2006	19	58	20.20	0.127

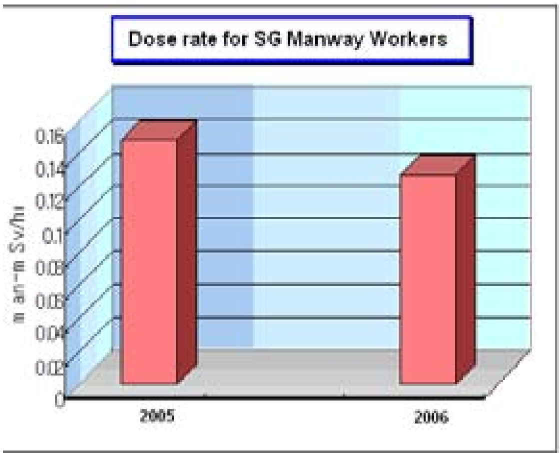

FIG. 6. Steam generator worker's dose rate.

4. Conclusions

- The radioactive corrosion products erode the fuel sheath and structures, increase wear on mechanical seals, and wear and tear the inner face of equipment. They also increase radiation levels inside the plant.

- To maintain the integrity of the PHT system, by reducing the effect of radioactive corrosion products, fine mesh filter cartridges can be used to remove CRUD.

- In our tests, we found the most effective filter change was changing the mesh of the PHT purification filter cartridge from 5 μm nominal to 0.45 μm absolute. in steps.

- Wolsong Unit 1 adopted 0.1 μm absolute filter cartridges in January 2008 so that we can get more detail results on CRUD concentrations in 2009.

PORTABLE DRYER USED AT CANDU STATIONS

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2. Point Lepreau Nuclear Generating Station, New Brunswick Power Commission, Canada
3. Gentilly 2 Nuclear Generating Station, Hydro Quebec, Canada
4. Munters Cargocaire, Amesbury, Massachusetts, United States of America

Abstract

Following the COG workshop on Heavy Water Management in 2000, the participants identified a need for portable vapour recovery dryers (VRDs) to be used during the station maintenance outage and to augment the existing heavy water vapour recovery dryers. A prototype VRD was designed, developed and successfully tested. At present there are a number of Portable VRDs in operation at numerous CANDU stations. At some CANDU stations, tritium emissions to the environment and tritium doses to operating personnel have increased during recent reactor maintenance outages, largely due to higher tritium concentrations within the rector's heavy water systems. The portable VRD is designed to provide effective tritium confinement as well as tritium emission and dose control to maintenance personnel during station outages. The size of the portable VRD is such that it is easily transportable into the accessible areas within reactor building to recover airborne heavy water that escapes from equipment that is opened for maintenance. The equipment can be housed in a temporary enclosure to eliminate the spread of tritium activity. This enclosure can also be dried by the portable VRD. The portable VRD is a rotary wheel desiccant type design. The wheel is composed of a fibreglass structure rotating slowly within a cylindrical casing and undertakes adsorption and regeneration simultaneously. The dryer unit consists of an adsorption fan, a regeneration fan, and an electrical heater with an integrated chiller unit. The rotary dryers can also be used in the CANDU-6 operation to minimize ingress of H_2O into the reactor building. In addition, the rotary dryers are also used at Pickering-A to dry the Calandria Vault atmosphere.

1. Introduction

The portable VRD is designed to provide effective tritium confinement, as well as tritium emission and dose control to maintenance personnel during station outages. The portable VRD size is such that it is easily mobile and equipped with flexible ducting that can be placed into inaccessible areas within the reactor building to recover airborne heavy water that escapes from equipment opened for maintenance. The equipment under maintenance can be housed in a temporary plastic tent or enclosure.

The portable VRDs can also be used in CANDU stations to augment the existing station heavy water ventilation system. The portable VRDs are ideally suited as a supplementary system in the event of an acute leak of tritiated heavy water at the station.

2. Standard desiccant dehumidification design

The dehumidifier's design is based on the unique honeycomb adsorption wheel that provides surface area for desiccant drying. The adsorption air passes through the desiccant drying wheel and leaves the dehumidifier as dry air. The heated regeneration or reactivation air removes the moisture adsorbed by the desiccant wheel and leaves the dehumidifier as wet air.

Figure 1 outlines the basic operating function of the desiccant dehumidification process. It should be noted in the dehumidification process, the wet air is discharged outside to the atmosphere. This type of dehumidification unit is used on the reactor building ventilation system where the outside air is dried to a dew point of -40°C prior to entering into the reactor building.

The maximum dew-point temperature achieved with this type of dehumidification arrangement is limited to -40°C. This arrangement adds heat to the process air stream during regeneration.

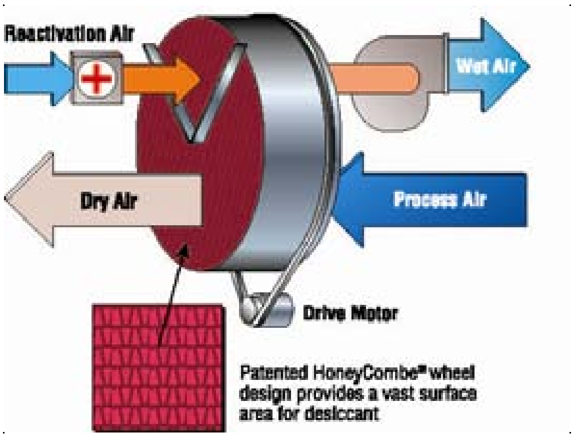


FIG. 1. Basic desiccant wheel configuration.

3. PowerPurge™ desiccant dehumidification design

The PowerPurge arrangement is a closed-loop system. This arrangement is slightly more energy efficient than the previous described scheme. In this arrangement heat is recycled to reduce reactivation energy requirement. However, due to the adsorption isotherm, it requires cooling of the purge air. Figure 2, below outlines the PowerPurge dehumidification arrangement.

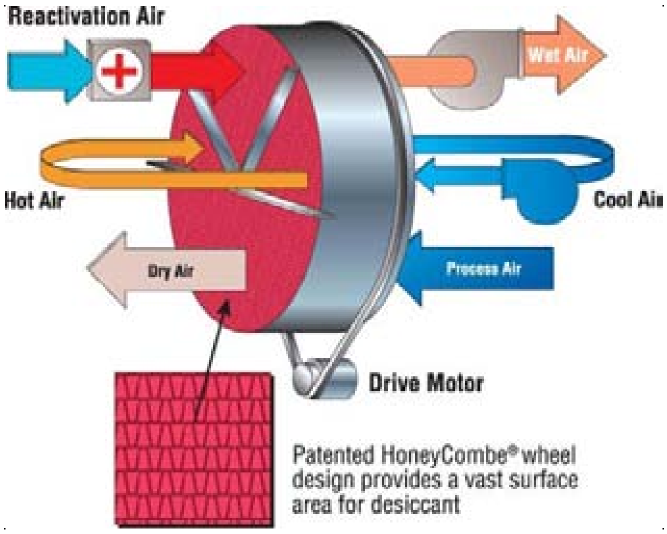


FIG. 2. PowerPurge desiccant wheel configuration[1].

4. Portable vapour recovery dryers (VRDs)

The portable VRD system can be operated in a closed circuit. The system's ductwork is used to establish the closed-loop circulation. The system flow schematic is as shown in Figure 3. In this configuration, no air is exhausted to the atmosphere.

The adsorption flow capacity is 600 scfm[2] of humid process air drawn from the enclosure. This passes through the desiccant wheel, which extracts the moisture from the air. The outlet dry air at a –26°C dew-point temperature is pre-cooled and returned to the enclosure. In the regeneration (or reactivation)

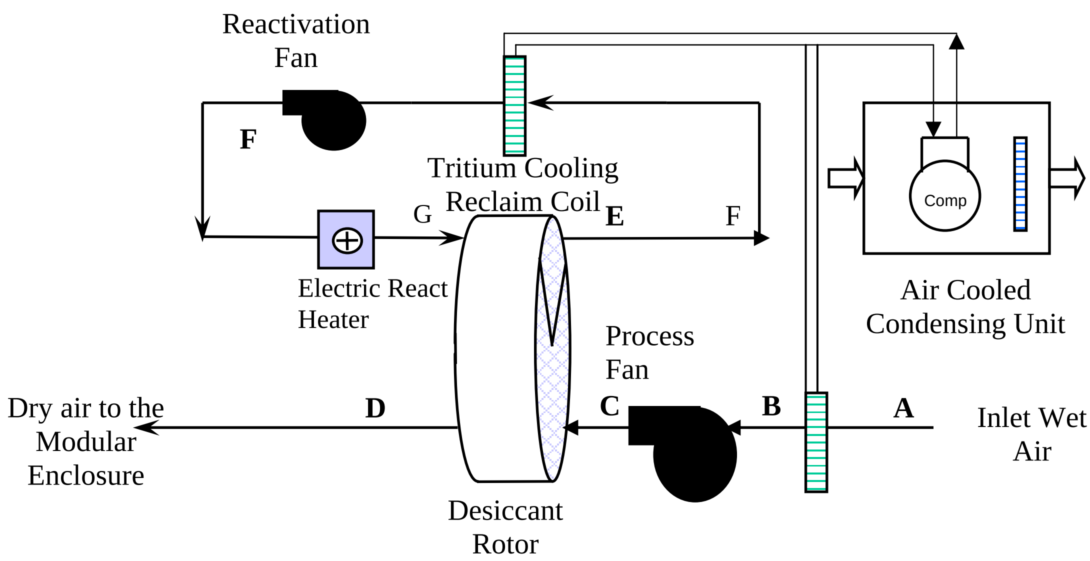
™ PowerPurge is a trademark of Munters Corporation.
[1] HoneyCombe is a registered trademark of Munters Corporation.
[2] Standard Cubic Feet per minute

loop the air stream is heated by an electric heater and fed to the regeneration segment of the rotating desiccant wheel in order to drive the moisture out of the desiccant. The wet air stream passes through a condenser cooled by an integrated chiller unit to recover the moisture. The recovered condensate is collected in a drum. The adsorption flow is counter current to the regeneration flow and the operation is continuous. This maximizes regeneration performance.

The entire VRD system is skid mounted with casters so that it can be easily moved. The dryer unit is also provided with flexible ducting if the dryer cannot be placed close to the maintenance area.

The dryer dimensions are kept to minimum in order to maintain the portable feature. The only service required to operate the dryer unit is electrical power. The dryer unit is a stand-alone unit, completely self-contained minimizing setup time and facilitating ease of operation.

Figure 3 outlines the Point Lepreau VRD flow schematic and outlines the VRD configuration.



Units	A	B	C	D	E	F	G
Airflow - SCFM	550	550	550	550	96	96	96
Temp (F)	80.0	80.0	82.0	91.5	140.0	45.0	300.0
Moisture (gr/lb)	4.40	4.40	4.40	0.140	153.80	44.00	44.00
Dewpoint (C)	-20.00	-20.00	-20.00	-51.10	26.30	7.20	7.20

FIG. 3. Schematic flow diagram of point lepreau portable VRD.

5. Minimize ingress of H_2O in the reactor building

An alternative use of the portable VRD is to reduce the ingress of atmospheric moisture into a reactor vault.

The reactor building ventilation system in the CANDU-6 design has a continuous, once-through ventilation flow of 10 000 scfm of outside air distributed to the accessible areas. This results in considerable ingress of H_2O into the reactor building. Rotary dehumidification units have been used at a CANDU-6 station to minimize the ingress of H_2O. The calculated H_2O seasonal ingress from the outside air for Pickering is as shown in Table-1. The H_2O ingress can be minimized with installation of rotary dryers prior to the inlet of the Air Conditioning Unit (ACU). The rotary dryer design is such that the adsorption and regeneration are done simultaneously and produces dry air continuously at a –

40°C dew-point temperature (0.5 gr/lb). The installation arrangement of the rotary dryers is as shown in Figure 4.

The following benefits are offered with installation of the dehumidification units:

1) Reduction in additional moisture load on the heavy water vapour recovery dryers.
2) The condensate recovered from vapour recovery dryers will have higher D_2O isotopic due to reduction of H_2O ingress.
3) Reduction in the quantity of condensate from vapour recovery dryers
4) The heavy water upgrading cost will be reduced due to high isotopic of the recovered down graded heavy water.

Table 1. Calculated H_2O ingress into Pickering reactor building

Month	Air Temp °C	Relative Humidity% RH	Monthly Average Absolute Humidity (gr/lb)	Moisture H_2O Ingress into reactor building (lb/hr) *(See Note 1)*
January	-4.83	80	14.0	70.2
February	-4.78	79	13.8	69.2
March	0.00	74	19.6	98.2
April	6.28	75	31.1	155.2
May	11.33	62	36.3	182.1
June	17.17	64	55.3	277.3
July	20.17	64	66.6	234.0
August	20.00	70	72.2	234.0
September	16.61	74	61.4	234.0
October	10.50	71	39.4	197.6
November	4.78	80	29.6	148.4
December	-2.00	81	18.1	90.8

Source: Weather data was based for Toronto Island Airport.

Note 1: The inlet air to the reactor Building is Conditioned by the Air Conditioned Unit (ACU-1) to 16°C (60°) DB and 13°C (55°F) and absolute humidity of 56 gr/lb.

Similar calculations were performed using the data obtained from Binmaker PRO for the KANUPP station. The computation used weather data available for Karachi. The total ingress of H_2O was calculated to be 5.1 million pounds/a entering the KANUPP reactor building, based on the ventilation airflow of 10 000 scfm. In comparison with Pickering, the annual ingress at KANUPP is three times higher. Some of this light water ingress will enter into the dried areas of reactor building. This adds additional moisture load on the existing vapour recovery dyers.

The majority of the CANDU-6 stations have dehumidification units installed. The only exceptions are Gentilly-2 and Embalse; the similarly designed Pickering units also lack inlet dryers.

6. Application of the portable dryers at CANDU stations

The following is the brief summary of the possible applications of the portable VRD at CANDU stations:

1) A spill of tritiated heavy water in the heavy water transfer chamber during equipment transfer could result in the elevated MPC(a) levels in the Airlock. A portable VRD is well suited to rapid detritiation of the heavy water transfer chamber atmosphere. In particular, the heavy water transfer chamber usage is most critical during the station maintenance outage period.
2) A portable VRD can be used in the main confinement rooms (moderator pump room).
3) In the event of a spill of tritiated heavy water after successful recovery of the fluid, a portable VRD can be used to detritiate the atmosphere in the area where the spill occurred.
4) A portable VRD can be used to dry the steam generator prior to the start of the chemical cleaning process.
5) At Gentilly 2, a portable VRD will be used to dry paper towels, mops and other items used to clear spill. These items will be placed in the enclosure to remove the tritiated fluid.
6) At Point Lepreau the plan is to remove tritiated fluid from PHT filters prior to disposal
7) At OPG (Nuclear Waste Management Division) Low Level Storage Building #2 (LLSB#2), a portable VRD is used to detritiate the building resulting following tritium outgasing from waste and the building structure.
8) Portable VRDs are used currently on the re-tubing project at Bruce-A Unit #1 and unit #2, and at Point Lepreau to dry the Calandria.
9) At Pickering-A rotary dryers are used to dry the Calandria vault.

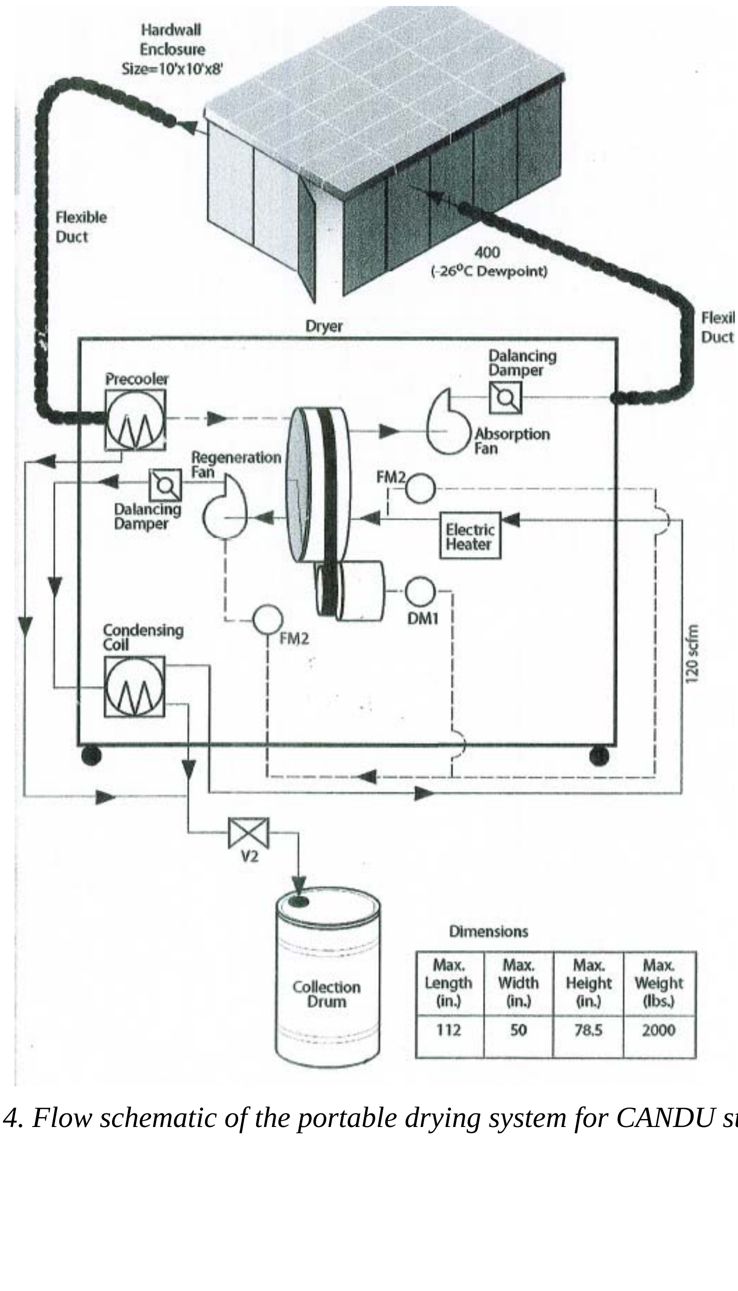


FIG. 4. Flow schematic of the portable drying system for CANDU station.

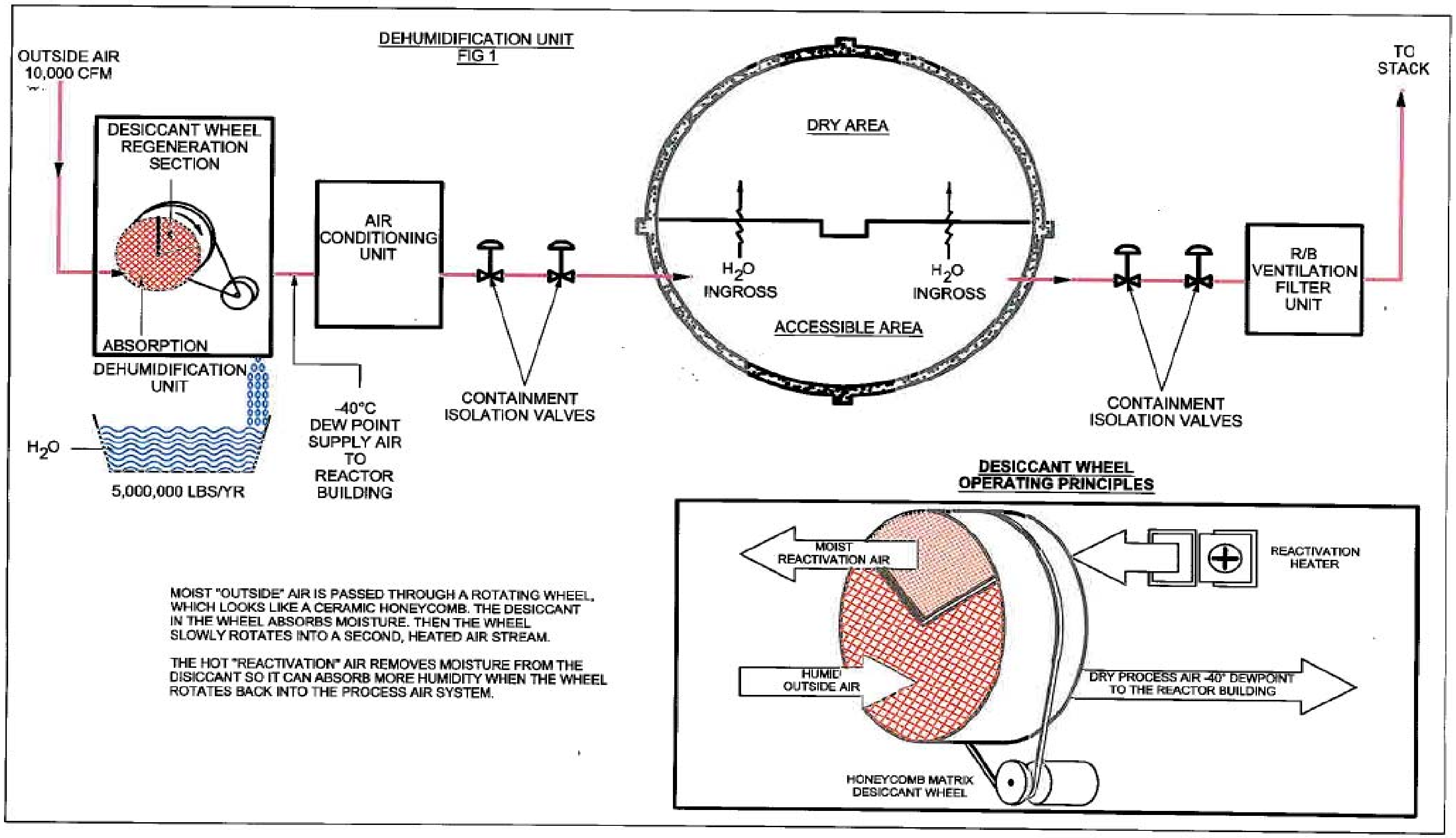


FIG. 5. Flow schematic for reactor building ventilation system.

7. Conclusions

In response to a request from COG members, a portable vapour recovery dryer (VRDs) was developed and tested. The portable VRD is designed to provide effective tritium confinement, as well as tritium emission and dose control for maintenance personnel during station outages and to augment existing vapour recovery dryers in the event of acute leaks or spills.

The portable VRD is very flexible, with numerous configurations available to meet a variety of duties. It is self contained and portable allowing it to be deployed in areas of restricted access.

A prototype VRD was designed, developed and successfully tested. At present there are number of portable VRDs in successful operation at numerous CANDU stations.

Portable VRDs are primarily used to augment the existing vapour recovery dryers. Rotary dryers are not suited as replacements for packed tower dryers that provide dew-point temperatures of $-60^{\circ}C$. These packed type dryers are needed to provide lower MPCa levels in the fuelling machine vaults, moderator room and boiler room.

DOSE REDUCTION INITIATIVES AT DARLINGTON NUCLEAR

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Abstract

In January 2008, Darlington Nuclear was selected by the Information System on Occupational Exposure (ISOE) North American Technical Center to receive the 2007 World Class ALARA Performance Award for setting and achieving high standards in radiation protection. The purpose of this presentation is to share some of the station's successes and challenges in reducing dose to operation and maintenance personnel. In particular, the following dose reduction initiatives will be discussed: 1) innovative shielding design including the use of water shielding walls for on-line airlock maintenance, tungsten shielding blocks for feeder replacement, and exposure reduction for horizontal flux detectors, 2) use of remote monitoring array to scan reactor face for hot spots to reduce dose and save critical path time, and 3) tritium reduction through improvements in human performance, outage scheduling and introduction of new technology. With increasing work scope associated with maintenance, refurbishment and retrofit activities, there is an upward pressure on collective dose making it a critical resource for many inspection and maintenance work groups. There is a need to reduce radiation source terms as a lasting solution to address ever-increasing collective dose. Darlington's long-term strategic goals to reduce tritium and gamma source terms and a bold vision for the future will be discussed.

1. Introduction

Over the past eight years there was an upward pressure on collective dose due to increased workload associated with refurbishment or plant life extension projects. For example, station outage work activities increased from 6000 tasks in 2000 to over 12 000 tasks in 2007 (see Figure 1). With increased workload there is a corresponding increase in collective radiation exposure (CRE), the number of exposed workers and the maximum dose they received. For example, the CRE as an average for all CANDU reactors has increased by more than 60% from 2000 to 2006 (Figure 2). A similar rate of increase in CRE and maximum individual dose was observed at Darlington. With the number of refurbishment projects either on-going or planned, there is a risk that the limited person-rem[1] resources available from the skilled labour pool may not be sufficient to complete the required work. Dose reduction and careful management of person-rem resources become a strategic business requirement.

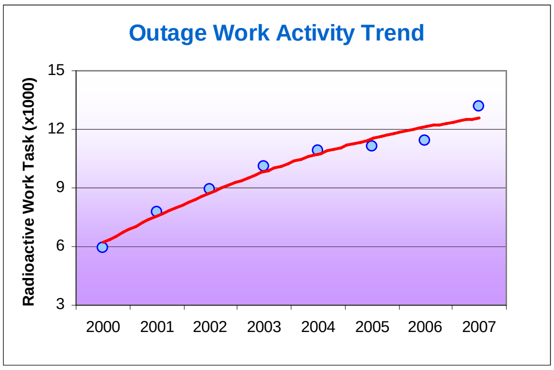


FIG. 1. Upward trend in workload.

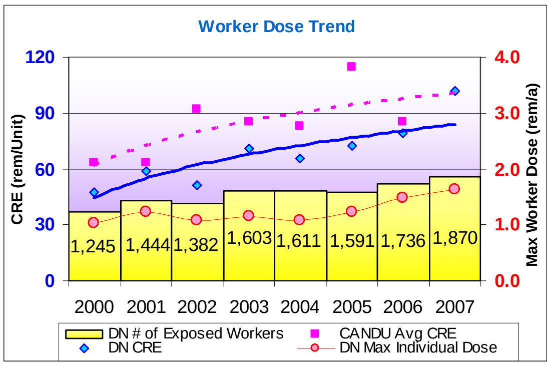


FIG. 2. Upward trend in CRE and individual dose.

[1] 100 rem = 1 Sv.

2. Dose reduction initiatives

2.1. Installation of submicron filtration

There is a need to reduce radiation source terms as a lasting solution to ever-increasing collective dose. Perhaps one of the most cost-effective solutions is to improve the effectiveness of the filtration system. Working with Station Engineering, the ALARA group at Darlington was able to initiate a planned reduction of HT and Moderator filter pore size in steps from 2 µm to 0.45 µm and then 0.1 µm. Significant dose-rate reduction was observed in both Fixed Area Alarming Gamma Monitors (FAAGM) and routine manual surveys. For example, containment dose rates decreased from 18.6 mrem/h in 2002 to 12.5 mrem/h in 2008. Based on the number of person-hours spent inside containment, a total dose saving of 130 person-rem for Unit 1 was estimated (Figure 3)

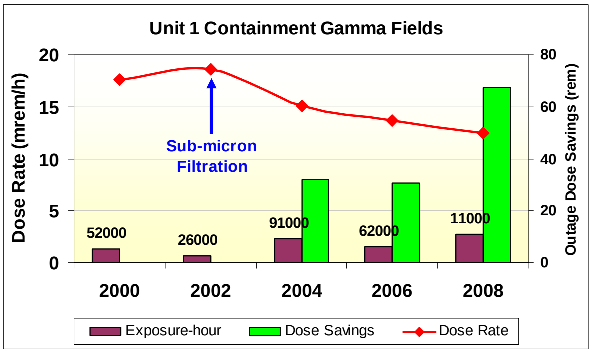


FIG. 3. Unit 1 Exposure trend and estimated dose saving.

2.2. Remote monitoring

With advances in monitoring and wireless communication technology, remote monitoring such as teledosimetry becomes an important tool to reduce worker dose. Some of our current and future applications are described below:

(i) Reactor face Scan — Teledosimetry is routinely used for reactor face scans to minimize the need for manual surveys. Twenty-four EPDs are installed on two 12-foot scaffold tubes attached to the RAB platform. The EPDs are spaced approximately 1 foot apart to align with the channels and cover the full width of the reactor face. fuel handling operators working with Radiation Protection (RP) drive the bridge upwards pausing briefly at each row to allow time for RP to record readings transmitted by the teledosimetry (Figure 4). This method was developed by Darlington ALARA and adopted by many CANDU utilities. The benefits are estimated to be 0.5 person-rem dose and one shift critical path savings per outage.

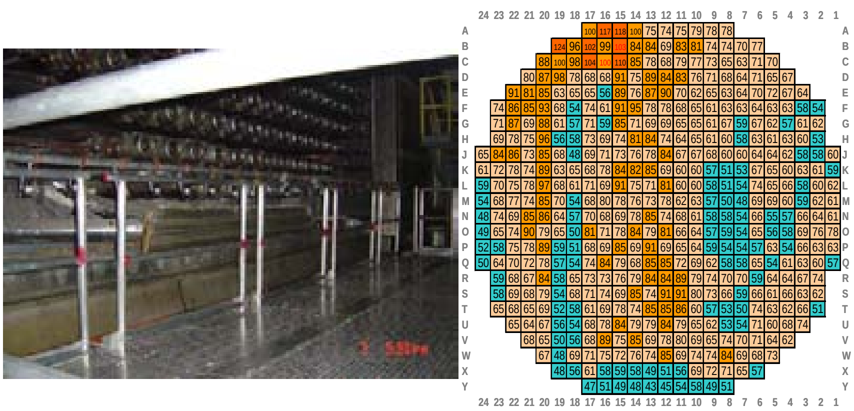


FIG. 4. Reactor face scan using teledosimetry.

(ii) Airlock and waste depot monitoring — Other examples of remote monitoring are remote readout of gamma and tritium levels at airlocks, and monitoring of waste-depot dose rate to prevent un-posted hazards (Figure 5).

FIG. 5. Remote monitoring at the airlocks and waste depot.

2.3. Shielding applications

Temporary shielding is used extensively during IPG and outage work activities and credited with large dose savings. Some of the more unique applications of shielding are described below.

(i) Water wall for on-power airlock EQ (environmental qualification) work — Each wall section consists of 2 units with a combined height of 9 feet. Typically, several sections are installed and filled with water to provide excellent shielding (Figure 6). Three sections of the water wall were installed at our main airlock to allow on-power EQ work to proceed. With a water thickness of 20 inches, our experience showed that the dose-rate reduction was six-fold for gamma and eleven-fold for neutron. The estimated dose savings were eight person-rem for three outages in 2007 and 2008. In addition, a total critical path saving of two weeks was realized.

FIG. 6. Application of water shield wall for dose reduction.

(ii) Water bag for ECI hot spot shielding — High dose rate hot spots of up to 50 rem/h contact and 2 rem/h working distance were encountered at ECI pipes due to crud accumulation. Shielding using the conventional method will incur high personnel dose. A flexible water bag complete with straps and hose was installed at a low dose section of the ECI and moved to the hot spot location. Once it was at the designated location, a small pump was used to fill the bag with water (Figure 7). A nine-fold dose-rate reduction was achieved with one-tenth of installation dose expected when using lead blankets.

FIG. 7. ECI hot spot water bag shielding.

(iii) Reactor Face Shield Block: A second generation shield block was used to reduce gamma fields originating from the reactor face. Unlike the original shield blocks (also developed by Darlington ALARA), the improved version is made of flexible polymer containing tungsten. The blocks measured about one-foot square with a seven-inch hole cut out to reduce weight. The decision to cut out the centre was based on our assessment and actual measurements that the end fittings contribute very little to the radiation fields due to the combined shielding effects of water, shield plug and closure plug. Detailed shielding information is provided below:

- Each block weighs 22 lbs with a 7" hole cut out
- 15 min and 14 mrem to install 35 blocks
- Dose rate reduction of more than two-fold (90 mrem/h before shielding and 42 rem/h after shielding).

FIG. 9. Reactor face shielding blocks.

2.4. *Decontamination*

Strippable Decon Gel paint has been used successfully to decontaminate floor and other surfaces. In one recent application, the floor of HX1 room was coated with the material and peeled the next day (Figure 9). The decontamination results were:

- Before – 5000 cpm
- After – 0 cpm
- Peeling – 4.5 mrem/h C

FIG. 9. Decontamination using strippable paint.

2.5. *Characterization of dose and dose rates*

A great deal of effort was made to understand and characterize radiation fields in a number of exposure environments. Results of the study were use to develop better shielding strategies and techniques:

(i) Reactor face field characterization – At the reactor bridge platform, workers conducting feeder inspection or channel-reconfiguration work are exposed to two roughly equal sources of radiation: one from the reactor face and other from the overhead feeder cabinet. This knowledge allowed us to develop a shielding strategy to protect workers at the reactor face. It consists of a combination of shielding blocks and overhead shielding structure to provide complete protection for workers at the reactor face. Figure 10 shows the configuration of fuel channels and feeders, and the variation of dose rates when moving up and down the reactor face.

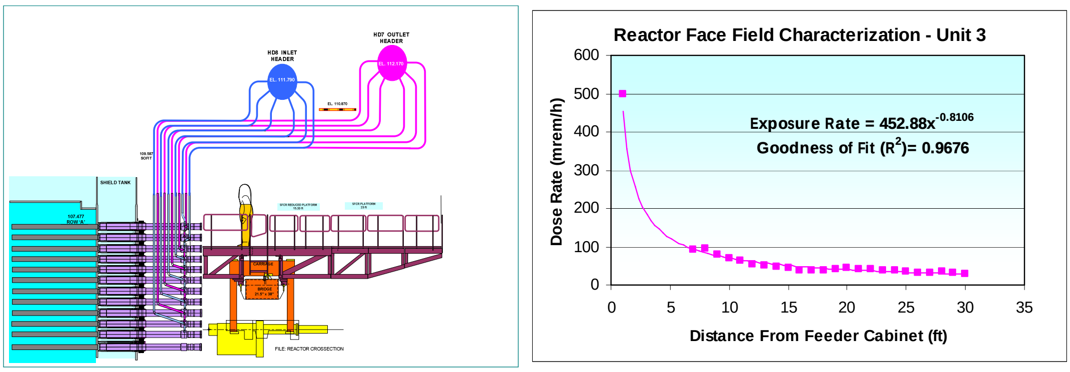
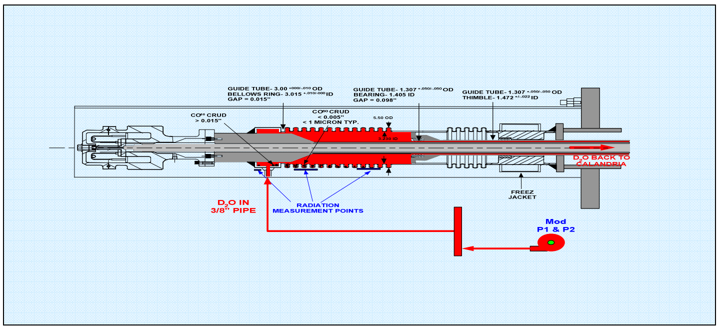


FIG. 10. Reactor face field characterization.

(ii) Ion chamber and HFD cable replacement shielding strategy – very high radiation fields of up to 60 rem/h contact dose rates were observed at the SDS2 bunker during EQ cable replacement work. Detailed analysis of system configuration was made to identify the source and location of high dose rates. Our analysis identified that the D_2O supply line is responsible for crud deposition at the D_2O bellows of the flux detectors. This was confirmed by field measurements and allowed the formulation of shielding strategies to reduce installation dose and improve overall shielding effectiveness. The result is a major reduction in job dose from 37.7 person-rem in 2007 to 18.1 person-rem in 2008 (Figure 11).



EQ Cable	As Found	Shielded	Job Dose
D721	271 mR/h	165 mR/h	37.7 rem
D811	280 mR/h	68 mR/h	18.1 rem

FIG. 11. Ion chamber and HFD shielding.

2.6. Internal dose reduction

(i) Source Term Characterization – Similar to external dose reduction discussed above, our initial efforts were focused to understand and characterize factors influencing tritium levels inside containment. Results of our study indicate that higher tritium concentrations occurred when PHT is full and depressurized or refilled to the gravity fill state (GFS). This created a pressure window where closure plugs are leaking the most. Further study indicated that the closure plugs started to leak at 2.3 MPa and stopped when the PHT is at the very low-level drain state (VLLDS). Vault tritium is also elevated during ice-plug work when the vault vapour recovery system is reconfigured in N_2 purge mode. Analysis of tritium uptake patterns showed that 70% of station annual internal dose was received during outages, with the balance of 30% during normal operations. In a typical outage (e.g. D631) a staggering 90% of outage tritium dose was received during a twelve-day period when PHT is either full or in GFS (Figure 12). Detailed information on tritium concentration and uptake patterns allows us to focus our efforts and resources to minimize internal dose.

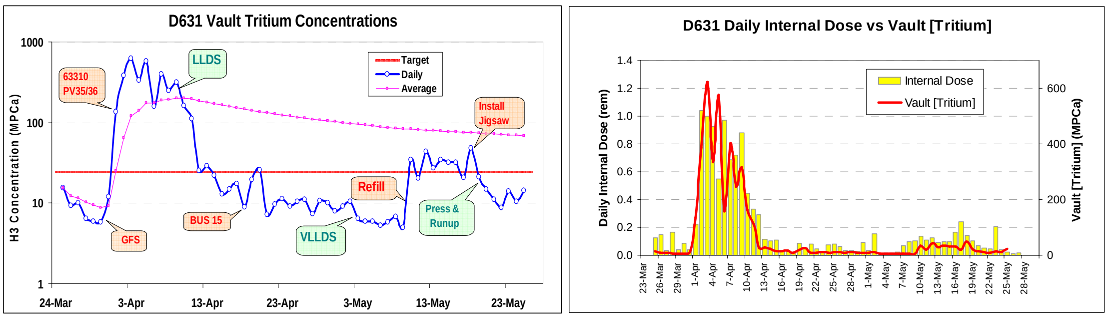


FIG. 12. Vault MPCa and tritium uptake.

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(ii) Tritium Reduction ALARA Plan – A comprehensive outage ALARA plan was prepared to control tritium exposures. Key programme elements include:

- Action levels established with pre-determined corrective actions
- Closure plug leakage mitigation (closure plug tightening, PHT low level drain state to reduce static pressure)
- Portable dryers installation and maintenance (D811 result: achieved a 96% capacity factor, 24 drums and 1200 Ci extracted from vault air)
- Tritium reduction and control at airlocks (Figure 13).

Significant dose reduction was achieved during our spring outage in 2008 (D811), in which tritium concentrations were reduced from 500 MPCa to less than 10 MPCa and tritium dose accounted for only 6% of total dose (Figure 14).

On-going improvements in controlling tritium source term have contributed to a continual downward trend in internal dose per task as well as the ratio of internal to total dose (Figure 15).

FIG. 13. Use of Munter dryer and plastic curtain to reduce and contain tritium.

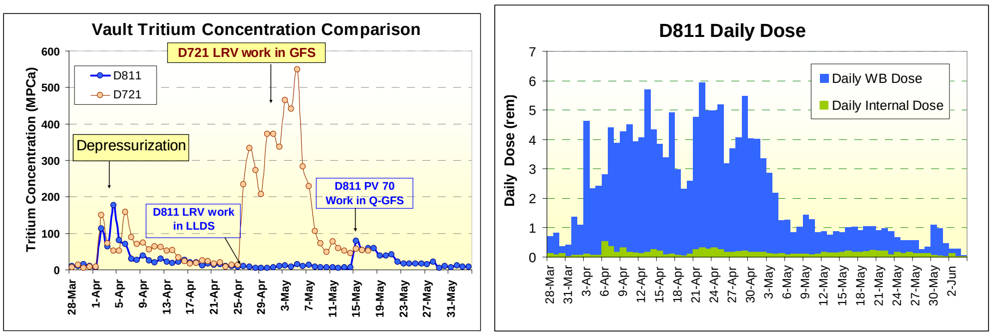

FIG. 14. D811 outage tritium result.

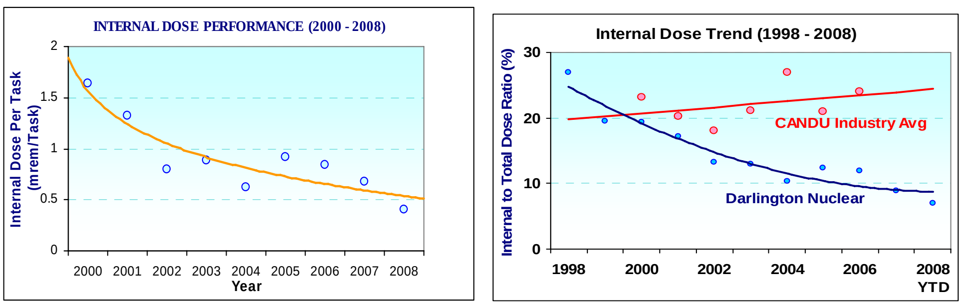

FIG. 15. Internal dose performance trend.

3. Human performance improvement

3.1. *Impact of worker practice*

Human performance has a large impact on the effectiveness of radiation protective equipment (RPPE). Our experience indicates that after a very focused communication campaign that included the use of video, poster and presentations at safety meeting topics, to increase worker awareness about the importance of remaining plugged-in to air headers, the plastic suit protection factor (PF) increased by a factor of 2. The same study also showed that the impact could decrease over time unless a new communication campaign is implemented. Obviously there is a need to re-invent the message to maintain its effectiveness.

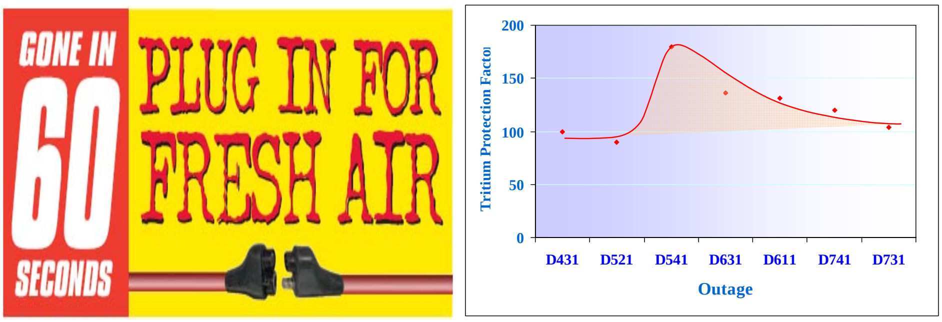

FIG. 16. The impact of worker communication programme on protection factor.

3.2. *Coaching, monitoring & follow-up*

Daily worker EPD dose and tritium uptake are monitored and unusual occurrences are flagged for follow up. Action levels are established to ensure consistent approach:

- 3 µCi/L or 10 mrem EPD − follow-up with worker/supervisor,
- 10 µCi/L above planned or EPD alarm − department EFDR, SCR,
- 35 µCi/L above planned or 100 mrem above EPD alarm − root cause investigation.

In conjunction with the action levels, focused observation and coaching (O&C) are used by RP staff to improve human performance. Performance management includes interview by RP management for significant RP infractions or repeat events.

4. Future challenges

Two ambitious goals were established to reduce internal and total worker dose:

1) Reduce Tritium in containment to < 1 MPCa by 2011

   Action plans:

   - Redesign closure plugs to be leak tight throughout the outage
   - Create alternate venting path to preserve the vault vapour recovery system (VVRS) during ice plug work
   - Install portable dryers to supplement VVRS

   Benefits:

   - Eliminate plastic suits for most outage work
   - Reduce outage duration by 2-5 days
   - Reduce worker dose (>20 rem/outage) and tritium emission

2) Reduce γ dose rate by 25% by 2011

Gamma scans identified Co-60 as the dominant source of radiation responsible for 75-80% of total exposure. Studies showed that more than 80% of Co-60 originates from fuelling machine Stellite ram balls. Corrective action plans include both long-term and short-term actions:

- Long term plan: COG project to develop replacement strategy,
- Short term plan: fuelling machine filtration improvement − submicron filters and ion exchange.

The following graph (Figure 17) shows that there is a gradual reduction of containment dose rates after installation of submicron filters. The same graph also shows that the reduction may be levelling off as the rate of source term removal is substantially equal to the rate of addition. Actions intended to reduce the addition of source term into PHT is necessary to continue the downward trend.

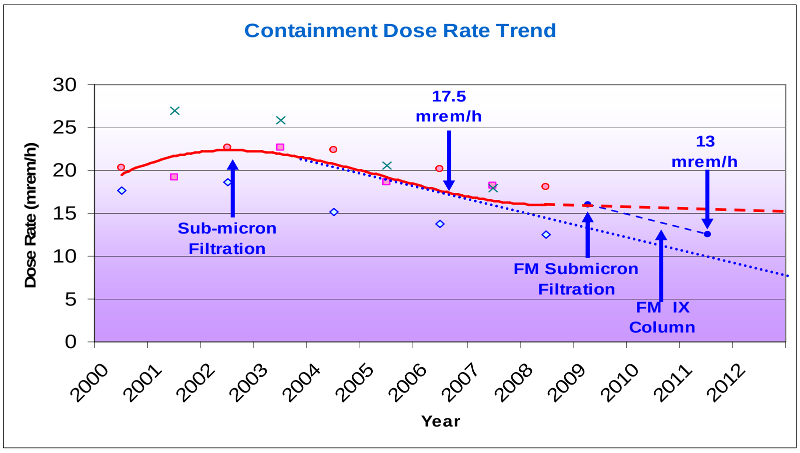

FIG. 17. Containment gamma dose rate trend.

5. ALARA achievements and recognition

The station's efforts in dose reduction were recognized by WANO. From 2001 to 2007, three consecutive WANO Strengths in ALARA were identified. For example, the most recent WANO evaluation (2007) identified areas of strengths in ALARA:

- High standards in RP have been set and achieved
- Robust controls prevent unplanned exposures
- Extensive ALARA planning, innovative shielding and aggressive tritium reduction.

In 2007, Darlington was given an 'A' rating in RP programme implementation by the CNSC and was the proud recipient of ISOE 2007 World Class ALARA Performance Award.

SESSION III. PERFORMANCE IMPROVEMENTS

USE OF HUMAN PERFORMANCE FIELD SIMULATOR AT PICKERING NGS, *Practising Event Prevention – Driving Culture Change*

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Abstract

The use of event free tools such as pre-job briefing and post job debriefing, conservative decision making, questioning attitude, procedure use and adherence, three-way communications and self check, had been widely taught in OPG. While staff were very familiar with the tools, OPG was also seeking ways of ensuring that staff not only knew the event free tools, but also could practise them and be supported when they did. Observing actual behaviour and coaching it to meet expectations without threat of failure or consequence is driving improved performance and achieving the desired culture change.

1. Background

The following pages provide an overview of the way that employees at Ontario Power Generation Pickering Nuclear Generating station planned and established an area and process to practise human performance tools.

It was recognized in 2002/2003 that OPG needed to improve human performance to reduce human errors and eliminate consequential events. The strategic focus coincided with advances being made in other areas of the business such as improved personnel safety, reduced radiological doses, and increasing productivity. The use of event free tools (pre-job briefing and post job debriefing, conservative decision making, questioning attitude, procedure use and adherence, three-way communications and self check) had been widely taught by this time in OPG. While staff were very familiar with the tools, OPG Nuclear (OPGN) was also seeking ways of ensuring that staff not only knew the event free tools, but also could practise them and be supported when they did.

Communication of the tools was strong and they were widely accepted. The next step was to provide a means of having staff demonstrate them in an environment that could tolerate a mistake. Pickering staff reviewed some internal and external examples of human performance simulators and error labs. These were used as examples to influence the Pickering project.

While the examples were of great help, each site has a very unique culture of its own. What might work at one site (games for example) may not work at others. A static error lab was discounted as a stand-alone because the group noted that it did not drive interaction and could be used just as an observation tool. In addition, the team at Pickering wanted to add an extra dimension. The process established was that the simulator would be used as much as possible by supervised work teams conducting tasks that they might be called upon to perform at any time. This ensured that the learning was focussed through the organization; e.g. a supervisor can receive coaching on their supervisory or feedback habits in real time, as well as their crew receiving feedback on the way they performed the task. Both groups can receive feedback on the way they interact with each other.

The facility at the Pickering site is a decommissioned electric boiler steam heating system. The entire process system is in place but is now physically isolated from energy sources (blanks in pipework and electrically de-energized). Since it is in the Pickering A powerhouse, it provides a realistic backdrop to the assigned tasks (i.e. subject to powerhouse temperature conditions; PA announcements; traffic, etc).

Establishing scenarios and then finding trainers who were willing to take on a substantially different role than regular on-the-job training could have been difficult, but the site responded with staff who were very committed and their enthusiasm overcame any reluctance by other staff to participate in the scenarios. The trainers' role is evolving and they are truly key to the success of the initiative. The progression through the scenario can take many branches or diversions if the trainer sees an opportunity for a learning space.

The critical aspect of this is that the spotlight is entirely on how the individuals perform – not how they tell you they would perform, nor even how we expect them to perform. Observing actual behaviour and coaching it to meet expectations without threat of failure or consequence is driving improved performance and achieving the desired culture change.

2. Objective

- Improve human performance by providing an error-tolerant environment where already qualified Trade staff can practice the application of the Event Prevention Framework;
- Identify and practise desired behaviours during real-life situations;
- Practise as a supervised work team − improve procedure Use and Adherence, Pre-Job Briefing/Post-Job Debriefing (PJB/PJD), supervisory skills, coaching skills and teamwork;
- Drive observation skills and reinforce higher material-condition expectations;
- Additional scenarios are added as negative human performance trends emerge.

3. Facility & process

The human performance field simulator located in a realistic working environment (non-classroom). It comprises a decommissioned heating steam system, and spans three elevations in the Pickering A plant on Unit 1. Pickering A/Pickering B staff sharing one training process. The focus is *always* on the *displayed* versus the *required* behaviours (removes role playing).

Advantages:

- Low risk to plant and personnel (decommissioned system)
- Isolated from other plant systems
- Distinctive identification in place (area outlined by floor painting)
- Real work packages
- On-the-Job-Training, not On-the-Job-Evaluation; i.e. no pass or fail
- Real work assignments displaying actual behaviours (seeing what the person does; i.e. actions, rather than what they should do provides significant improvement opportunities)
- Focus on supervised work teams (requiring PJB/PJD; coaching to best practice, etc)
- Enhance working relationships with peers, and helps to develop supervisors
- Work group specific scenarios
- Keep staff current with changing procedures and standards
- Identify gaps in knowledge that could negatively impact on human reliability
- Field First Line Manager/First Line Manager's assistant are Trainers (qualified/mentored)
- Self administered.

4. Programme

The plan had specific phases recognizing that changing human behaviours is a continuous process.

Establish the programme

- Training Qualifications established (supervisory and non-supervisory qualifications)
- At least one scenario per work group
- Straightforward scenarios – no 'tricks' although error traps exist
- At least one trainer prepared per trade
- Supervised work teams (driving improvements in Pre Job Briefing, debriefings, Procedural Adherence)

Establish the process; drive discussion including use of self check simulator

- Concentrate on aspects of behaviour: attitude/awareness/fitness
- Conduct pre-job briefings
- Understand and rigorously comply with procedures
- Monitor important operating parameters
- Perform independent verification
- Communicate vital information, reports/station condition records
- Use of logs/records/changes
- Train others on the job (peer to peer)
- Perform turnovers
- Capture Lessons Learned (post job debrief)

Improve the process

- More challenging scenarios based upon analysis of HP related trends
- Implement performance simulator for supervisors and managers — scenarios based on identifying deviations from standards.
- Integrate a static error lab
- Integration with the human performance working committee
- More line ownership

- More integration of workgroups − driving improved turnovers and handoffs

- More use of precursor information (low level events and trending) to drive scenarios and prevent events.

5. Summary

This is not skills training, but reviewing actual on the job performance of supervised work teams. Performance correction, coaching and recognition takes place on the spot. Each major trade group is looking at trends/behaviours to develop scenarios to practise and eliminate errors.

PLGS LEVEL 2 PSA

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Abstract

NB Power Nuclear has developed a PSA that incorporates a high degree of operational information and site-specific data to evaluate risk metrics based on the way the plant is actually operated and maintained. Unique characteristics of the PSA prepared for Point Lepreau Generating Station was presented. The high degree of model depth or resolution has enabled quick turn-around time in response to operational queries and the ability to quickly evaluate the impact of any system design modification.

1. Introduction

Point Lepreau Generating Station (PLGS) has completed a full-scope Level 2 probabilistic safety assessment (PSA) to assess the risks associated with the post-refurbishment configuration of the plant. The PSA includes internal events, and external events such as internal fires, floods and a PSA-based seismic margin assessment (SMA), to evaluate the risks associated with severe core damage and large releases from full-power operation. Shutdown operation has also been included for internal events only. The PLGS PSA was developed with a very high degree of resolution (i.e. large models) largely incorporating pure site-specific component reliability data; integration of a significant amount of operational input affecting model structure to reduce conservatisms; determine realistic fault exposure times; and determine expected plant response from a desktop simulator. Most mitigating systems with their support functions have been modelled as well and are fully integrated into a single cohesive master fault tree with multiple tops. This paper discusses these various aspects, the benefits they provide and the next steps to align plant operation, programmes and processes with the PSA.

2. Safety goals

The safety goals for severe core damage frequency (SCDF) and large release frequency (LRF) defined for PLGS are derived from those typically expected of refurbished nuclear power plants. They are:

	Safety Goal	**Safety Limit**
SCDF (/yr)	1E-05	1E-04
LRF (/yr)	1E-06	1E-05
Seismic (HCLPF)	>0.3g for SCDF (level 1) >0.4g for LRF (level 2)	

Note that the PSA-based seismic-margin-assessment (SMA) results are not provided as an expression of SCDF or LRF, but are rather expressed as a measure of seismic capacity in terms of High Confidence Low Probability of Failure (HCLPF). The PSA-based SMA is for all intents and purposes a Seismic PSA, but without the seismic hazard curve convolved into the analysis due to the high degree of uncertainty it would introduce into the results.

3. Overall PSA results

The estimated severe core damage frequencies for at-power and shutdown events are provided below for both the Level 1 PSA and Level 2 PSA[1].

[1] Canadian regulatory review of the PLGS PSA is on-going and therefore results are subject to change.

Level 1 PSA results

	Severe Core Damage Frequency During Full Power (events/year)	Severe Core Damage Frequency During Shutdown with HTS Full and Depressurised (events/year)	Severe Core Damage Frequency During Shutdown with HTS Drained to Header Level (events/year)
Internal Events	1.32E-05	8.15E-06	1.35E-07
Internal Flood	1.15E-06	N/A	N/A
Internal Fire	2.59E-05	N/A	N/A
Total	4.03E-05	8.15E-06	1.35E-07

Level 2 PSA results

	Large Release Frequency During Full Power (events/year)	Large Release Frequency During Shutdown with HTS Full and Depressurised (events/year)	Large Release Frequency During Shutdown with HTS Drained to Header Level (events/year)
Internal Events	9.22E-08	2.19E-07	2.09E-09
Internal Flood	4.20E-09	N/A	N/A
Internal Fire	5.05E-07	N/A	N/A
Total	6.01E-07	2.19E-07	2.09E-09

PSA-based seismic margin assessment results

	PSA-based Seismic Margin Assessment
Severe Core Damage	0.30 g
External Releases	0.42 g

As shown, the LRF estimates for all cases not only meet the Safety Limit, but also meet the Safety Goal following consideration of a variety of design upgrades to protect the containment structure during severe accidents. The SCDF estimate during shutdown operation also meets the Safety Goal. However, the SCDF estimate during full-power operation is between the Safety Goal and Limit.

4. Mitigating system fault tree development

Like all other PSAs developed around the world, the PLGS PSA mitigating system fault trees were developed to a resolution that is supported by available component reliability data. However, the PSA for Point Lepreau appears to be much larger than most others based on its scope, and the resolution of its fault tree and event tree models. To put this into context:

	TOTAL	INTERNAL
Basic Events	55 332	55 332
Event Trees	449	172
Fault Tree Tops	895	895
Sequences	23 894	6 870

The number of basic events and fault-tree tops are the same because the fire PSA, flooding PSA and internal-events PSA are fully integrated and the results for each element of the PSA are evaluated from the same model.

The above data would indicate that the PLGS PSA is much larger than a comparable PSA in the United States of America. Why so large? There are five answers to this question.

1) The PLGS PSA explicitly models *all* mitigating systems using fault trees except for two, as opposed to including only an undeveloped event that represents the entire system. The benefit of this approach is that the probability of system failure can be quantified and compared against expected results. If results are not as expected, dominant failure modes can be examined for vulnerabilities at the system level using importance measures and improvements to the systems identified where there is the largest impact in terms of risk reduction. Improvements considered included validity of the component failure model, accuracy of failure rates, reduction of conservatisms in the fault tree model structure and assumptions, procedural or maintenance changes, and as a last resort, design modification.

2) For renewal of the Power Reactor Operating Licence for PLGS in 1992, the Canadian Nuclear Safety Commission (then the Atomic Energy Control Board) stipulated that PLGS must develop the capability to generate site-specific component failure rates and apply them to all system fault-tree models. At that time a decision needed to be made in terms of the level at which the failure data would be collected—what should the data represent? A decision was made that the data should be collected at the first replaceable component by maintenance staff so that the data could provide information regarding the effectiveness of the maintenance programme. For example, if an alarm unit fails and the maintainer simply replaces the whole unit, then the component boundary for data collection was set at the alarm unit level, and it is treated like a black box without consideration of the components internal to the alarm unit. As a result of this approach, fault tree analysis at PLGS also extends down to the first replaceable component because that is the level at which the site-specific data is collected.

   However, during fault tree development at PLGS it was also realized that sometimes relay contacts or alarm-unit contacts within the first replaceable component could have different functional failure effects that lead into different parts of a fault tree or could be tested using different surveillance procedures. As a result, the models go a bit deeper by also modelling relay and alarm unit contacts as well.

3) PSAs intended to provide insights into design vulnerabilities typically do not need to model duty cycling modes of configuration for redundant equipment. Numerically, evaluated risk metrics are accurate even when assuming a single configuration for the redundant equipment. Such modelling detail is normally applied to the model for an on-line risk monitor. However, during model development, it was decided to model the full duty cycling capability of redundant equipment, which adds to the complexity of the model. This complexity was added to the base PSA models due to the vision of how these models would be used to support operation of the station should a plant evolution be more complex than future on-line risk monitor could handle.

4) The electrical distribution system (EDS) for the plant was fully modelled and integrated so that the likelihood of failure for each load as a result of panel, motor control centre or bus failure could be understood. Given that the EDS can be configured in many different ways depending on various plant states, models were developed for 11 plant configurations at 120 equipment loads. These models were then integrated with the front-line mitigating system models.

5) Mission modelling within the PSA was handled in a unique fashion. When dealing with redundancy and a mixture of components being poised and running in a mitigation function, modelling guidelines were defined and followed in order to truly reflect the state of the

equipment prior to the initiating event, the availability of the component being known or unknown at the time of the initiating event and the degree of redundancy which needs to be accounted for during the dormant and mission periods.

The benefits of such a modelling approach are:

- the mitigating system models can be fully integrated, and all dependencies from support systems and their components can be readily identified

- surveillance frequencies can be modified with little uncertainty involved. If a mandatory test or surveillance frequency is modified, only the model needs to be quantified to evaluate the impact without any additional effort or specialized mathematics that caters to those components that might not be in the model

- As no additional modelling is required, any configuration-based query from operations staff can be quickly assessed. If the turn-around time to respond to such queries were long due to reduced PSA quality, the PSA would rapidly lose value in the view of operations and maintenance

The detriments are:

- the models can be rather large and unwieldy in terms of maintaining them

- it poses a challenge to computer resources for PSA accident sequence quantification

- the models will require simplification for use in on-line risk monitoring applications. They cannot be used directly as the quantification times are too long to adequately support day to day control room operations and work clearances

Despite the challenges associated with managing such large mitigating system fault tree models, it is considered that the benefits far outweigh the detriments.

5. Linkage to plant processes

During the development of the PSA models, it was considered essential that the PSA reflect the way that business is actually carried out at PLGS. Therefore, data included in the basic event database used by CAFTA must reflect our plant processes. For the time-based component failure models, the exposure time to a critical failure mode is determined based on three measures:

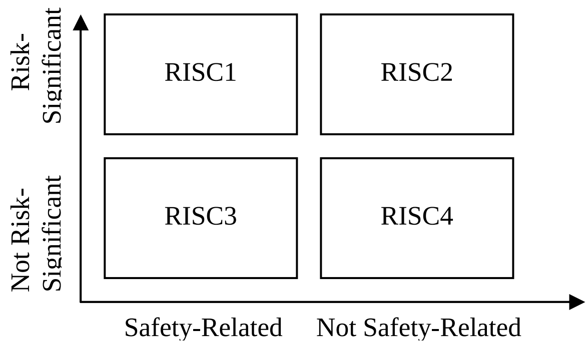
Exposure Time = Detection Time + Access Time + Repair Time

Many utilities represent access time and repair time as a single entity—repair time. PLGS splits this up into two terms for the reasons discussed below.

Detection Time = one-half of the interval (in hours) for the surveillance and/or test procedure that detects the critical failure mode. The procedures are considered part of a plant mandatory surveillance programme, which stipulates strict requirements for scheduling and performing such tests within prescribed scheduling tolerances.

Access Time = that time (in hours) from detection of the failure until maintenance staff are able to place their tools on the equipment for repair. This value is established from a decision chart that takes into consideration accessibility of the equipment while the reactor is at power, the availability of spares and the priority of work. As part of a sixteen-week forward scheduling process, the priority of work is determined based on a prioritization matrix provided in the work management procedure in *Station Instruction SI-01365-P90.*

It was recognized during PSA development, that *SI-01365-P90* does not provide guidance on how to determine whether or not a component is risk significant so that it can be assigned an appropriate priority of repair. The PSA can determine risk significance from a safety perspective utilizing criteria from the Electric Power Research Institute (EPRI) for importance measures that can be directly calculated from the PSA. To do this, a modification to the Work Management process is being progressed to reflect risk informed safety categorizations (RISC) as follows.



As PLGS uses SAP™ for its Work Management software, this is done through the use of an ABC indicator. That indicator is being modified to

- A – Special Safety System (risk significant)
- B – RISC1 or RISC2 (risk significant)
- C – RISC3 (not risk significant)
- N – Not Applicable/Not Credited in PSA (i.e. RISC4)

SI-01365-P90 is being revised to ensure that the ABC indicator is referenced as part of the process to determine risk significance for the prioritization matrix.

As part of the access time calculation, a review of inventory was also performed to determine the average procurement time if equipment spares are normally not maintained at site. The procurement time was heavily dependent on whether or not the equipment was commercial grade or nuclear grade. If the PSA identified that particular equipment was risk dominant, and the cause was determined to be a lack of spares inventory, then recommendations were made to procure sufficient spares to reduce the access time contribution.

Repair Time = the hands-on wrench time for the maintainer. The average repair time for various component types is determined directly from maintenance records for the equipment.

The Risk & Reliability (R&R) Group at PLGS is responsible for maintaining the PSA and protecting the integrity of its models. In that regard, all procedures or deterministic analysis credited in the PSA are to be reviewed by the R&R Group for frequency and content changes. If the impact on the PSA is unacceptable, modifications to the procedure or analysis are required. In the PLGS PSA model, one column is provided within the basic event database labelled SOURCE for the sole purpose of identifying the alarm, surveillance procedure or maintenance plans that detects the critical failure mode. In this way, when the frequency of a procedure is modified, all affected basic events can be easily identified and adjusted for evaluation. To ensure that all credited procedures and analyses in the PSA are easily identifiable, an information report *IR-01500-10* is being prepared, which can then be referenced by a modified Documents and Records process to determine when a document must be submitted to the R&R Group for review.

6. Use of site-specific data

As previously mentioned, the Canadian Nuclear Safety Commission (CNSC) required the use of site-specific data at PLGS in 1992. While the typical approach applied in the industry is to blend site-specific data with generic data sources using Bayesian combination techniques to gain as much statistical accuracy as possible, PLGS adopted a different approach. It was decided that as much pure site-specific data should be applied to PLGS models without combining with generic data sources so that the probability of system failure and overall plant risk metrics of SCDF and LRF would represent the way that PLGS is operated and maintained as closely as possible.

However, PLGS is a single-unit station and concern existed that although the plant has operated since 1982, there might not be enough experience to justify the use of site-specific failure rate data alone. To ensure this was not the case, an adaptation of Sequential Test Plan 9D was adopted from *MIL-HDBK-781A* to screen the site-specific data against published generic data sources. This ensures that the PLGS site-specific data has at least four times the experience of the published data source if the observed number of site-specific failures is zero. If the site experience is insufficient, then the site-specific failure rate is blended with a generic data source using Bayesian combination. This approach has resulted in about 85.5% of the data used in the PSA being purely site-specific, about 13% being blended with generic data, and 1.5% solely from generic data sources.

7. Fault tree multiple top approach

The EPRI Risk & Reliability Workstation (or CAFTA) allows users to link mitigating system fault trees to event trees at either the header level or at individual sequence failure branches. During development of the mitigating system fault trees, direction was provided from NBPN to contractors to use the latter approach and thus avoid using flags to turn off and on initiating events within the mitigating system fault trees. The reasons for this were:

- To allow future integration of all sequences within a single quantification file, so that the PSA can be quantified or evaluated in a single run using the code PRAQUANT or so that multiple initiating events can be run within a single quantification file.
- To allow full integration of all sequences leading to SCDF into a single comprehensive fault tree, which would significantly simplify technical assessments for less-experienced staff without having to worry about the state of initiating event flags.

The detriment for this approach is that it added complexity to the models and made them larger still, and it also made the event trees larger, which adds to quantification time for the full PSA and effort to maintain the models.

8. Model integration

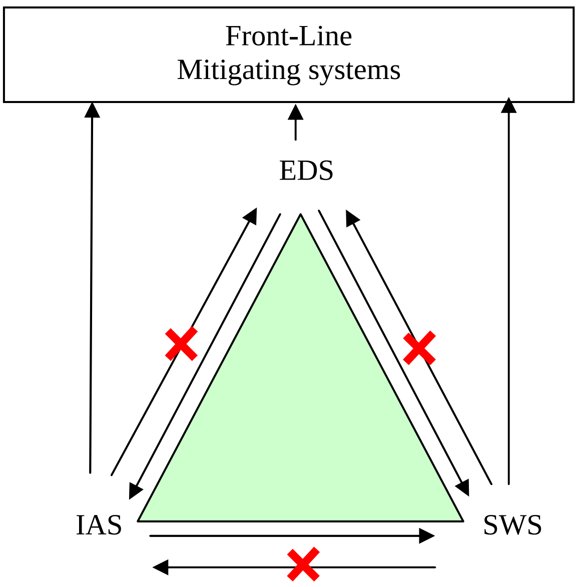
To reduce the burden on PSA resources at PLGS, it was always envisioned that PLGS would maintain as few models as possible, and therefore, the way that the PSA was carried out and models integrated between internal and external events was considered very important. In fact, under the site Reliability Programme, once the PSA modelling for internal events was essentially completed, the unavailability models[1] were retired in favour of using the PSA models to meet Canadian Regulatory Standard S-98 requirements.

The integrated PSA model is labelled ASQ_MASTER, and it includes all events necessary to quantify internal events PSA, flooding PSA and fire PSA for both Level 1 and Level 2. Having used a multiple top event approach, changing the model in one place can affect multiple tops and possibly internal and external events. This approach, therefore, leads to less effort in maintaining the models and making adjustments as they typically only need to be made in one place in the fault tree….not in three separate PSA master fault tree models where configuration control could become problematic.

[1] Unavailability models at PLGS were considered separate from the PSA and did not include contributions from common cause failures or human reliability analysis.

Only the PSA-based seismic margin assessment (SMA) models are not integrated appropriately with the ASQ_MASTER. This was largely due to time constraints associated with the PSA development. Seismic fragility information for equipment and surrounding structures was entered as separate seismic events within a duplicate copy of the ASQ_MASTER, resulting in a separate seismic fault tree. To enable integration and ensure that PLGS is only dealing with a single model, a concept has been developed where the seismic capacities for the individual equipment can be modified to reflect the surrounding structures and the individual seismic basic events removed. Planning is in progress to perform these model upgrades, and it will ensure that PLGS will only need to update a single PSA master fault tree whenever operational, maintenance or design changes are implemented at site.

The methodology for integrating support systems (Instrument Air System – IAS, Electrical Distribution System – EDS and Service Water System – SWS) with the front-line mitigating systems was important to ensure integrity of the model results while eliminating circular logic since support systems also support each other. The following shows an overview of how the support systems fed into each other and the front-line mitigating systems. An ‘X’ denotes where a direction of support was not allowed to eliminate the circular logic.



Since only one flow path was allowed to be selected in order to break-up the circular logic, the integration flow was chosen as such due to the fact that the EDS contribution towards IAS and SWS was deemed to be more dominant than the inverse. Furthermore, the SWS contribution towards IAS was judged not to be as dominant as the opposite since the cooling supply to the IA compressors has additional inherent redundancy.

The Level 1 and Level 2 PSA have also been integrated. When performing the Level 2 PSA, the top Level 1 plant damage sequences, which account for most of the total severe core damage, were grouped based on the similar plant configuration. For example, for internal events 150 sequences from Level 1 sequences representing 99.7% of the severe core damage were used under the Level 2 analysis. The dominant Level 1 sequences were then grouped based on similar accident progression and end state conditions and analysed under MAAP-4-CANDU. Accident sequence event trees were then developed taking into account the following:

a) The status of the mitigating systems which can still provide a heat sink to the non-damaged and damaged fuel channels,
b) The impact on the containment system availability and the ability of the containment envelope to box-up,
c) The hydrogen mitigation requirement,

d) The availability of the systems providing a first line of protection against Containment pressurization,
e) The requirement for a last line of protection against Containment pressurization.

The Level 2 accident sequence quantification (ASQ) was then performed by integrating the respective Level 1 sequences as the initiator for the Level 2 accident progression event trees and by using the integrated ASQ_MASTER fault tree file, which includes the Level 1 mitigation and support functions as well as the Containment and Level 2 mitigation functions.

9. Plant response simulation

One of the most challenging aspects pertaining to development of a PSA is determining how the plant will respond under beyond design basis conditions. Deterministic safety analyses, although tending to be rather conservative, provides plant response information for a variety of design basis accidents, however, the PSA deals with many more scenarios than are covered under the deterministic single/dual failure approach. Expert solicitation from licensed operations staff and engineering judgment is one approach but can be somewhat difficult to defend. This approach was taken at PLGS, however it was also supplemented by simulations using both a full mock-up Main Control Room (MCR) simulator and a desktop plant simulator that provides equivalent results as the large MCR simulator. The simulators allowed PSA staff to simulate various failure scenarios to determine behaviour of various systems, timing of alarms and events, which assisted both with event tree development and determination of Human Reliability Analysis diagnosis and execution times.

10. Treatment of uncertainty (error factor)

Parametric uncertainty in the PSA was assessed for each basic event within the ASQ_MASTER fault tree and quantified using the programme UNCERT version 2.1, developed by Nuclenor, Iberdrola and Data Systems & Solutions. Since the ASQ_MASTER is shared between all elements of the PSA with the exception of the seismic margin assessment, parametric uncertainty was performed on the internal events, fire and flood PSA. Basic events for which their probability of failure is calculated from a site-specific failure rate were assigned an error factor based on the statistical confidence of the site-specific failure rate data itself as opposed to making a general assumption. This ensures that the overall parametric uncertainty reflects actual historical experience as much as possible. For those basic events that have probability emanating from other sources, a unique error factor was provided based on the source of the data.

11. Fire PSA – cable routing

Cable and conductor routings from the power source, control or instrumentation to each end device for all devices that are considered to be of importance with respect to the PSA was established principally from the Device Installation and CONnection (DICON) and Integrated Electrical and Control (IntEC) databases in use at PLGS. This information was substantiated by the Fire Hazard Assessment (FHA) tables created by Professional Loss Control (PLC), various other sources of information available, as well as walk-down of the plant. The complete device circuit and cable route location information was then compiled under a database and used during the Fire PSA when assessing the consequential damage of postulated fires. This allowed for end devices to be readily identifiable when interconnecting cables are damaged during a fire. Future steps include ensuring that any revisions to IntEC or the FHA are reviewed by PSA staff to ensure the fire PSA is maintained up to date.

12. Conclusions

NB Power Nuclear has developed a PSA that incorporates a high degree of operational information and site-specific data to evaluate risk metrics based on the way the plant is actually operated and maintained. This paper has presented the unique characteristics of the PSA prepared for Point Lepreau Generating Station and the benefits and detriments of the approaches that were followed. The high degree of model depth or resolution has enabled quick turn-around time in response to operational queries and the ability to quickly evaluate the impact of any system design modification. The next project includes using the PSA models as the foundation for development of an on-line risk monitor.

IMPROVEMENT OF POWER MEASUREMENT AND PLANT EFFICIENCY AT ATUCHA 1 NPP

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Abstract

Turbine Performance is an important parameter in any nuclear power station. Its surveillance not only improves plant economy and equipment maintenance but also may be used as a control of power reactor measurement and to improve instrumentation maintenance. This paper describes a practical case.[1]

1. Brief description of Atucha I NPP

ATUCHA I is station with a heavy water pressure vessel reactor. Its total thermal power is 1179 MW (t) and its gross electric power 357 MW(e) at a river water temperature of 17°C. It was built and designed by KWU A.G. and it is under commercial operation since 1974.

A simplified scheme of main reactor circuits and part of feed-water system is shown in Figure 1 (only one loop is shown in the figure, but actually there are two loops for coolant and also two loops for moderator).

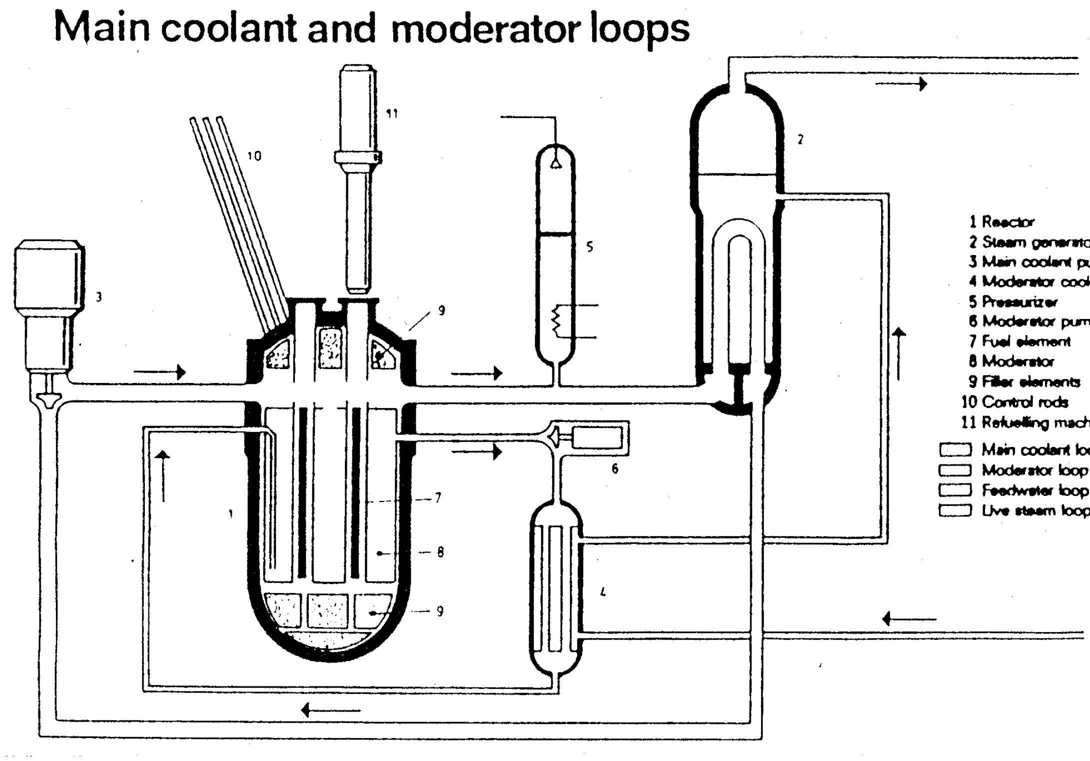

FIG. 1. Main coolant and moderator loops.

[1] For additional information or comments on this paper send e-mail to ingenieriacoord@nasa-com.ar

In this kind of reactor the main coolant heat transport D_2O is guided through 250 coolant channels made of Zircaloy-2. The channels are installed inside the moderator tank and all of these components are placed inside the pressure vessel. Coolant and moderator are not full separated circuits; they are connected by a circular 10 cm wide slit at the top of moderator tank to achieve pressure equalization because the moderator tank is not a pressure vessel.

The moderator system receives not only the energy coming from neutron moderation and gamma radiation, but also receives additional energy because of heat conduction through moderator tank wall, the coolant channels walls, and partial mixing of coolant and moderator systems. As a consequence of these facts, the moderator system takes 10% of the total nuclear power compare with the 5% in a CANDU Type Reactor; however, this energy is used to preheat the feed-water as may be observed in Figure 1.

The feed-water system, shown partially in the figure, has a tank (not shown in the figure) that receives: condensate water, water coming from water-steam separator at the HP turbine outlet, and steam from a bleeding of the LP turbine. Feed-water is pumped to the steam generators via the moderator heat exchangers that work, in this case, as HP pre-heaters. steam generators are U-tube type top-feed.

The turbine is of the type having one high-pressure cylinder and three low-pressure cylinders. A cyclone separator is placed at the outlet of the high-pressure turbine. Each low-pressure cylinder is provided with three double extractions (bleedings), two of them are used to preheat the condensate, and the other one is injected directly in the feed-water tank.

The condenser is a two-pass, river-water-cooled type. Condensate is pumped via the low-pressure pre-heater and injected in the tank by means of a special kind of valve that sprays it.

Design Operational Parameters of the Plant

Main Primary system at full power

D_2O inlet temperature to reactor 265°C

D_2O outlet temperature from reactor 300°C

Total primary flow throw the core 6000 kg/s

By pass flow to the top of RPV 4%.

Differential pressure at the RPV 7 bar

Pressurizer pressure 113 bar(g)

Moderator system at full power

D_20 outlet temperature 160°C

D_20 outlet temperature 220°C

Total flow 440 kg/s

Feed-water system

Total flow 520 kg/s

Temperature in the feed-water tank 120°C

Temperature after moderator heat exchanger 165°C

Steam and Turbine

Steam Properties: 43 bar(g) saturated

Turbine Pressure before admission valve 40.5 bar(g)

Turbine pressure at low pressure stage admission 4.7 bar(g)

Condenser

Temperature in the hot-well 30°C to 50°C depending on river temperature

Temperature at LP pre-heater outlet 80°C

Total Flow 430 kg/s

Generator

Maximum Power 400 kW

Rotating speed 3000 rpm

Voltage output 21.5 KV

Cooled by H_2

2. Atucha I instrumentation and control technology

The instrumentation of Atucha I corresponds with the end of the 1960’s and early 1970´s technology.

Most of the electronic circuits are low integration and discrete component. The electronic part of the controllers is also mostly discrete. Valves actuators are mainly motor driven type.

Temperature sensors

Platinum resistance thermometers (Pt100)
These sensors, which constitute the majority of temperature sensors in the plant, are mounted in DIN normalized wells. Most of them have a typical response time of 5 second. These sensors are connected to the transducer in the three-wire mode. The transducers produce a signal in the range 0-20 mA. No correction for Pt100 non-linear behaviour is available at this type of transducers.

Thermocouples
Thermocouples used are of K type. They are used where a faster response is required, i.e. reactor power control. The ones to measure temperature jumps are connected in opposition. Those that measure absolute temperature have a platinum resistance for cold junction correction. In most cases voltage generated by the thermocouple is converted in a 0-20 mA signal.

Pressure Sensors

Line pressure or differential pressure measurements are either inductive or capacitive ones; all of them transmit the signal in 0-20 mA standard.

Flow-meter measurements

Most of flow-meter measurements are based in orifice plate.

Electric measurements

All electric measurements (voltage, current, resistance) are also converted to 0-20 mA.

Signal transmission and interfacing

Transmission are mainly based on 0-20 mA signals. If a signal is used for more than one purpose, (i.e. control, information and safety), the same signal is transferred through a magnetic galvanic separator. Hence, safety related and non-safety related circuits exist for the same signal.

Flow is obtained from analog root square of delta p signal from DP cells. When required, analog temperature correction is applied to measured flow

Accuracy and repeatability (precision) of measurements

Most process measurements mentioned in the previous paragraph have accuracy about 1% F.S, but reproducibility (precision) is usually between 0.1 and 0.3%. F.S

Process Data Acquisition at Atucha I

The main process control computer collects 568 analog signals and over 2500 logic signals. Originally data were collected and maintained for 20 minutes in the computer memory, and some selected data printed regularly. The conversions were performed by means of four multiplexed 11 bits analog digital converter. Several upgrades to the computer adapted it to new technologies, allowing the saving of one year of operating data and performing lots of additional calculations compared to the few calculations that the original computer was able to do. A new software ‘VISUAL DATA TM’ was installed and helped visualization of the data. Next year converters will be replaced and sampling time of the variables (currently maximum sampling is every 10 seconds), will be increased to 1 measure a second. On the other hand, many variables that have instrumentation not connected to the computer are collected with a field-data collector, and by means of a code (TRONADOR TM), sent to the same database, making possible to visualize this data together with data obtained from the main computer.

3. Using process characteristics to improve accuracy of process measurements and maintenance performance

3.1. Process measurements

Since precision is higher than accuracy, it is possible to improve accuracy by applying crosschecking techniques to measurements or to a measured variable and a calculated one; e.g. pressure and temperature when saturation is present. The background for these data-reconciliation techniques is partially derived from experience but also from papers of Hashemian *et al*. A list of Hashemian papers may be found at *http://www.patentstorm.us/patents/6973413.html*

This technique takes advantage of measurement redundancies that usually exist at any power station.

A comparison is performed between different variables when it is possible and without jeopardizing plant safety or availability. One redundancy, in most cases one in a spare line sensor, is measured with a laboratory instrument and used as reference. By this means the accuracy of measurements can be increased, achieving estimates closer to the true, exact value. This is done by a programme resident in the on-line computer. The computers A/D[1] converters are controlled by means of a constant current loop that creates a voltage signal, which is measure by each of the four converters. In this way, converter accuracy is checked since it is very improbable that more than one of the four converters will shift simultaneously. In addition, every two months a high-accuracy voltage signal is injected in one converter.

3.2. Example 1 — Pt100 temperature measurements

Pt 100s precision: Figure 2 shows the time history of three different PT100 measurements working at the same temperature over several months. It can be seen that the overall accuracy is only 3°C but the relative precision is close to 0.2°C.

Procedure to increase accuracy: With the reactor at its nominal output and under manual control, a spare Pt100 on a line is measured using the four-wire technique and a 0.03 Ω instrument. This provides the reference temperature to be compared with other temperature measurements on the same line. These corrections are entered in the computer programme. In Table 1, one example of this procedure is presented. It should be noted that these measures have a 0-300°C range. The total 0.23 Ω difference in the resistance indication, equivalent to 0.6°C, is examination. It should be noted that 1°C accuracy is a reasonable error for this measurement.

Table 1. Example 1

Signal	Resistance (Ω)	Temperature (C)	Visual Data	Correction
RL10T003	145.50	118	114.75	+3.55
RL11T001	145.73	118.5	116	+2.5
Rl12T001	145.67	118.3	114	+4.3

Notes: 0.22 Ω was subtracted because of internal wire resistance of Pt100 device

[1] Analog to digital

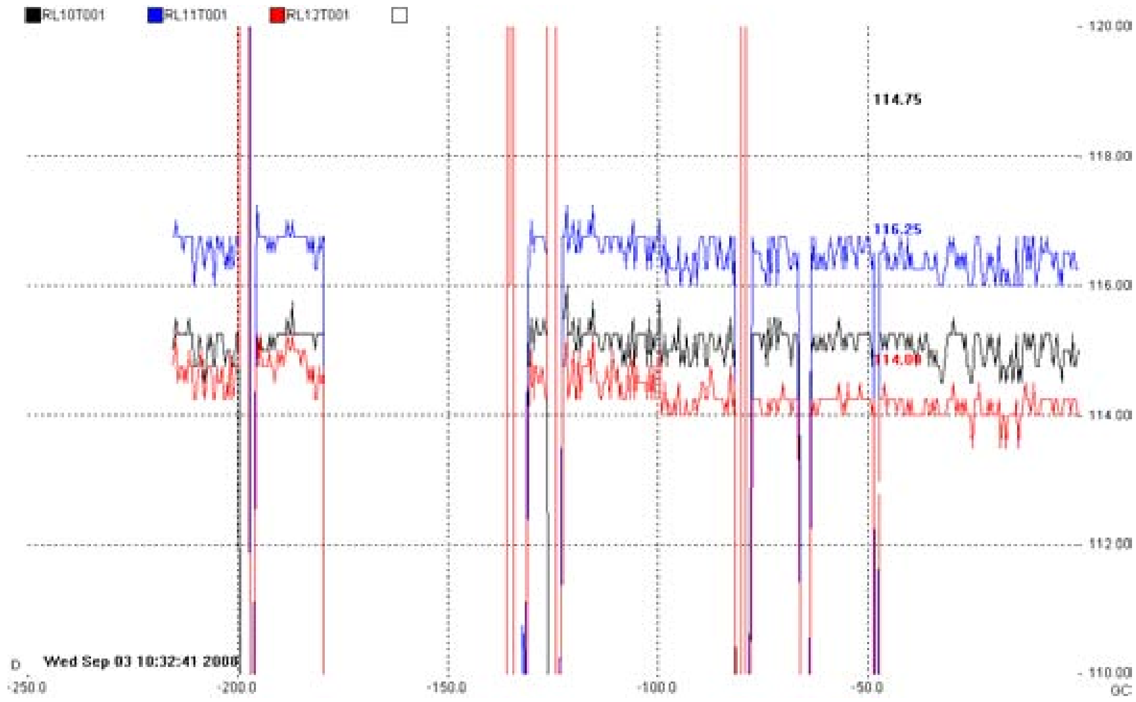

FIG. 2. Three temperature measurements of feed-water vs. time in days, for these measures 1 bit is equivalent to 0.25°C.

3.2. Example 2 – feedwater flow measurements

Feedwater flows are measured by means of orifice plates that at full power produce a differential pressure close to 10 m water column. The differential pressure signal is converter to a 0-20 mA current. The signal is corrected for water temperature, and then square root of the signal is obtained. These two operations are made using analog equipment. After correction the signal is proportional to flow. In order to increase precision and accuracy, the differential pressure signal was also sent directly to the computer, where corrections are done digitally. The analog circuit was used only for analog indication and control.

Both feedwater lines (see Figure 3, where only components referenced in this text are shown) are provided with two flow meters, one at the steam generator inlet (FM2A, FM2B) and other at the Moderator Heat exchanger inlet (FM1A, FM1B).

Under standard operation conditions the flow meters in each line do not show the same indication for two reasons:

- The valves VSA and VSB are normally open and about 3 to 4% of the flow passes to the steam generator bypassing the Moderator Heat Exchanger; and
- The valves VIA and VIB are also normally open and water can also flow throw this line.

For short (2-3 hours) periods of time (and under another conditions not discussed in this paper) it is possible to close the above-mentioned valves and obtain a system configuration where it is possible to cross check precision of the flow meters. At the time of writing insufficient data have been collected to establish a final figure for cross-precision, but it is possible to confirm that it is better than 0.5% of full flow.

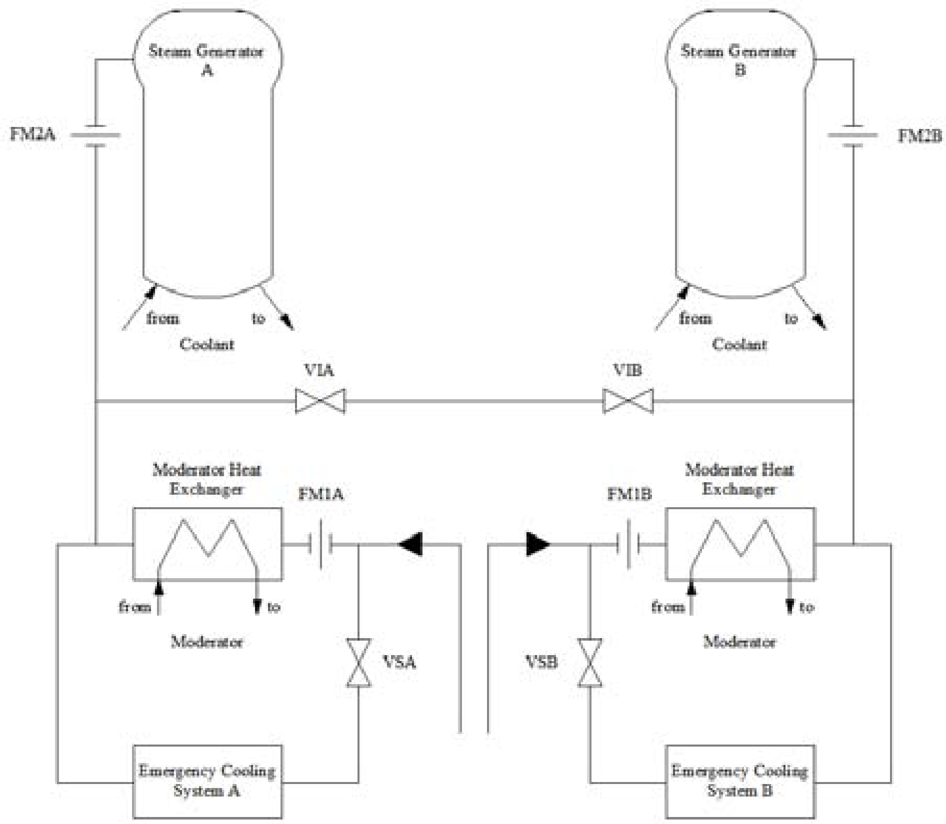

FIG. 3. Feedwater lines.

3.3. Example 3 — main coolant pump flows as a method of checking the overall reliability and surveillance of instrumentation used to calculate reactor power.

In Atucha I, like in most stations, the coolant system does not have flow sensors. It is simple, however, in Atucha I to calculate coolant flows since the steam quality at the reactor outlet is zero. Flow is obtained by calculating the power in each steam generator from the secondary side, plus a calculation of the enthalpy change on the primary side.

Since the geometry of the coolant system is almost invariable, the coolant flow is expected to be constant for a given operational condition. As a consequence, flows at the main coolant pumps depend on electric grid frequency but this is also a constant.

Figure 4 shows calculated flows for the two pumps. The calculation is done once every 100 seconds and in this case averaged over 1 hour period. Note that the flow remains constant within 1%. It means that flows, temperatures and pressures have an overall cross precision close also close to 1%.

3.4. Example 4 — turbine-generator output as a method of checking the reliability of reactor output measurement

Electric measurements usually have higher precision and higher accuracy than temperature and flow measurements. Measuring the electric output of the generator provides a better method of checking the performance of instrumentation than in the method describe in Example 3. This method also permits the surveillance of turbine performance.

In Figure 5 the histories of turbine efficiency, reactor power, exhaust steam temperature and river water temperature during the current year are shown. It can be seen that approximately equal efficiency corresponds to equal river temperature; it means that electric power, reactor power and river water are properly measured.

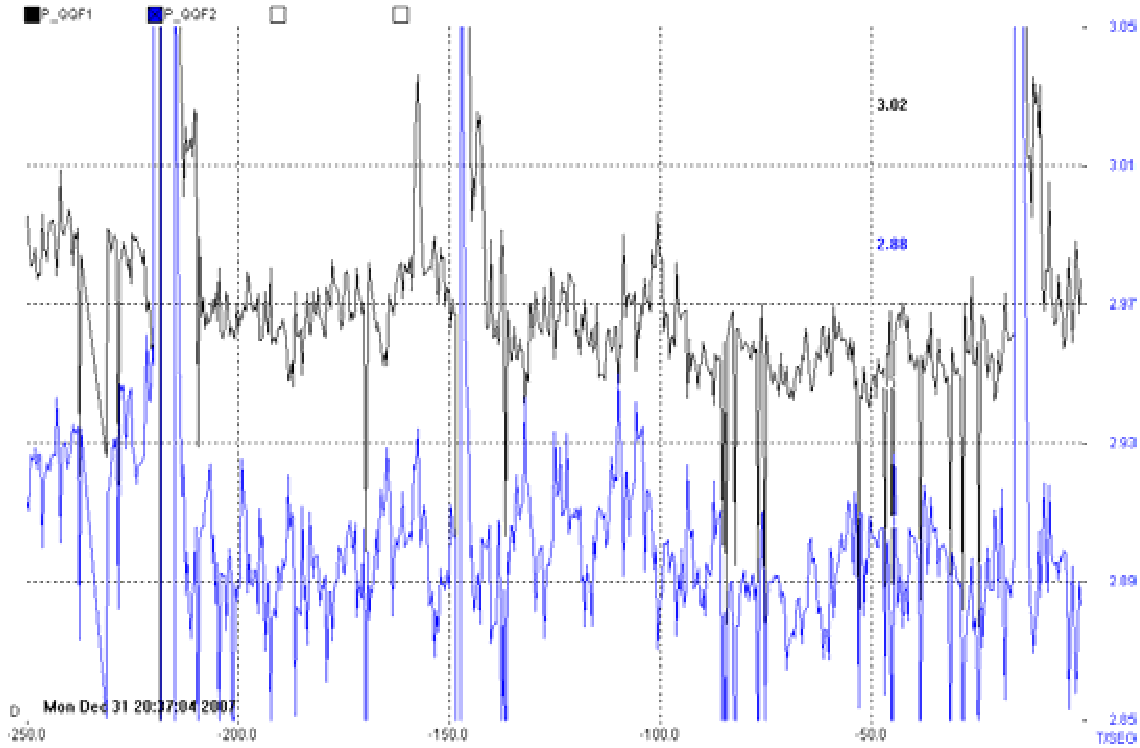

FIG. 4. One hour averaged main pumps flow vs. time in days, the scale for upper line is 2.9-3.1 metric tones a second.

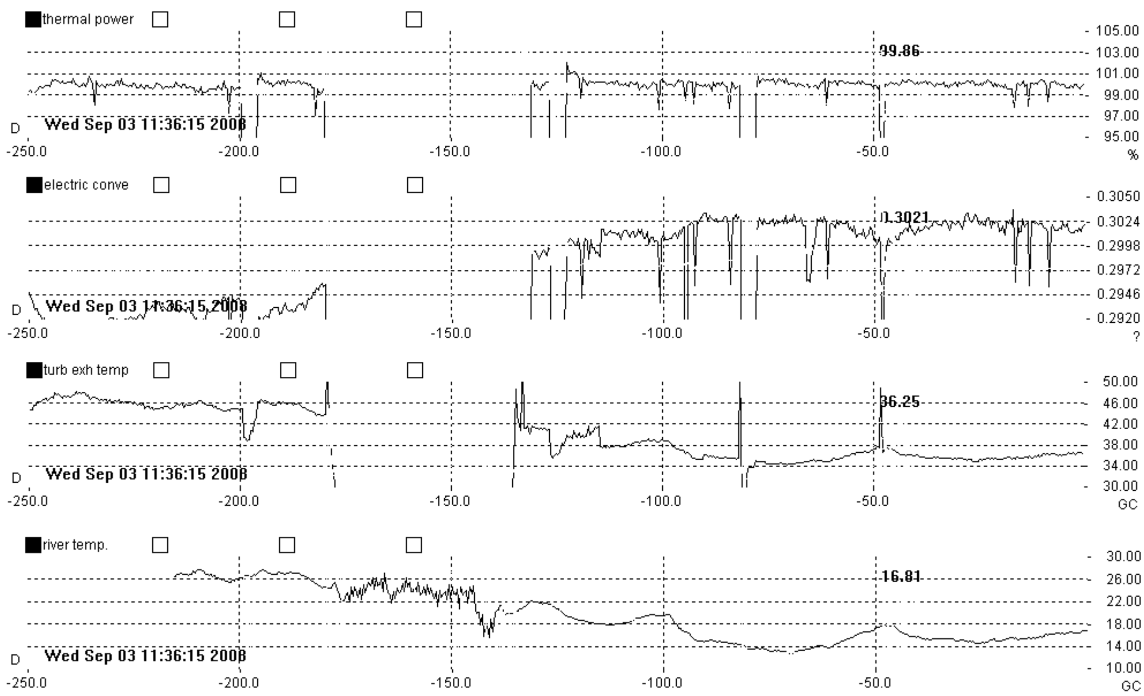

FIG. 5. Reactor thermal power, conversion efficiency, turbine exhaust temperature and river temperature vs. time.

To increase the accuracy and precision of measured generator output, gross power is compared with the sum of:

a) Net power obtained from the energy measurement done at the station by the grid operator
b) Internal station Power
c) Main transformer losses

From the data in Table 2, it may be concluded that the cross precision of the two measures is 0.2%. The specified accuracy for the net energy meter is better than 0.2%.

In conclusion, if all processes in the turbine-condenser-generator are working properly and reactor power is being measured correctly, turbine efficiency should be a constant for a given set of plant state parameters. This demonstrates that the method is a good diagnosis tool of turbine performance and reactor power measurement accuracy.

Table 2. Example 4

Date	25/01	26/02	28/02	06/03	29/04	13/05	24/06	11/07	23/07	08/08	27/08
Net power from plant data (Mw)	321.7	324.3	323.8	323.0	330.8	332.9	334.7	333.0	334.5	333.8	333.9
Net power from grid operator (Mw)	324.3	326.3	326.2	325.7	332.8	335.2	337.0	335.4	337.1	335.8	336.2
Difference	2.6	2.0	2.4	2.7	2.0	2.3	2.3	2.3	2.6	2.0	2.3

4. Instrumentation maintenance improvements

The techniques presented above have also allowed us to arrive at some conclusions regarding instrumentation maintenance.

- When cross precision remains constant, calibration may be deferred or avoided;

- The use of spare ‘on line’ sensors to increase measurement accuracy should be extended as much as possible;

- Global measuring techniques that not involve removing of modules from racks and provide good predictive information should be implemented; e.g. total voltage drop along the current circuit, voltage drop across components, noise measurements, and in general on line diagnosis techniques;

- Comparative noise measurements of variables in some cases permit surveillance of instrumentation, especially in cases when process noise may be discriminated. This technique has begun to be implemented at Atucha I. Mean square roots are calculated and recorded for several signals taken from the process computer. Due to limited sampling time, limited information has been obtained using this method, but it should be noted that for redundant measurements that are working properly, noise would have to shown the same behaviour;

- Techniques that permit the discrimination of instrumentation noise from process noise are promising and should be developed.

To apply these maintenance techniques to instrumentation related to safety and/or high-cost equipment, additional work in the areas of risk analysis and safe procedures should be done.

EVOLUTION OF THE WOLSONG NPP F/M D_2O PRESSURE CONTROL SYSTEM

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Abstract

The four units at the Wolsong site are 700 MW(e) class CANDU 6 reactors. Wolsong Unit 1 went into service in 1983 and Wolsong Units 2, 3 and 4 began commercial operation in 1997, 1998 and 1999, respectively. Fuel handling control systems for Wolsong Unit 1 are designed using analog cascaded PID controllers, while Units 2, 3 and 4 use digital PID controllers. Fuel handling operators and technical staff at Wolsong Unit 1 had encountered difficulties in operation and maintenance because of frequent system instabilities and failures that stem from aging and the use of substitute parts due to obsolescence of hardware. The performance, reliability, maintenance and operational convenience of the Wolsong Unit 1 fuel handling D_2O pressure control system has been improved within the framework of an existing fuel handling control system. An improved control algorithm has been developed and implemented in the system to improve pressure-control performance. Backup pressure controllers that are supplied by diverse control power sources have been implemented to enhance the system reliability, preventing the system from experiencing a total loss of pressure control. Some alarms are provided in the fuel handling control board for operational convenience so that operators can easily recognize abnormal conditions in system process and the electric circuit of the controller. An ABB 53MC5000 Process Control Station digital controller was selected rather than a programmable logic controller for economic reasons. It was decided to replace the existing pressure controller and algorithm for Wolsong Units 234 with the improved control algorithm from Wolsong Unit 1 because they did not meet our control-performance, system-reliability and operational-convenience expectations. These evolutionary design changes have been successfully completed and will be implemented in Wolsong Unit 4 during the next outage. The next stage of design evolution is the scheduled adoption of distributed control system (DCS) features for the fuel handling control system in order to replace all electronic devices with software and to remove fuelling operations and monitoring in the Digital Control Computer-Y (DCC-Y).

1. Introduction.

The CANDU fuel handling system provides on-power fuelling capability at a rate sufficient to maintain continuous reactor operation at full power. During on-power fuelling, the F/M[1] becomes an extension of the reactor fuel channel end fitting and is subjected to the pressure in the primary heat transport system. A heavy water environment in the magazine housing of F/M is required because this region is in contact with the heavy water of the reactor primary coolant system during fuel changing and coolant is required to remove the heat of irradiated fuel bundles in the F/M. The F/M D_2O control system provides the heavy water environment at the required conditions to different parts of the F/M.

The F/M D_2O control system is divided into two D_2O process systems: the F/M D_2O supply system and the F/M D_2O system, as shown in Figure 1. The F/M D_2O supply system provides pressurized D_2O to two F/M D_2O system valve stations (‘A’ side and ‘C’ side). The pressure levels of F/M D_2O supply system depend on operational modes of HIGH, MEDIUM, and LOW.

The F/M D_2O system is composed of the actuator control circuits and magazine pressure control circuit. The actuator control circuits provide environmental control at the local regions of hydrodynamic seals on the drive shafts for the magazine; hydraulic actuation of the C-ram and fuel separators. The D_2O flow of the magazine pressure control circuit is regulated by PCV-1 to maintain one of four magazine pressures (HIGH, INTERMEDIATE, PARK, and LOW), which is interfaced with the F/M D_2O supply pressure mode.

The existing control system of the F/M D_2O supply system for Wolsong NPP Unit 1 is based on an analog cascaded proportional-integral-differential (PID) control. A PID controller is a hardware module used widely in the control system to control a process parameter to the target value. The analog control system has many electronic devices and requires much effort to set up the system during system calibration. Recent developments in digital control technology attract considerable interest in system design and maintenance. In particular, a digital controller provides considerable flexibility to change the control logic. To alleviate the effort required during system operation and

[1] Fuelling machine.

maintenance and to improve the system performance, a new digital control system was an option to replace the existing analog control system of the F/M D_2O supply system for Wolsong NPP Unit 1. The F/M D_2O supply system of Wolsong NPP Units 2, 3 and 4 have already adopted digital PID controls.

Operating experience has shown that a large overshoot in common-header pressure sometimes challenges the pressure setpoint of the system relief valve installed to protect the system from overpressure, thus causing a heavy water spillage. It is quite difficult, however, to identify the dominant factors affecting the common-header pressure behaviour during system operation and maintenance.

Operating experience has also shown that the common bleed valve in the F/M D_2O supply system goes to the full closed position during mode change in either direction when one pump operation. This valve closing is undesirable because it may damage the valve seat, but has been experienced in both the existing analog and digital controllers. Resolving the problem requires considerable effort in setting the parameters of the F/M D_2O supply system.

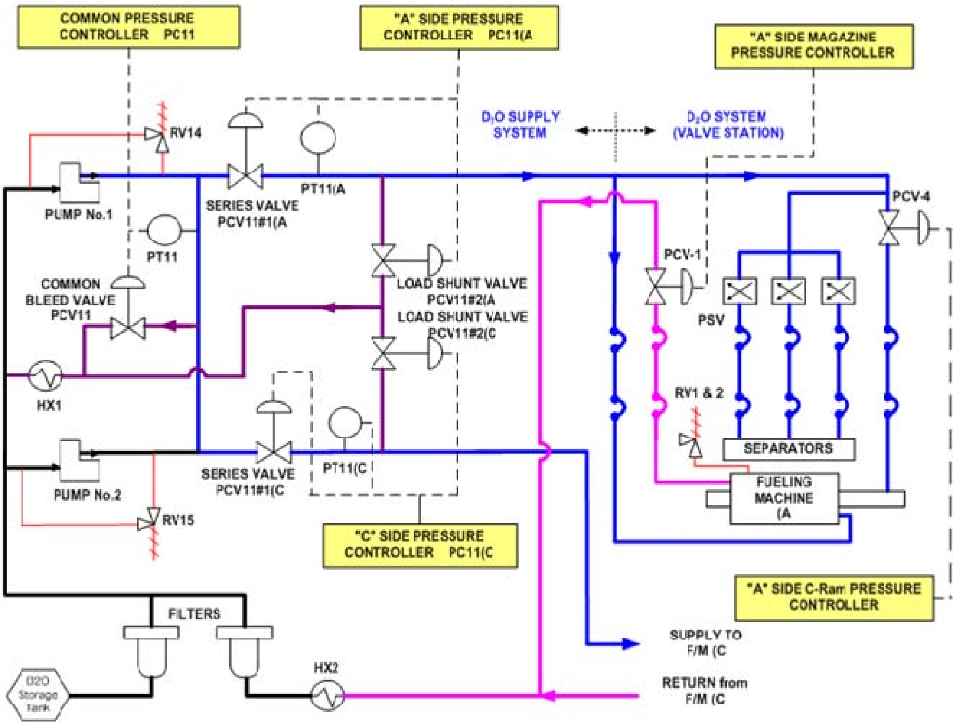

FIG. 1. Schematic diagram for the F/M pressure control system.

2. System description

D_2O is drawn from the primary heat transport system storage tank through the positive displacement pumps (PDPs). During normal operation only one pump is in use with the second pump as a standby. Each pumping circuit is identical, and each pump can provide a minimum flow of 4.54 L/s (72 gpm) at pressures of up to 18.62 MPa(g) (2700 psig). Relief valves are installed at the discharge of the pump to protect the system from pressures above 18.62 MPa(g).

The interconnection line from the two pumps serves as the take-off point for the line to the common bleed valve. The series and load shunt valves in each subsystem perform the basic pressure control function for D_2O supply to each of two fuelling machines. The common bleed valve serves as a decoupling mechanism between two-subsystem output supplies so that a major disturbance in one subsystem will not seriously affect the other. The common and load shunt valves bypass the excess D_2O to the pump inlet through a bypass cooler.

Three automatic pressure controllers (PC11, PC11-A, and PC11-C) are provided for the F/M D_2O supply system. One is for the common bleed valve control and the others are for the series and load shunt control valves of each D_2O supply channel. Each controller functions independently to maintain its associated system output pressure. The setpoints for each controller are set independently, either manually on the control console sub-panel or automatically by the fuel handling system control computer.

Pressure control is achieved by simultaneously adjusting four control valves: the series valves (PCV11#1-A and PCV11#1-C) and load shunt valves (PCV11#2-A and PCV11#2-C). The series valve acts as a throttling valve, passing a little more flow than that drawn by the load (F/M D_2O system), and reducing the pressure to the appropriate level. The load shunt valve acts as a bleed valve to drain the portion of flow not required by the load, and to ensure that the output pressure stays at its preset value regardless of routine fluctuations of the load flow.

Provision is also included for regulation of the common-header pressure (PT11). The common bleed valve (PCV11) is controlled to hold the common-header pressure at a preset value. This serves as a decoupling mechanism between two output supplies (PT11-A and PT11-C), so that a major disturbance in one system will not seriously affect the other and the stability of control is not seriously impaired by interactions between two systems. It is also helpful to attenuate the disturbances imposed upon the system when the second pump in the system is started or stopped.

D_2O from the supply system reaches the F/M via its valve station and a catenary system, which is a set of hoses to allow the F/M movement necessary to complete its tasks. At the valve station the D_2O supply splits into two: one line for the actuator control circuit and the other for magazine control circuit. The magazine circuit controls the pressure conditions in the F/M head magazine while the actuator circuit controls ram actuation, feelers, retractors, stops and seals supply. The actuator circuit is referenced to the magazine pressure via the magazine pressure sense line. A pneumatically actuated control valve (PCV-1) located in the magazine return line controls the magazine pressure. The associated pressure controller sets up the desired pressure condition with a feedback signal from the pressure transmitter in the magazine pressure.

3. Design and limitations

3.1. D_2O supply pressure control

The previous control system of the F/M D_2O supply system for Wolsong Unit 1 was based on an analog cascaded proportional-integral-differential (PID) control. It was manufactured by Canada GE and installed in 1981. The analog control system had many electronic devices and required considerable effort to set up during the system calibration. Fuel handling operators and technical staff encounter difficulties in system calibration, operation and maintenance from the start-up stage in Wolsong Unit 1. The analog control system was difficult to maintain due to frequent electronic device failures and obsolescence of the hardware. The control system required more manpower for inspection and tuning, and went through an instability in system performance that affected safety function.

The control system of the F/M D_2O supply system for Wolsong Units 2, 3 and 4 used digital PID controls, but its original algorithm did not incorporate sophisticated process characteristics. Operating experience had shown that a large overshoot of common-header pressure sometimes challenged the pressure setpoint of the system relief valve installed to protect the system from overpressure, and also caused heavy water leakage. It was quite difficult, however, to identify the dominant factor that affected the common-header-pressure behaviour during on-line system operation and maintenance.

Resolving these problems was motivation to develop a control system optimized in performance, reliability, maintenance and operational convenience within the framework of an existing fuel handling control system.

3.2. F/M magazine and C-Ram pressure control system

Depending on the F/M mode of operation, the magazine pressure control system maintains the magazine D_2O pressure at four pressure levels: Low, Park, Intermediate and high. The C-Ram pressure control system provides five pressure setpoints corresponding to the five specified C-Ram force levels of force 1 through force 5. The setpoint of the magazine pressure control system and the C-Ram pressure control system is selected by variable resistor, multi-contact relay and 5VDC power. The setpoint is greatly affected by malfunctions of these devices, and enters the condition of setpoint stall. The setpoint disturbance induces the undesirable pressure transient, large heavy water spillage from fuelling machine, or loss of cooling function for spent fuel in the F/M magazine.

A standby controller is provided in the magazine pressure control system to improve system reliability, but it does not provide bumpless transfer when changing from normal to standby controllers to avoid upsetting the process. The existing design configuration of normal and standby controllers causes the undesirable pressure transient on operator error.

4. Design considerations

Figure 2 shows a block diagram of the F/M D_2O supply control system for Wolsong Unit 1 before the design change. To ensure system reliability, each controller independently provides automatic and manual functions and has diversified power supplies.

Figure 3 shows a block diagram of the F/M D_2O supply control system for Wolsong Unit 2 before the design change. The digital control system is implemented but it does not provide a backup controller nor diversified power supply. This system is inferior to that used in Wolsong Unit 1 before the design change.

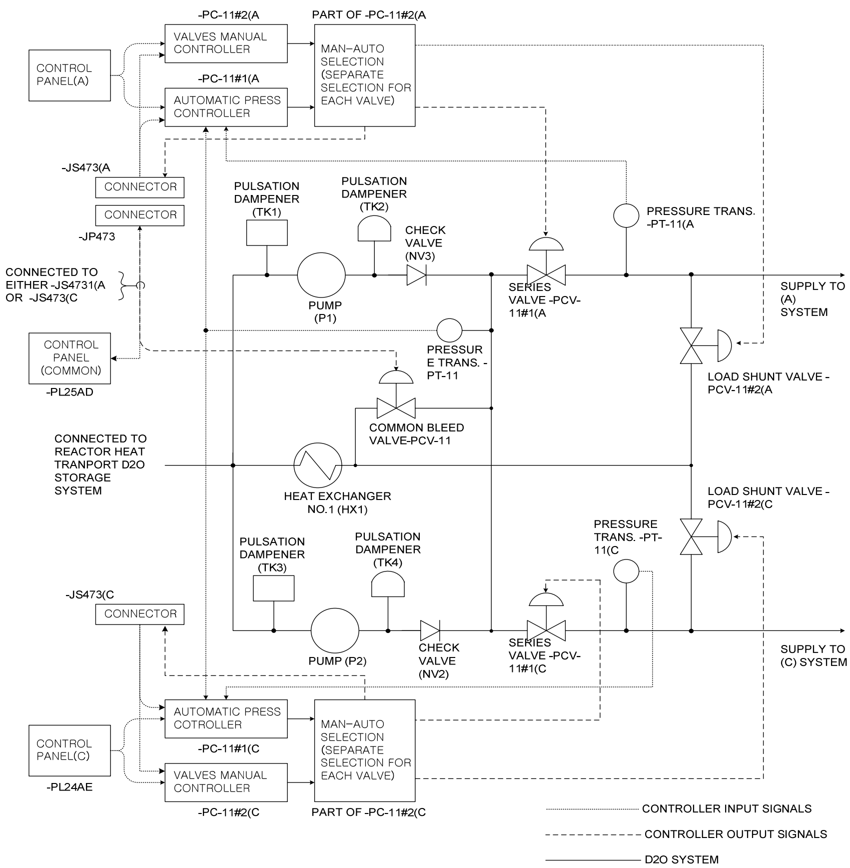

FIG. 2. Block diagram for F/M D_2O supply control system for Wolsong Unit 1.

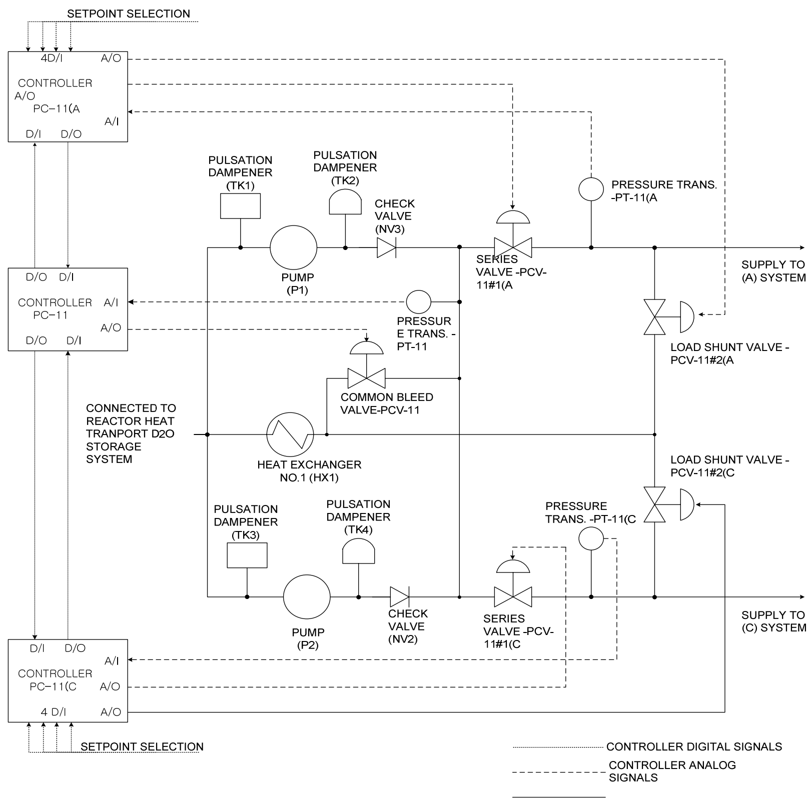

FIG. 3. Block diagram for F/M D_2O supply control system for Wolsong Unit 2.

At the first stage of the design change for the F/H control system, the philosophy of the design change was set forth as follows;

1) The system shall meet the requirement for safety functions of integrity of the HTS pressure boundary, spent fuel cooling in F/M magazine, limit of radioactive material release.
2) The system shall incorporate all the control functions of the analog PID control system for Wolsong Unit 1.
3) A control algorithm shall be developed and verified using simulation results from a realistic process model.
4) Each control system shall be composed of normal and backup controllers and upon the failure of the normal controller it shall be automatically transferred to the backup controller.
5) The power supply for normal and backup controllers shall be diversified.
6) The controller shall be the same model as used in Wolsong Unit 2.
7) Magazine pressure and C-Ram pressure controllers shall be replaced with digital controllers
8) The hard-wired logic of pressure setpoint selection for magazine pressure and C-Ram pressure control system shall be replaced with soft-wired logic.
9) The current alarm unit shall be furnished in a digital controller to provide the process interlock related with magazine pressure.

5. New design/development

The F/M pressure control system in Wolsong NPPs had required some design changes to prevent potential events and reserve obsolescent parts. A potential problem in a system eventually results in an event or system failure. It is very important to resolve potential problems that are acknowledged during system operation. The motivation for a design change in the Wolsong fuel handling area was to prevent the system from losses of various safety functions. The mayor losses of safety functions that have occurred in Wolsong NPPs are as followings:

- Loss of cooling for spent fuel in the magazine due to a pressure setpoint stall for F/M magazine;
- Spent fuel exposure to air at the spent fuel discharge port because of a loss of function of the pressure control valve in the F/M D_2O supply system;
- Large heavy water spillage through relief valves due to overpressure in the magazine during closing the channel plug.

The F/M pressure control system design change for Wolsong NPPs has been performed to resolve the lessons learned from various safety-related events and to protect humans from hazards. The design considerations described in Section 3 are also considered in this design change because the stability of pressure control is important in fuel changing.

5.1. D_2O supply pressure control system

In the F/M D_2O pressure control system for Wolsong Unit 1, the analog cascaded PID controller was replaced with a Fisher & Porter model 53MC5000; the same model is installed in Wolsong Unit 2. A state-of-the-art algorithm was developed and implemented in the system for Wolsong Unit 1. It incorporates the existing control logic found in the analog controller with simulation results from a realistic model as shown in Figure 4. For the optimized pressure control, it becomes a plant specific algorithm to incorporate the process and system specific characteristics.

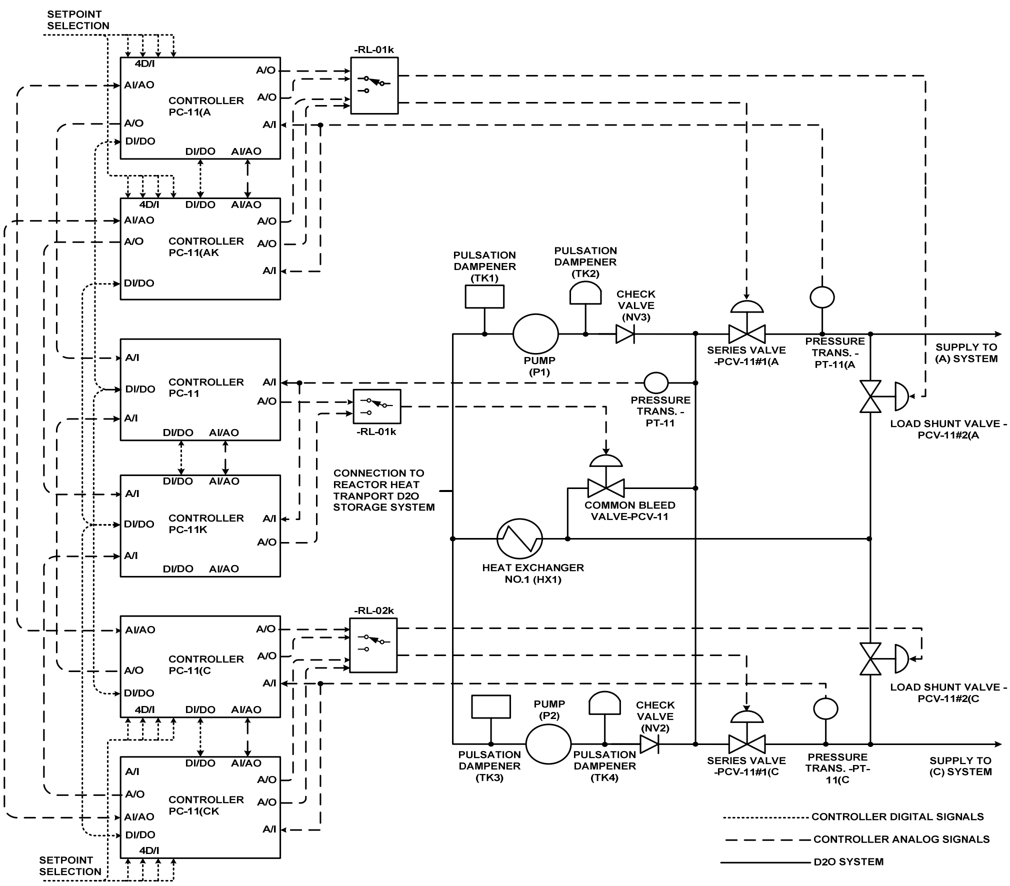

FIG. 4. Block diagram for F/M D_2O supply control system for Wolsong NPPs.

During the start-up test for Wolsong Unit 2, pressure control was not achieved for normal operating conditions and algorithm changes were required for the dual setpoints of the common-header pressure in high mode. The pressure control system for Wolsong Unit 2 has the potential problem losing pressure function if one of three controllers fails due to controller or power supply problems. The plant-specific control system for Wolsong Unit 2 is developed and implemented with design change experiences in Wolsong Unit 1 and system characteristics for Wolsong Unit 2.

The same configuration and hardware has been used in the two plants to allow an operator to effectively manage plant configuration management, maintenance, and reserved parts and to share operating experience.

5.2. F/M magazine pressure control system

In the F/M magazine pressure control system for Wolsong Unit 1, the analog cascaded PID controller was replaced with a Fisher & Porter 53MC5000 model, which is the same model as is installed in Wolsong Unit 2. The major design change is to add bumpless transfer and replace the hard-wired logic with soft-wired logic for the pressure setpoint selection. Bumpless transfer allows a change from normal to standby controllers without upsetting the process, and the backup controller always tracks the normal one. The soft-wired logic removes the root cause of pressure setpoint stall from functional failure of electronic devices and erratic relay contact.

5.3. F/M C-Ram pressure control system

In the F/M C-Ram pressure control system for Wolsong Unit 1, the analog cascaded PID controller was replaced with a Fisher & Porter 53MC5000 model, which is the same model as is installed in Wolsong Unit 2. The major design change is the same as for the F/M magazine pressure control system, above. PID control logic is applied to the force 1 through force 4 modes, and On/Off control logic is applied to force 5 to protect the F/M components against C-Ram overspeed.

6. Summary

Design changes for the fuel handling system of the Wolsong NPPs have been successfully performed. The first stage was to improve the performance and system reliability of the pressure control system. The next scheduled stage of design changes is to adopt distributed control system (DCS) features for the fuel handling control system. DCS features for the fuel handling control system permits the replacement of all electronic devices with software, and the removal of fuelling and monitoring from the Digital Control Computer-Y (DCC-Y). The maintenance inferences between the fuel handling system and plant computer system can be corrected during the periodic plant overhaul.

For the application of DCS to fuel handling system, the DCS for the fuelling machine test rig (FMTR) has been developed from the programmable logical control (PLC) features and it will be completed by the end of 2009. For the protection from hazards and system reliability, the digital control functions of fuel handling system will be continually enhanced in Wolsong NPPs.

PLATON AND SPV PROGRAMME – MEANS OF IMPROVING RELIABILITY OF CERNAVODA NPPs

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Abstract

A data-acquisition system for plant information named PLATON is connected to the plant digital control computers and allows inspection of trends, bar charts, alarms and hard copies of main control room displays. It allows the construction of historical trends of process variables. PLATON helps plant expertise be quickly focused during plant transients and is a good tool in emergency exercises. SPV means ‘Single Point of Vulnerability’ project in Cernavoda NPP. The main goal of the SPV initiative is to reduce to as low a level as possible the vulnerability of the plant to events as a result of failures of equipment representing SPV. The experience gained during this initiative development will be used to improve the plant preventive maintenance programme.

1. Cernavoda NPP plant information system

From the beginning of commercial operation of Cernavoda Unit 1, personnel in the Technical Division felt the need for a tool to help them perform system surveillance. In 2000, after four years of operation, personnel from the Computers Technical Department developed and installed a data-acquisition system for plant information named **PLATON** (PLAnt data Tailored informatiON). This is a collection of software applications developed by the plant staff to meet the process-information needs for management, engineering and systems surveillance. They also developed operation and maintenance applications.

PLATON is connected to the plant digital control computers (DCC) and allows inspection of trends, bar charts, alarms and hard copies of Main Control Room displays. It allows the construction of historical trends of process variables (See an example in Figure 1).

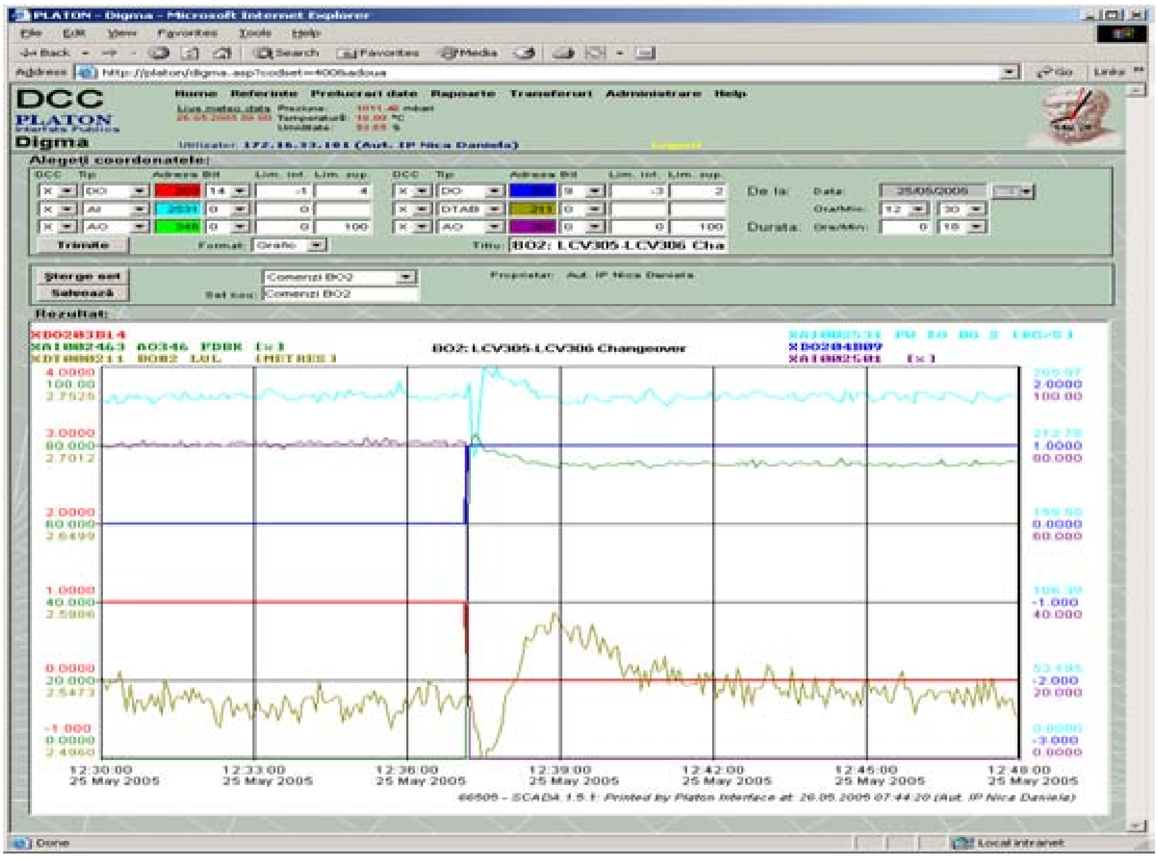

FIG. 1. PLATON – trends of process variables.

The **PLATON** web server allows station-wide access for quasi real-time and historical parameters monitoring and is used by 350 users for:

- trending visualization,
- historical archiving,
- system health and performance monitoring,
- work management,
- training.

In **PLATON,** 7000 analog parameters and 2240 digital parameters are available at four-second time intervals.

PLATON displays reports produced by the Plant Digital Control Computers DCCX and DCCY as Alarm pages (Figure 4), Station Logs (Figure 5), and Flux Mapping Printouts (Figure 2 & 3). Real-time and historical hardcopies of the Main Control Room Displays are available also. Real time Main Control Room alarms are available on every system engineer’s desk.

PLATON helps plant expertise be quickly focused during plant transients and is a good tool in emergency exercises.

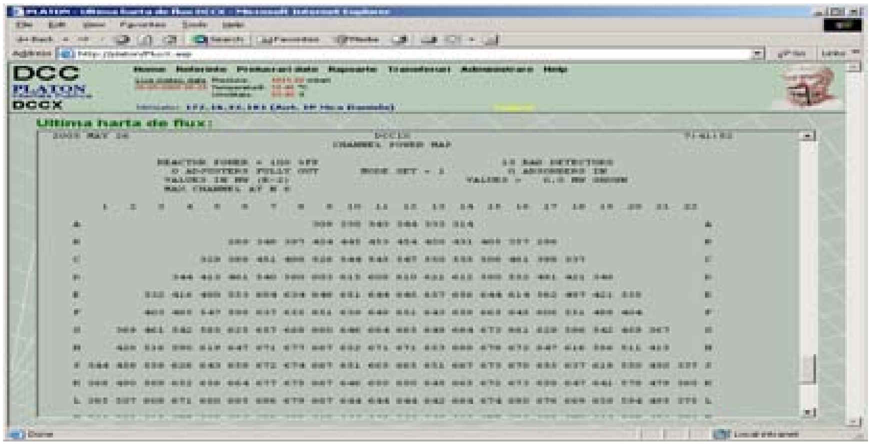

FIG. 2. Flux mapping printout.

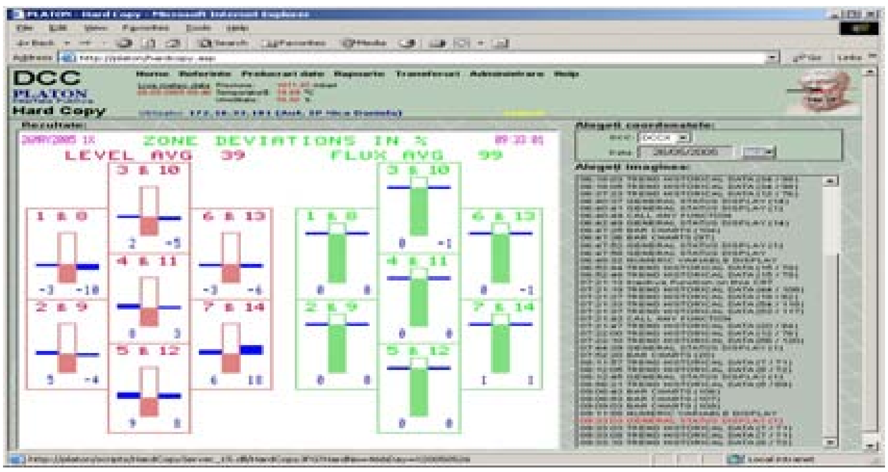

FIG. 3 Flux mapping printout.

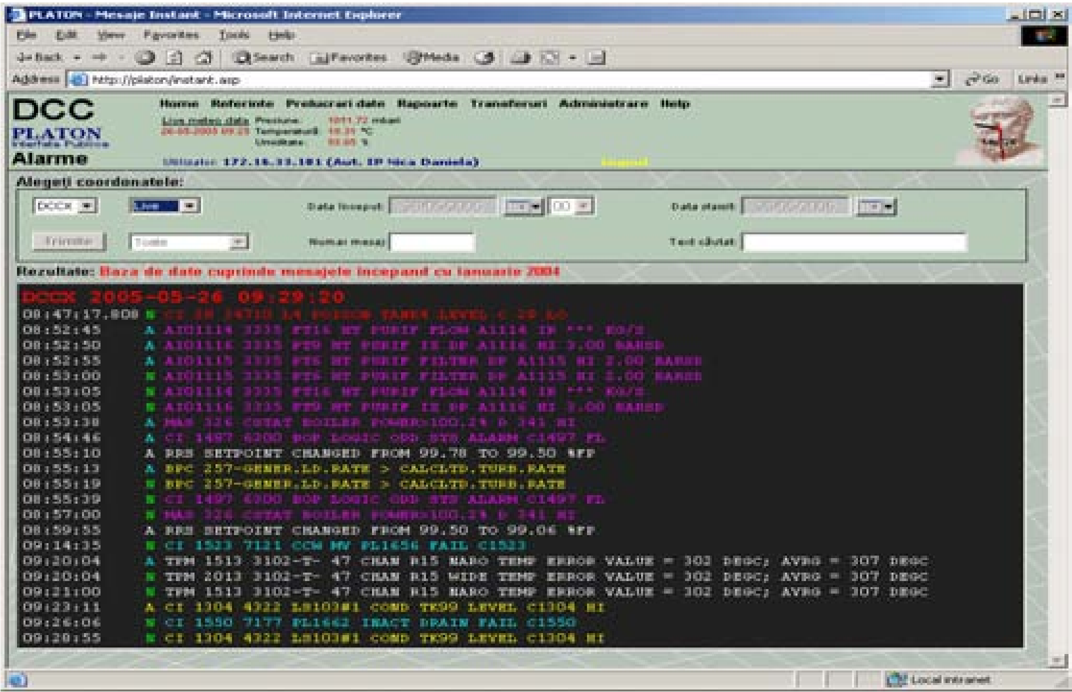

FIG. 4. Alarm pages.

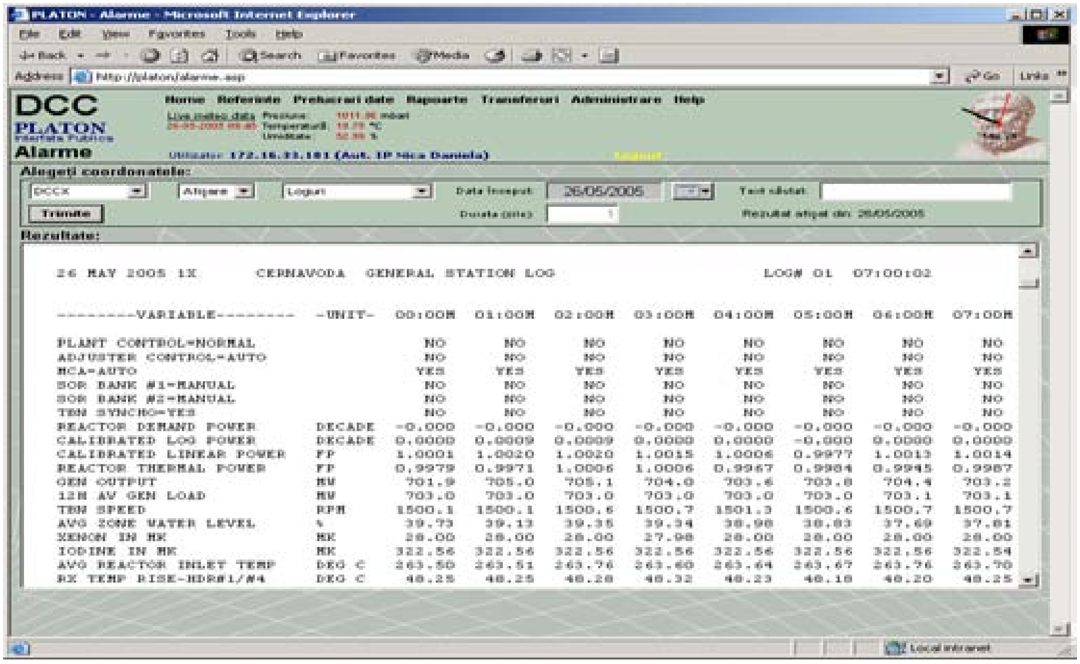

FIG. 5. Station logs.

PLATON was designed by Cernavoda NPP computer staff. During project development they used in-house built software programmes. Their development was performed under existing Control Computers Quality Management procedures. Few components were externally ordered.

To build **PLATON,** different software technologies (OPC, Data Socket) and programming systems (Varian assembler, Delphi, DSC Lab View, Java, and ASP) were used.

Figure 6 shows the way that **PLATON** extracts information from the plant systems and displays it for users.

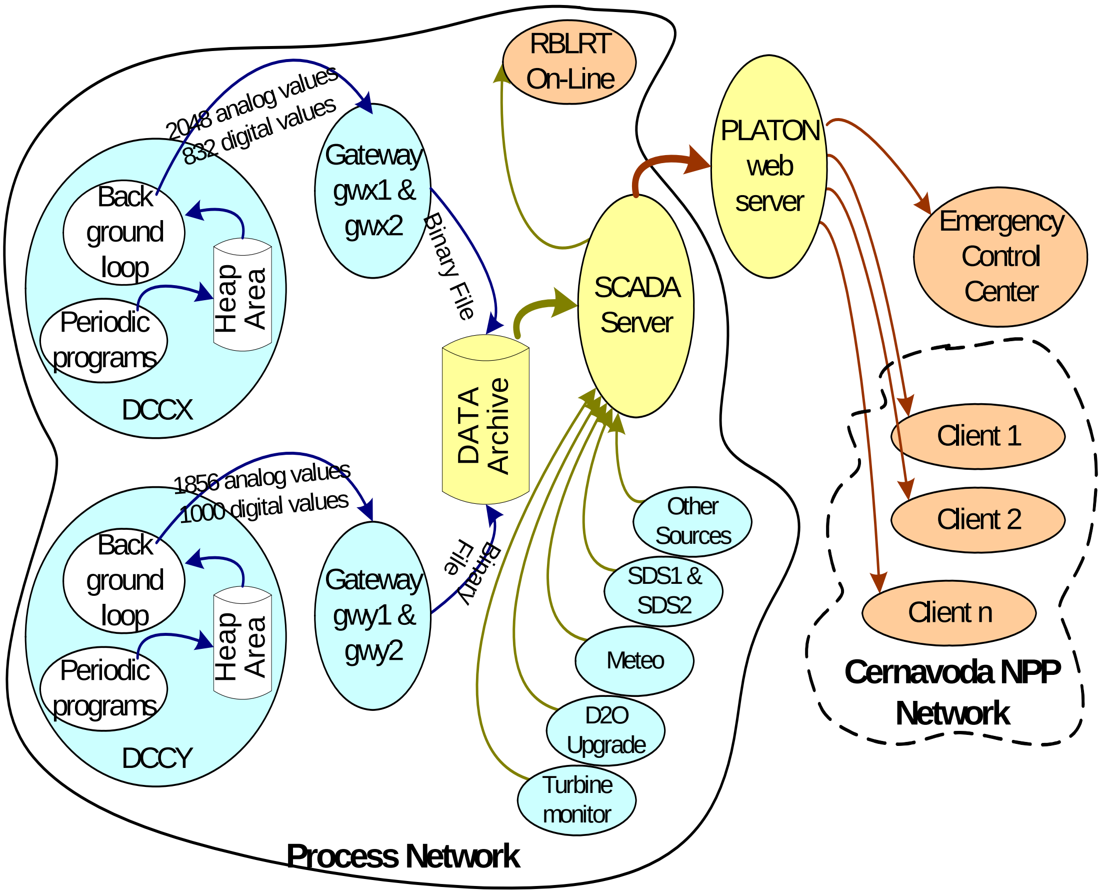

FIG. 6. Display of plant system information.

PLATON provides the following benefits:

- Quality of technical work dramatically increased after system was commissioned.
- Statistically, between 40 and 100 people use **PLATON** daily for current information needs.
- Manpower savings and added knowledge allowed a complete return on investment before the system was completely deployed.
- This system proved to be a crucial tool in analysing transients; a unique facility for fast capturing and reporting real systems behaviour during the event.
- After Unit 2 Commissioning, management requested extension of the system into the new plant. As result, computer staff will deploy the ʻCernavoda Unified Plant Information Systemʼ, covering both units. This will allow expertise sharing between the two units, systems functions optimization and cost reductions.

2. Single point of vulnerability project Cernavoda NPP

Due to a numerous events and plant shutdown caused by component failures, Cernavoda NPP management decided to pay more attention to the critical systems and critical components. Based on this decision, a project has been started to improve the plant reliability. The project was called ‘Single Point of Vulnerability’ (SPV). A team was created to initiate and develop the SPV project. This team established the goals of the SPV project and the steps they had to follow to complete the project.

The goals and the steps are summarized below.

2.1. SPV initiative goals

- The main goal of the SPV initiative is to reduce to as low a level as possible the vulnerability of the plant to events as a result of failures of equipment representing ‘Single Point of Vulnerability’, (hereinafter called SPV). It should be noted that this initiative is similar to other on-going initiatives from U.S. and Canadian NPPs.
- In addition, the experience gained during this initiative development will be used to improve the plant preventive maintenance programme (PM).

2.2. SPV initiative summary of main steps

2.2.1. SPV identification

- Establish criteria for equipment selection as a SPV. The equipment boundaries should also be defined.
- Organize an expert-panel meeting with responsible system engineer, Operation and Safety & Licensing, and select the SPVs.

2.2.2. Establish necessary PM tasks for SPV

- Establish failure mechanisms for the selected SPV and select maintenance tasks to prevent SPV failures. PM templates for SPVs similar to the EPRI templates should be issued. During the process, random degradation mechanisms that cannot be prevented by a reasonable PM programme should also be identified.
- Compare the existing PM programme for the SPV with previously defined PM templates and establish the gaps. Calculate SPV vulnerability to identify the sensitive areas.

2.2.3. Determine the gap and required actions

- Based on vulnerability results, establish priorities for necessary actions to reduce the gap between existing PM programmes and defined PM templates.
- The work will be processed according to plant procedures (assess the required tasks, issue working procedures, ensure personnel training/materials/tools/spare parts, execute the work in field providing feedback for ‘as-found’ condition).

2.2.4. Perform feedback analysis and establish future actions

- Analyse the SPV as-found condition and establish if supplementary actions are necessary.
- Random failure modes that cannot be addressed by a reasonable PM programme will be also addressed in the analysis.

2.3. SPV identification

The first set of criteria to be used for ‘Single Point of Vulnerability’ type equipment was defined in 2005. Due to some inconsistent criteria used in the first stage of the initiative, about 1000 ‘Essential Equipment’ items were initially selected.

Plant management later decided to upgrade the selection criteria using information obtained from WANO and benchmarking of similar initiatives at other plants.

Currently, the SPVs are defined as:

> Dynamic equipment or assembly that, if it failed to perform its mission or became unavailable and the failure or unavailability cannot be mitigated by control action, would cause:

- Reactor trip (automatic or manual) due to special safety system impairment 1/2 or unavailability of a system with a safety function;
- Turbine/Generator trip (automatic or manual).

The SPV boundaries were established according to EPRI guidelines for generic types of equipment or based on responsible system engineer engineering judgement for special types of equipment.

The SPVs were identified for Unit 1 via the expert-panel method. The activity was coordinated by Technical Division, and 336 SPVs were identified for Unit 1. A similar analysis was later performed by Technical Division for Unit 2, where 356 SPVs were identified.

Given that operating equipment in Unit 1 is more than twelve-years old and ageing effects could affect support equipment such as power supply sources, printed circuit boards, etc, it was deemed appropriate to extend the analysis and include in the Unit 1 SPV list the supporting equipment for Unit 1 SPVs. The supporting equipment for Unit 1 SPVs was identified and the impact of their failure was analyzed. New SPVs from supporting equipment were identified and included into the SPV list. For Unit 1, 728 support equipment SPVs were identified.

A special actions to identify support equipment SPVs in Unit 2 are not necessary at this time because their failure, due to ageing effects, is not expected for several of years. Such actions will be included in the on-going functional failure mode and effect analysis for critical systems.

At this time the final list of SPVs contains 1420 items. Going forward, this list will be updated in accordance with a new procedure issued for this activity.

A special code (code E) was also introduced into the Master Equipment List (MEL) to allow immediate identification of SPV equipment.

It should be mentioned that on-going projects from other plants are considering for inclusion in the SPV list if the equipment is defined as dynamic equipment or an assembly that, if it failed to perform its mission or became unavailable, and the failure or unavailability cannot be mitigated by control action, would cause a power decrease by more than 5-10%.

2.4. Establish necessary PM tasks for SPV

The owner for each SPV was identified in the final list:

- The component engineers for generic types of SPV (AOV, MOV, NV etc);
- The responsible system engineers for special types of SPV.

For the initial 336 SPVs identified in Unit 1, the following activities have been done:

- The failure mechanisms for each type of SPV were identified by component owners using the following sources of information:
- EPRI maintenance basis guidelines;
- Manufacturers/suppliers maintenance manuals;
- Internal or Industry OPEX.
- Based on the failure-mechanisms analysis, the necessary PM tasks were identified and the existing maintenance-programme requirements were revised accordingly. The results were documented in PM templates included into a document.
- SPV vulnerability calculations were performed in the initial phase of this initiative using the EPRI PM Database application. The results were later considered to be not relevant because the initial assumption was that the existing PM programme requirements for SPV were implemented as

initially planned. This assumption was questionable when a WANO peer review revealed a lack of PM programme execution control. The WANO conclusion was confirmed by the 'as-found' data collected during work execution on SPVs during 2006 planned outage. The 'as-found' condition of SPVs was inconsistent when compared with the existing PM programme requirements.

- It should be noted that the PM programme requirements for SPVs were developed to cover as much as possible all the failure mechanism identified by the analysis.

Similar activities will be performed for the remaining SPVs.

The PM programme can be optimised in the future considering FMEA analysis, which will determine what SPV failures could impact system operation. Based on the possible impact of various defects on SPVs, the significant failure mechanisms can be determined and PM tasks with low added value in preventing such failures will be eliminated.

Another significant source for SPV PM programme optimization is the analysis of 'as-found' data collected from performed work. The implementation of an 'as-found' data collection and analysis process was also recommended by WANO and EPRI experts.

2.5. Determine the gap and required actions

Because the vulnerability analysis results obtained using the EPRI 6.0 MDB application performed for the 336 SPVs from Unit 1 revealed inconsistent results due to a lack of proper control of PM programme execution, the maintenance history was checked for each of the first batch of 336 SPVs to identify if the required PM tasks were already executed. The results were documented in PM templates and also included in a document.

The following results were obtained when the existing maintenance records were compared with the PM revised requirements for the 336 SPVs from Unit 1:

- For 90 SPVs, the required PM tasks are delayed;
- For 185 SPVs, the required PM tasks should be scheduled no later than the 2008 planned outage;
- For 61 remaining SPVs there are no PM tasks required before 2009.

The significant volume of PM activities for SPVs required for the Unit 1, 2008 planned outage was generated mainly by lack of approved call-ups/predefines, with some contribution from deferred activities.

All the PM tasks and their required execution dates were sent to the Work Control Department. All the PM tasks delayed or required were scheduled no later than the Unit 1, 2008 planned outage and were already performed in that outage. Work Control used special a code to track required SPV PM tasks to completion.

For the 336 SPVs from Unit 1, 585 tasks/activities were included in the Unit 1, 2008 planned outage. During the outage work preparation, some problems related to the SPV spare-parts inventories were discovered. A special code was allocated to easily identify and track the spare parts for SPVs. The identified problems were addressed but the corrective actions required significant effort from the component engineering group.

In the future, the requirements for spare-parts inventories will be revised to ensure proper support for all SPV PM Programmes. This action will be coordinated with the initiative to compile the Unit 1 and Unit 2 spare parts databases.

A procedure was issued for vulnerability calculation and after a field trial period this procedure will be revised.

2.6. Perform feedback analysis and establish future actions

The procedure for 'as-found' data collection was issued and some data were already received from work performed during the Unit 1, 2006 planned outage and more recently from the Unit 1, 2008 planned outage. The preliminary analysis already discovered some areas for improvement (e.g. the status of SPV internals is generally better than expected; some of the BOP valves suspected for internal erosion /corrosion were discovered better than expected).

Future actions are required to use the 'as-found' data to improve the PM programme for SPVs (modify the activities frequencies, add/remove PM tasks, etc).

Random degradation mechanisms for generic SPVs were already identified during the initial phase. During a WANO TSM held in December 2006, as support for the SPV project, the experts from North America presented the graded approaches used by US stations to address these problems.

The solutions for addressing SPV random-degradation mechanisms should include:

- Analysing the possibility of increasing margins on trips due to failure of the SPV;
- Analysing the possibility of changing equipment operating alignment or changing operating procedures;
- Implementing modifications to create redundancy and eliminate the 'single point of vulnerability'.

A document was issued to describe in more detail the random mechanisms for generic SPV failures and required actions.

3. Other proposals to improve the SPV initiative

During benchmarking missions performed by plant personnel, other actions to improve the SPV initiative were identified. These improvements are mainly related to administrative controls to reduce possible human errors that may impact SPV operation:

- The work orders on or near a SPV should be easily identified by operation and maintenance personnel. Other plants have implemented some specific mark-ups for work orders on SPVs. SPV vicinities were also clearly identified and special controls were in place for access control into these areas;
- Specific error-prevention tools were required in other plants for work orders on SPVs or requiring access into SPV vicinities (pre-job briefing conducted by work group supervisor, clearance approved by shift supervisor, etc).

4. Conclusions

The project covered both units from Cernavoda NPP, including Unit 2 which was commissioned in 2007. The team identified approximately 140 critical systems, 8500 critical equipment items and 1064 SPVs, 336 process equipment items and 728 support equipment items for Unit 1. It further identified 10000 critical equipment items for Unit 2 and 356 SPVs for process equipment items alone. The support equipment SPVs will be identified until the 2009 Unit 2 planned outage.

For all the SPVs identified, preventive maintenance programmes have been developed and spare parts have been assured. Based on the project results, 70% of the Unit 1 SPVs have been maintained. The others will be maintained in 2009.

The SPV programme will continue for both units to complete the identification of SPVs and to revise the PM tasks. The SPV programme will be updated in the future based on 'as-found' resolutions and operation experience. Since we applied the system, Cernavoda has not had new events or unplanned outages caused by equipment failure.

REVIEW OF CANDU PLANT PERFORMANCE

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Abstract

Benchmarking is a useful tool for improving performance. To assist member in benchmarking their performance, the CANDU Owners's Group (COG) monitors the worldwide performance of CANDU type reactors, preparing various reports and presentations for use by COG members. This paper provides a synopsis of information included in the COG report CANDU Operating Experience Annual Report — 2007 Events.

1. Overall fleet performance

As shown in Figures 1 and 2, a number of CANDU type reactors have achieved average load factors in the upper and second quartiles of world performance. The CANDU 6 plants are particularly notable, having four in the upper quartile and three in the second quartile out of a total CANDU 6 fleet of nine. There are also good improvement trends across the fleet of CANDU type reactors, particularly in the area of outage control. Areas for improvement, however, remain.

As shown in Figures 3 and 4, the Unplanned Capacity Loss Factor (UCLF) and forced Loss Rate (FLR) for the CANDU type fleet exceed the world average and INPO targets. These are key areas for potential improvement. The relatively high FLR is largely attributed to poor equipment performance.

2. Causal factor analysis — FLR

As a tool to assist its members, COG prepares an annual operating-experience (OPEX) report summarizing and analysing performance. The most recent report[1] provides some insights into the causal factors behind the observed FLR of the CANDU type fleet.

Figure 5 shows the distribution of direct causes for forced losses during 2006 and 2007. Of these, equipment deficiencies – be they mechanical, electrical or I&C – were the dominant contributors. The importance of equipment performance is also shown by the causal factor breakdown shown in Figure 6, and deserves closer analysis.

Figures 7 and 8 show the distribution of FLR causal factors for all equipment and for nuclear systems, respectively. These distributions indicate that:

- Most of these factors have a human element,
- Ageing and not correcting known problems are significant factors,
- Nuclear systems are increasing contributors, especially the HTS,
- Fuel handling is improving,
- I&C is generally improving.

3. COG initiatives on key focus areas

COG actively supports and promotes a variety of activities to assist members to improve performance. Some of these are listed below.

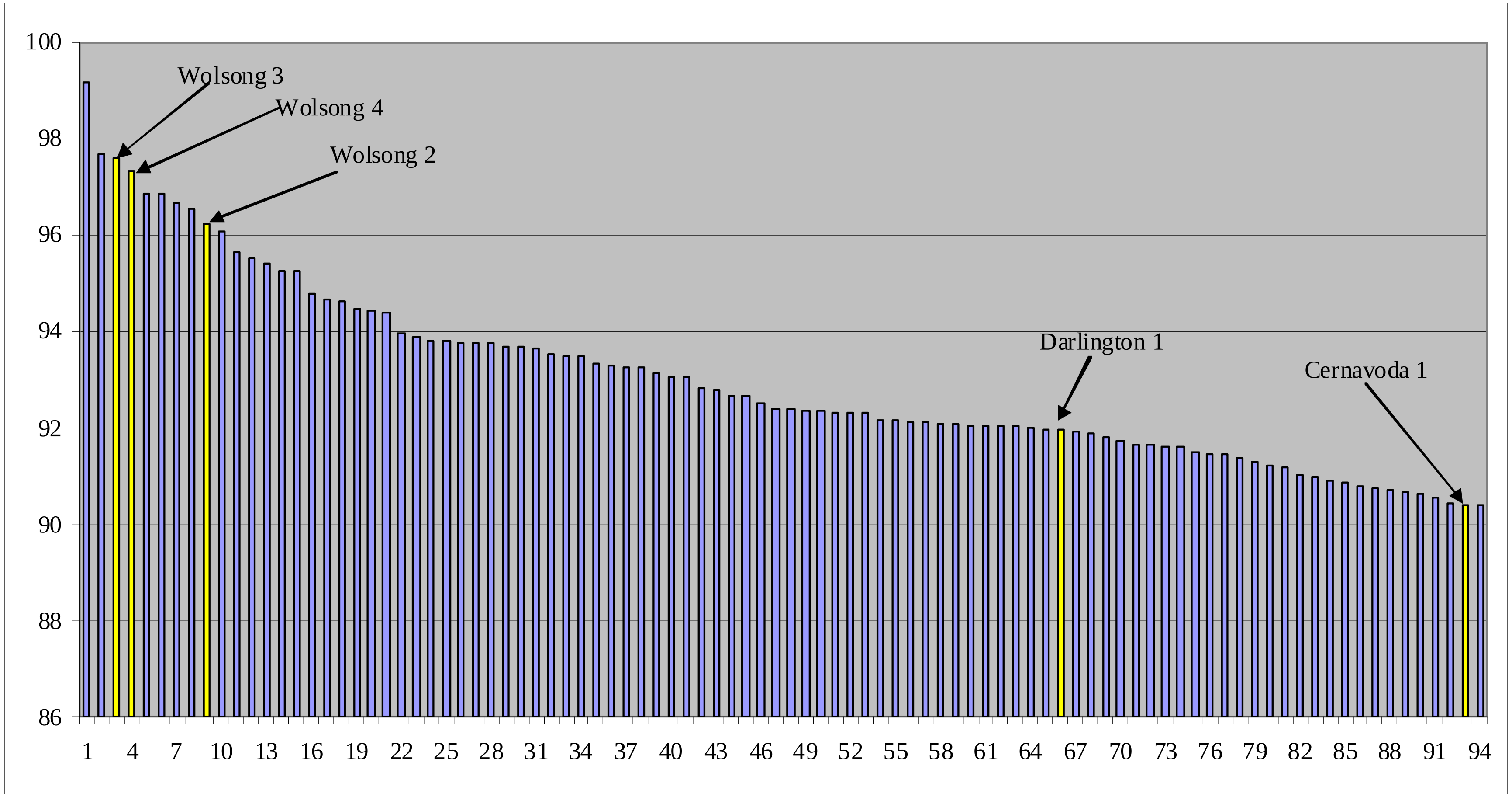

FIG. 1. Three year average load factor — (2005-2007) upper quartile.

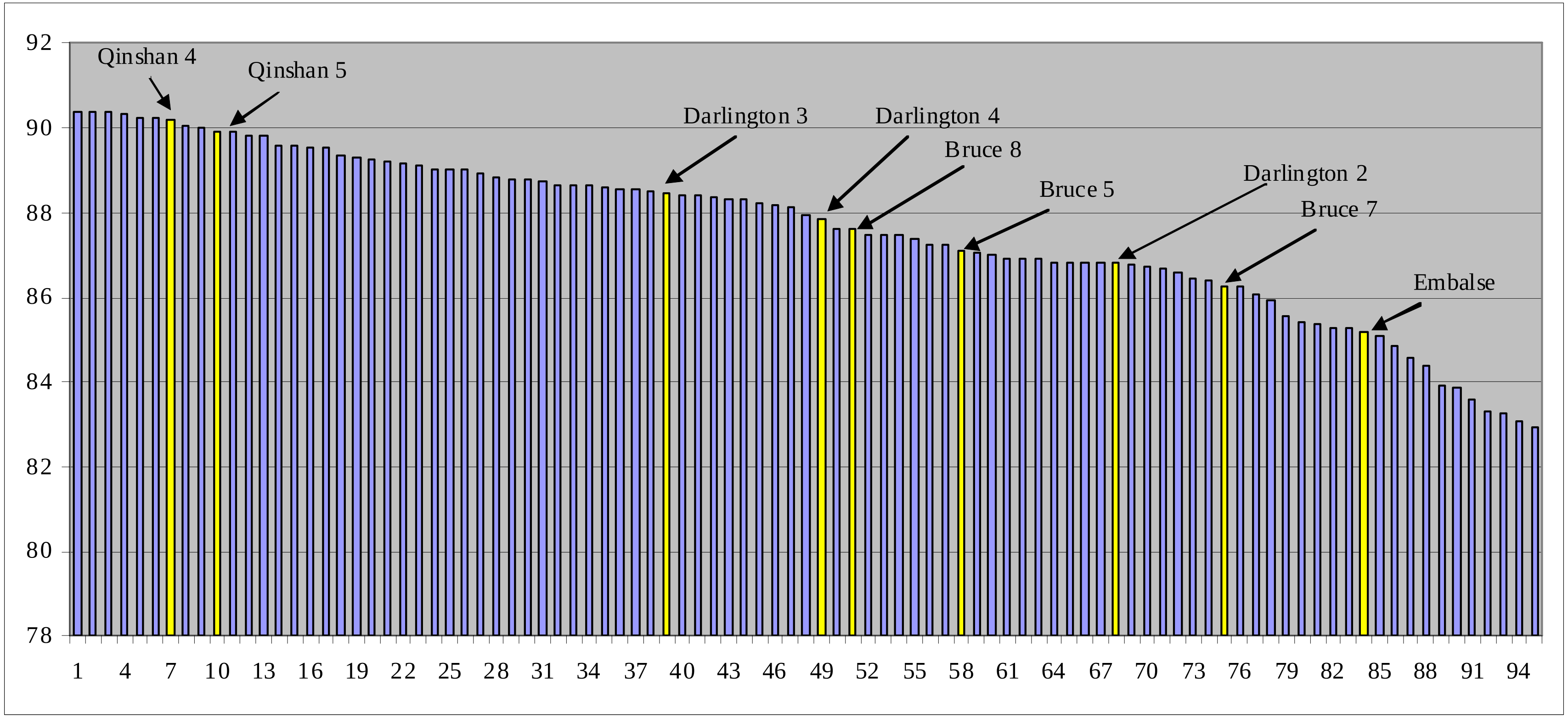

FIG. 2. Three year average load factor — (2005-2007) second quartile.

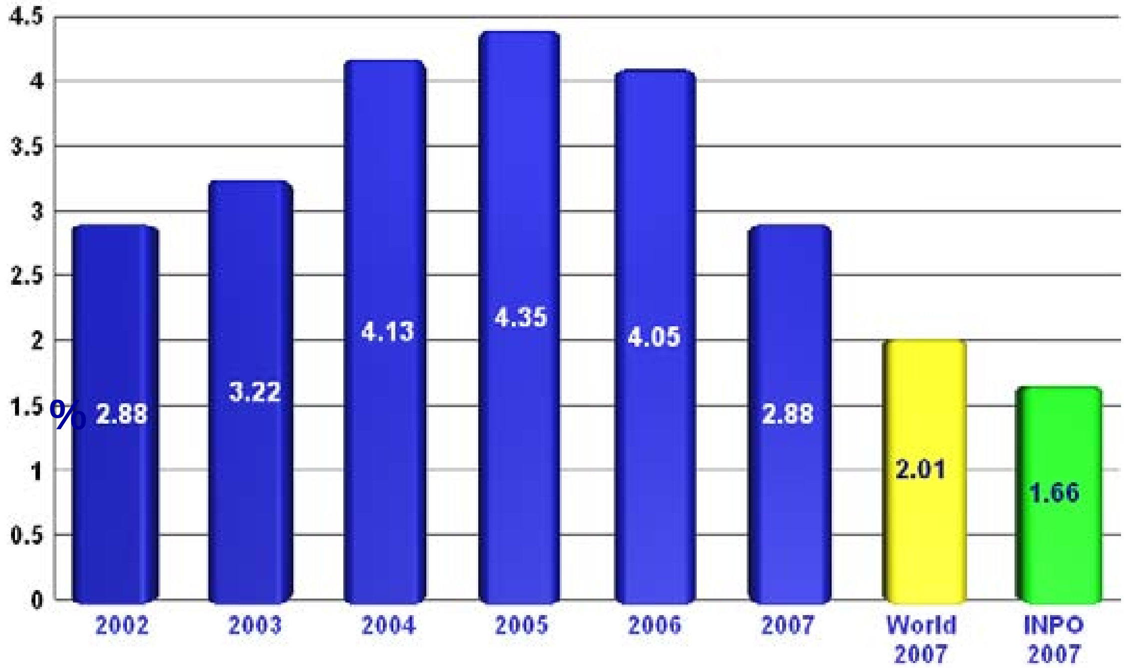

FIG. 3. Unplanned capability loss factor for CANDU type reactors (median).NOTE: Excludes India/Pakistan

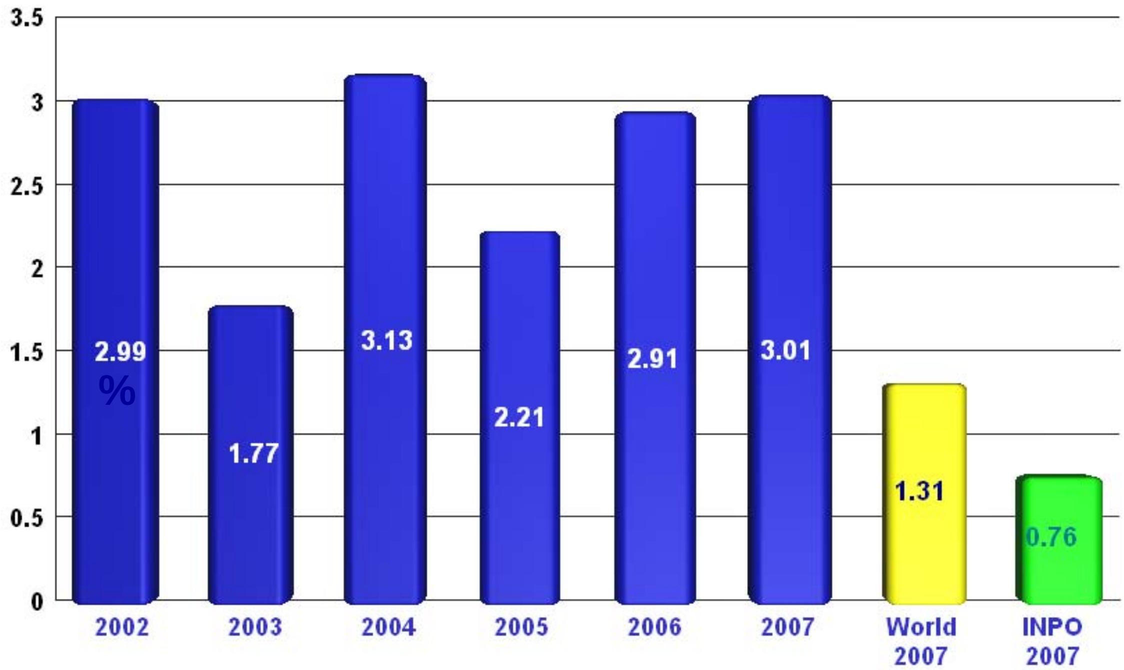

FIG. 4. Forced loss rate for CANDU type reactors (median).

NOTE: Excludes India/Pakistan

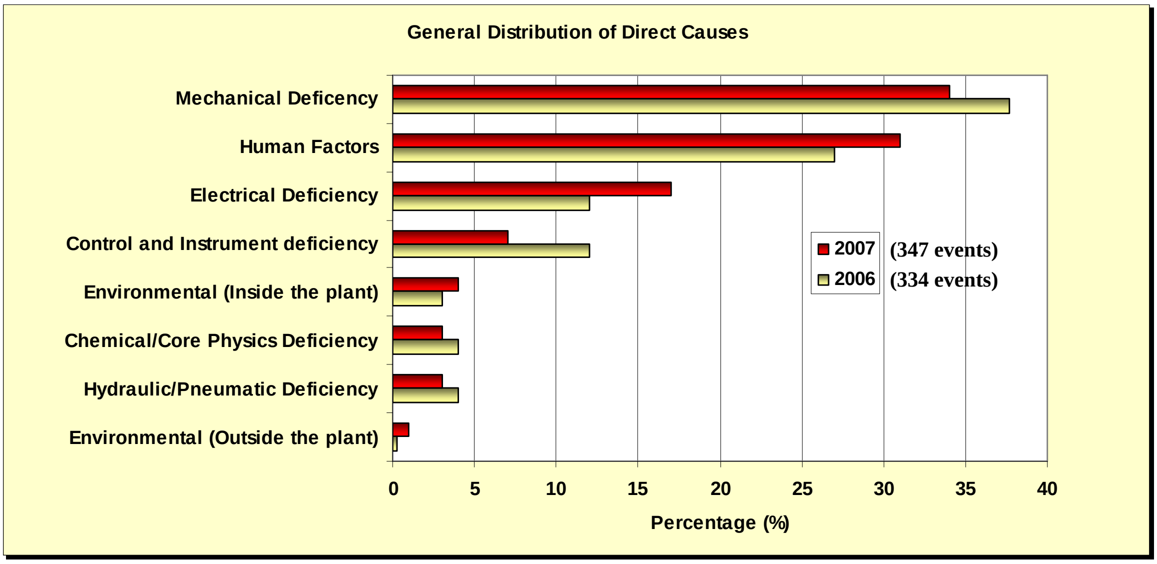

FIG. 5. Forced loss rate – direct causes.

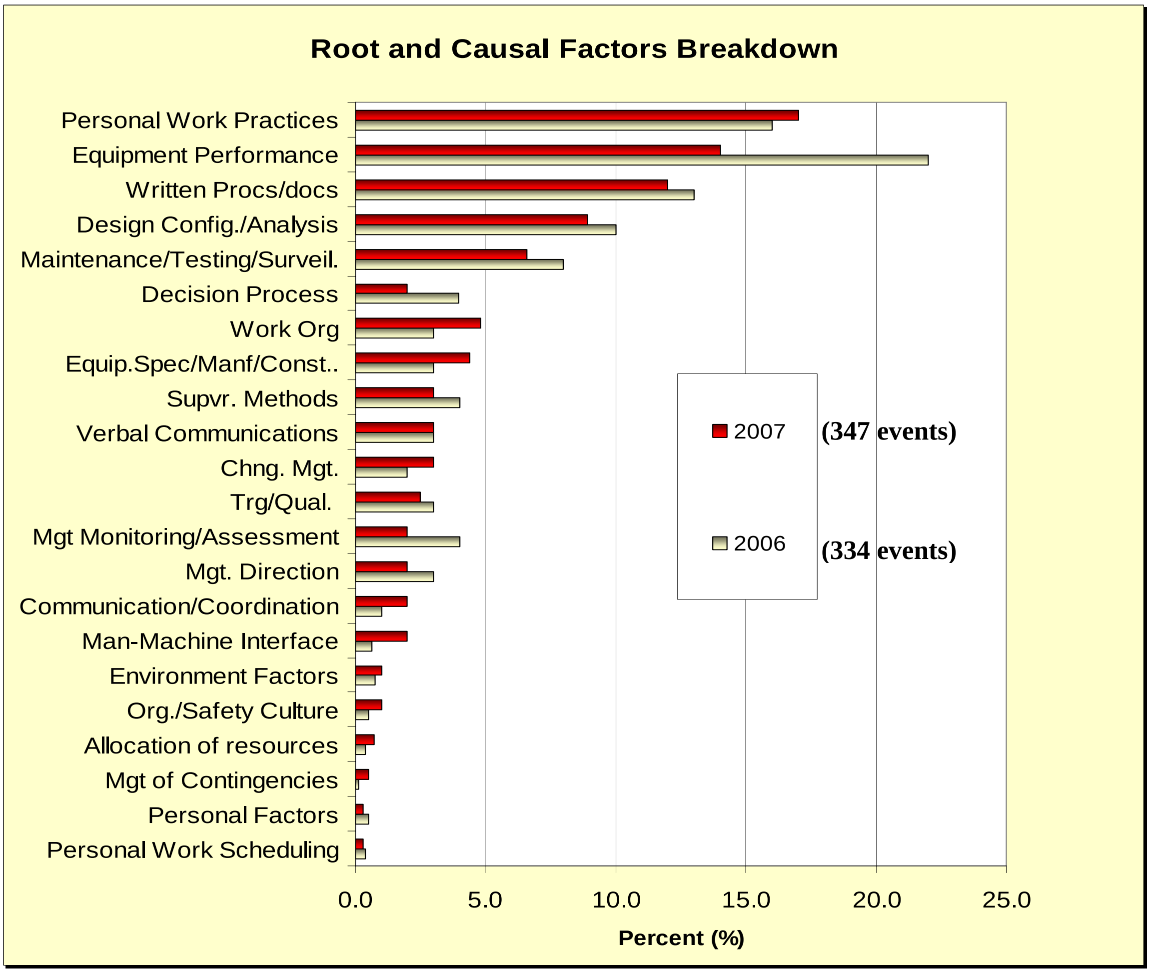

FIG. 6. Forced loss rate – causal factors.

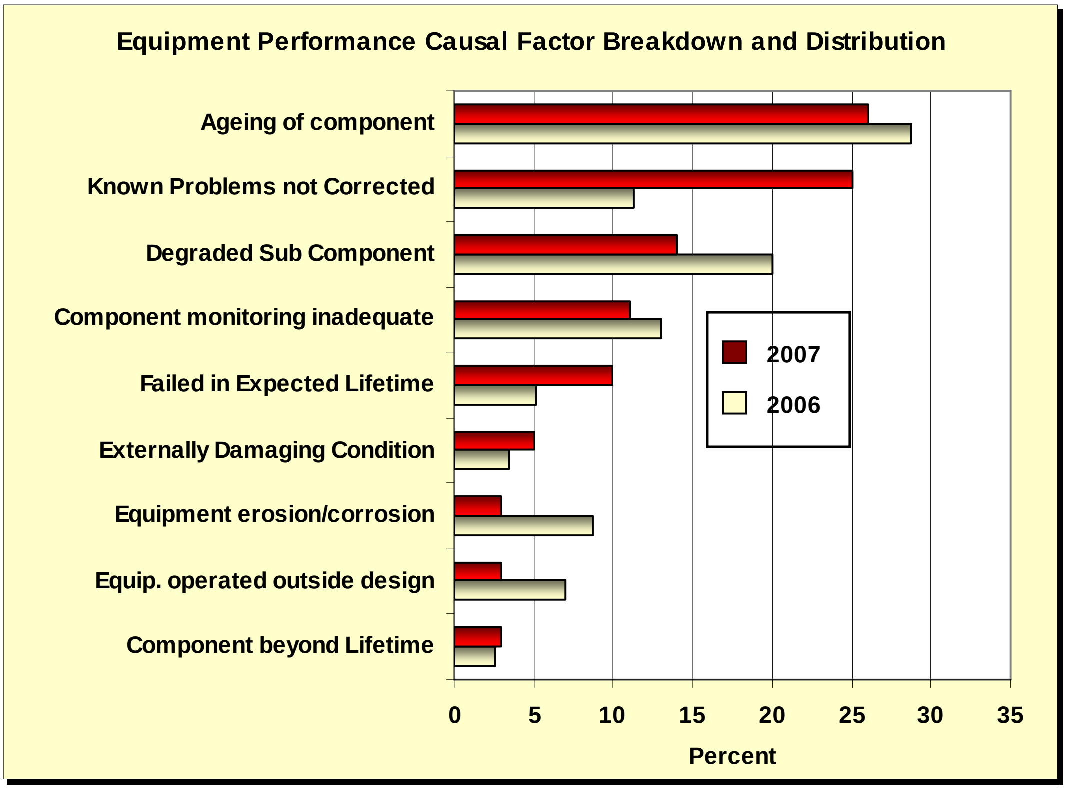

FIG. 7. Forced loss rate – equipment performance causal factors.

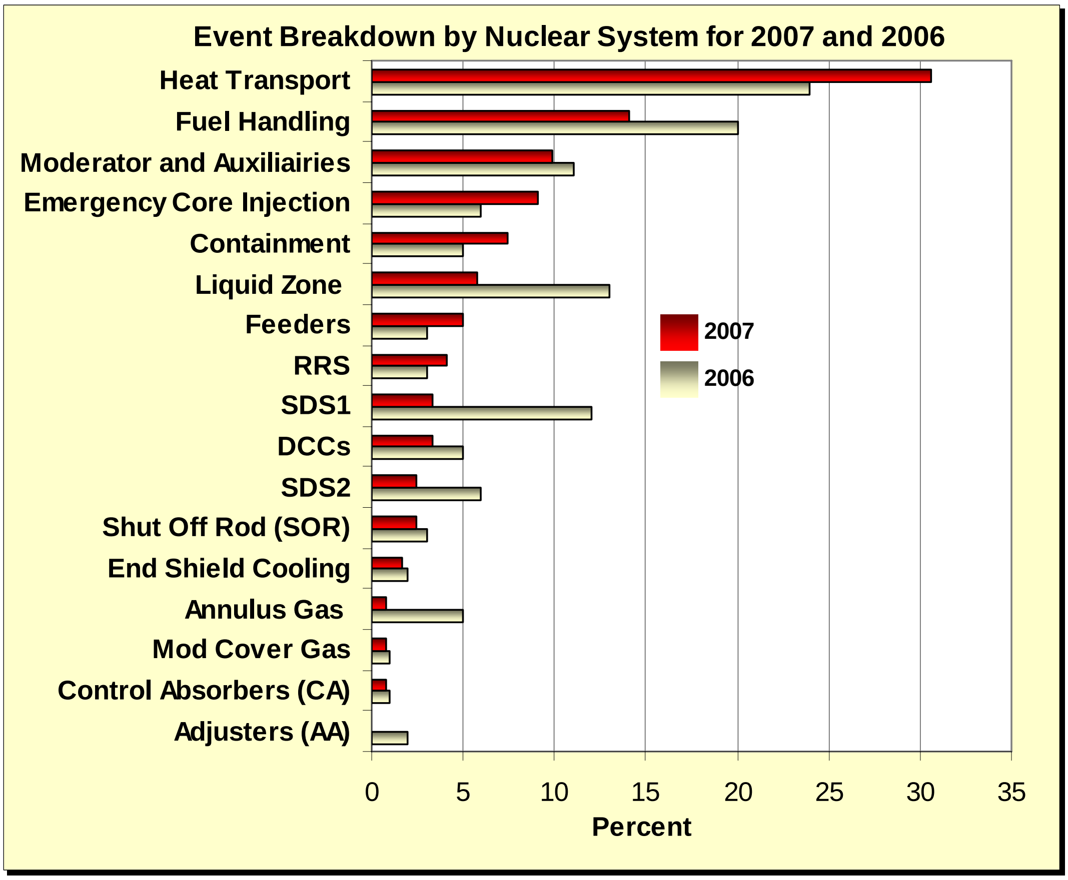

FIG. 8. Forced loss rate – equipment performance causal factors for nuclear system

3.1. *Materials and equipment performance*

- Critical equipment failure analysis and alignment of the maintenance strategies required to improve equipment reliability.
- A COG Joint Project is in place, forming a Equipment Reliability Group.
- A need for preventative maintenance strategy improvements has been identified.
- A COG Equipment Reliability Guideline has been developed, including an equipment-reliability index.
- The Group is reviewing ageing management initiatives at all participating sites.
- Critical spares remain an issue across participating sites, although there have been, some successful COG projects to improve the situation.
- COG Liquid Zone Best Practice Team (2008)
- A cross-functional team that includes representatives from engineering, operations and maintenance has been formed to develop a best-practice guide.
- A System Health Guideline has been developed via the COG System Engineering Working Group.
- fuel handling
- A COG Joint Project was recently completed to benchmark and identify good practices.[1]
- Through COG, members are evaluating a common approach to buying parts.
- Through COG, members are evaluating common standards such as for foreign material exclusion from irradiated fuel bays (spent fuel bays)IFB FME
- Follow-up actions are being transferred to the ongoing COG Fuel Handing Working Group.

COG has also identified specific plant systems that have a large impact on transients, the FLR and/or OPEX for which there are no current working groups. COG has recommended to its members that Working Groups be formed for:

- Nuclear mechanical systems, and
- Balance-of-plant (BOP) systems.

3.2. *Human performance*

- The COG Human Performance Working Group is in place.
- The group is developing a strategy for interacting with other COG working groups.
- Coordinated INPO delivered training.
- The COG Reactivity Management Working Group is in place.

[1] See paper by P. Lafreniere in this same session.

- The group is benchmarking, developing common practices and indicators to drive improvements in performance.
- COG holds OPEX weekly screening meetings.
- Human Performance Newsgroups focus.
- OPEX Annual and Quarterly Trend Reports with Causal Factor Analysis.
- The COG Corrective Action Working Group is in place.
- This was recently restarted as a working group.
- The group is developing a strategy for improving root causes.
- The group involves all COG members and INPO.

4. Conclusions

This review of CANDU type performance confirms that CANDU type reactors can deliver best-in-world performance, but utilities need to focus on forced loss factors (FLR) and unplanned capacity loss factors (UCLF). In particular:

- The FLR is generally higher than the nuclear-industry average;
- Utilities need to continue Causal Factor analysis and implement meaningful Corrective Actions;
- Common approaches have shown benefits (liquid zone control, reactivity management, human performance working groups);
- There is a need to expand working groups into nuclear systems and BOP areas (EPRI does this well);
- There is a need to address long-standing problems; and
- There is a need to address issues in improved designs.

REFERENCES

[1] CANDU OWNERS' GROUP, "CANDU Operating Experience Annual Report — 2007 Events", Toronto; 2008).

REVIEW OF THE PERFORMANCE AND BEST PRACTICES OF THE WOLSONG NUCLEAR POWER STATIONS

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Abstract

KHNP's stations have a long track record of impressive performance, and are currently one of the top five reactors worldwide in terms of capacity factor. Their success can be traced back to the company goal of 'one cycle trouble free.' This goal has grown into a working culture and has permeated throughout the entire organization. With this strong focus, the initiatives in maintenance, operations, task management, information management, worker safety and research & development all contribute to improving plant performance, reliability and economics.

1. Introduction

Korea Hydro and Nuclear Power (KHNP) is a world leader in nuclear power plant operation. KHNP's stations have a long track record of impressive performance, and are currently one of the top five reactors worldwide in terms of capacity factor [1]. The Wolsong units, which are PHWRs of CANDU design, have been consistently represented in the top ranks of the Korean NPP fleet and have habitually achieved excellent capacity factors. This paper reviews public-domain publications on their programmes for achieving this performance.

Korea's operational successes stem from the high-level goal, adopted company-wide, of 'one cycle trouble free (OCTF) [2].' Realizing this target involves continuous operation of the units between planned maintenance outages, with no forced outages. The OCTF goal is coherent with KHNP's mandate to provide a stable, economic supply of electricity to Korean residents. Their successes with OCTF have had a very positive impact on the national economy and their performance is world-renowned.

Though it may seem a daunting task, the station staff have consistently achieved the OCTF endpoint, several times in succession; Wolsong-1 has achieved three consecutive cycles of OCTF. Notably, KHNP has developed several initiatives that are focused on improving plant performance. The holistic approach taken to achieve OCTF across the entire company is evident from examination of these initiatives:

- Forced outage reduction programmes (human and equipment),
- Enhancing the quality of the facilities through good maintenance practices and implementation of a comprehensive maintenance-management programme,
- Effective management of lessons learned and operational experience, and
- Research and development initiatives that feed in to the longer-term goals to upgrade equipment and to improve plant efficiency.

2. Forced outage reduction programmes

2.1. Human factors studies and developments

In 1997, KEPCO funded the development of human performance enhancement systems and human evaluations systems to aid in reducing the number of reactor trips due to human-error [1,3]. The Korean Electric Power Research Institute, KEPRI, developed the CAS-HPES (computer aided system for human performance enhancement systems) as part of this initiative. The CAS-HPES (now called K-HPES for Korean-HPES) is a browseable database of company procedures, and was specifically designed for an operating end-user. The system has undergone validation tests to ensure that the CAS-HPES is functional, efficient and user-friendly, ensuring that the software is specifically geared towards minimization of human errors.

An additional KEPRI activity was the analysis of worker safety [3]. To gain insight into the worker safety culture, the station employees were surveyed on their daily routines and were asked about their thoughts on workplace safety. Analysis of the surveys aided in identifying further areas of future safety- and human factors-related research.

Standardization of comprehensive worker training courses and education programmes is the most direct method of preventing human errors [3]. There is a well-defined training programme at Wolsong where all operators are trained in a simulated environment. A training facility was first constructed at the Kori site in 1977 to provide a facility for worker training courses. Employees are continually educated through online courses and through participation in workshops and seminars.

2.2. Minimizing equipment failure trips

There is a strong focus at the Wolsong plants on minimizing the number of forced outages where the root cause was equipment failure [2,4,5]. KHNP has approached this challenge from several angles, and the success of this initiative has had a significant contribution to achieving OCTF. Prominent equipment failure issues were largely resolved by a variety of complementary methods:

- additional on-line monitoring capability for critical equipment
- predictive maintenance programme
- continuous upgrading of equipment

2.2.1. Additional on-line monitoring capability

On-line monitoring functionality was added for key equipment to ensure critical parameters were within the expected range [2]. Temporary instrumentation and monitoring equipment provide additional real-time and historical information, such as temperature transients on critical piping or vibration data for the turbines. Temporary sensors are connected to terminals in the department offices, and the information can be relayed to the DCC computers through a common gateway. This allows the operator to compare current trends with the historical data stored in log files, which in turn facilitates the identification of potential equipment risks.

2.2.2. Predictive maintenance programme

The predictive maintenance activities are carried out on essential plant equipment during scheduled planned maintenance. Predictive techniques are a key part of the risk-based equipment maintenance prioritization scheme. Predictive maintenance tasks, such as temperature and vibration analyses, are logged using cards attached to the machines. During these maintenance tasks, equipment is inspected for abnormal noises and checked for oil and lubricant replacement. Predictive maintenance, in addition to preventive maintenance, limits and ideally eliminates corrective maintenance practices. On top of applying preventive maintenance to all equipment at regular intervals (usually at intervals specified by the supplier), a more focussed effort can be made to maintain or fix the selected equipment based on its performance history and operating trends, hence avoiding a forced outage altogether.

2.2.3. Equipment upgrades

Plant performance is also improved by upgrading equipment to remove obsolescence or to add beneficial functionality [2,5]. For example, the air conditioner in Wolsong 1 was replaced with an improved A/C unit (ACU ; air conditioning unit) coupled with a desiccant bed; this upgrade had the added functionality of humidity control, which reduced the load on the D_2O recovery dryers.

As another example, all four Wolsong units upgraded their raw service water systems and recirculated cooling water systems. The upgrades added redundancy to the current design, allowing maintenance activities on isolated sections of either system without loss of cooling-water availability. Furthermore, as illustrated by this example, a benefit of having a multi-unit complex is that successes at one unit can be duplicated across all units. Replacements and upgrades can be installed sequentially to make the best use of lessons learned during installation and commissioning of upgrades.

3. Good maintenance practices and maintenance management

3.1. Operating ranges

A best practice adopted at the Wolsong units was to narrow the allowed operating ranges of some parameters [2]. The operating ranges have been tightened throughout the plant by placing narrower restrictions on chemistry control, allowable temperature and pressure oscillations, maintenance thresholds, etc. One example is the moderator design, which stipulates an acceptable operating pH range of 0.4 pH. KHNP tightened this range to 0.2 pH. Having tighter control over the systems is not an easy task, but the payoffs are that operating the plant closer to the intended operating design conditions will minimize unexpected plant behaviour and the plant will be running closer to its optimal conditions. These advantages result in increased plant stability, improved plant economics and enhanced thermal efficiency.

3.2. Efficient management of maintenance activities/reduced maintenance backlogs

Maintenance management is extremely well planned and organized [6]. All maintenance activities are planned, tracked, and closed-out through KHNP’s online maintenance-management system. The Digital Real-Time Enterprise Asset Management System, or DREAMS, is an efficient method of executing management plans since real-time progress is tracked against the plan. DREAMS was developed two months after KHNP became a separate entity from KEPCO in 2001. Since then, it has produced incredible economic benefits stemming from increased work-planning efficiency, reduced inventory, and a company-wide improvement in reliability of operations.

DREAMS also provides for the development of clear strategies [6]. Short-, medium- and long-term maintenance activities for each piece of equipment in the plant can be strategically planned and executed. It also enables management to plan ahead for changes in maintenance routines from season to season; due to changing humidity, precipitation levels and temperature conditions throughout the year, different maintenance strategies are adopted for optimal performance.

In addition to being very organized in maintenance planning, KHNP is very disciplined in completing maintenance activities [2]. This diligent work ethic has essentially eliminated any maintenance backlogs. If any discrepancies are reported during the operator walkdowns, they are discussed at the daily maintenance management meetings, and are classified based on their urgency and importance. The discrepancies are then pursued aggressively until the issue is resolved. This approach has kept the discrepancy backlog to less than 5%.

4. Lessons learned and OPEX

4.1. Management of operating experience

Proper management of operating experience is vital to ensuring that past errors are not repeated. There are several strategies that have been adopted at the Wolsong NPP to take advantage of the operation and maintenance experience accumulated across the units [2].

- When equipment maintenance is performed, it is recorded in DREAMS. This is to reduce human errors in the field and is part of KHNP's 'one shot' maintenance programme.

- A maintenance call-up list has been developed for each piece of equipment. Each piece of equipment has a call-up frequency for maintenance activities, which can vary from 2 weeks to a year.

- All discrepancy reports are reviewed every morning and are followed through to resolution. Discrepancy reports can even be filed for equipment that is operating within its specified range, if it is making irregular noises or has an irregular performance trend. For example, a line in the P&IC system for Wolsong 2 was making an unexpected level of noise and inspection showed high levels of vibration. Investigation revealed an abnormally high temperature in the P&IC system. The line was temporarily rerouted until the P&IC could be repaired to solve the temperature problem.

- Close monitoring of units can result in equipment upgrades where a system is not meeting the internal performance expectations. This can be the installation of new components to an existing system or upgrading an existing component to improve performance. For example, a pneumatic valve and a dew point monitor were installed on the drain line of the D_2O recovery dryers. The new behaviour of the drying system as it switches from regeneration to adsorption mode is to close the drain valve. This ensures that less D_2O escapes to the reactor building atmosphere.

4.2. Feedforward of OPEX

The OPEX and lessons learned at one unit are communicated to other units [3]. This is a significant advantage of operating multiple units. Notably, Wolsong-1 was put into service in 1983, fourteen years before Wolsong 2, 3, 4 went into service (in 1997, 1998 and 1999, respectively), so there was already a collection of OPEX to feed into those projects. Wolsong 2,3,4 were constructed one after the other, so construction OPEX was fed into subsequent projects.

In addition to PHWR OPEX from Wolsong, KHNP has three other sites (Kori, Ulchin and Yonggwang) housing PWRs. Good practices can be applied company wide, and some lessons learned in a PWR station may also be relevant to PHWR operations.

5. Research and development

The prominent research organizations for R&D in Korea are the Korean Atomic Energy Research Institute (KAERI) and the Korean Electrical Power Research Institute (KEPRI). KAERI was founded in 1959 and is involved in several facets of R&D[4]. Notably, they constructed the HANARO nuclear research reactor, based on Atomic Energy of Canada's MAPLE design. KEPRI was the central research centre for KEPCO, and provided R&D services to all areas of electric power generation. They have several nuclear power research groups, including one group entirely dedicated to CANDU technology [3].

These dedicated research groups are important contributors to the high capacity factors being achieved by KHNP. There are several projects by these groups that aim to reduce the number of forced shutdowns and to improve plant economics. Some projects that exemplify the contributions of R&D to achieving the OCTF goal are shown below.

- Human factors and performance assessments − Development of the CAS-HPES Computer Aiding System for Human Performance Enhancement Systems. This system is a robust and user-friendly database of operational procedures.

- Equipment performance surveillance − Furthering development of predictive maintenance techniques and technologies. For example, the proper diagnosis of a motor-operated valve requires disassembly of the unit. This sparked the research for non-invasive sensors that can quantify the forces acting on MOVs.

- Operations and maintenance − improving performance, reliability and economics through management of activities. Highly developed software prioritizes tasks and gives the history of previous maintenance activities.

- Development of basic design technology − Ongoing research in the fields of reactor physics for improved fuel cycles, component and safety system design improvement and life extension, and improved safety evaluation systems.

6. Summary and conclusions

KHNP is the world leader in operational performance of PHWRs and PWRs. Their success can be traced back to the company goal of ‘one cycle trouble free.’ This goal has grown into a working culture and has permeated throughout the entire organization. With this strong focus, the initiatives in maintenance, operations, task management, information management, worker safety and research & development all contribute to improving plant performance, reliability and economics.

REFERENCES

[1] CHUNG, B., *Nuclear Power Development in South Korea.* Nuclear News, volume 38, No. 8.1995.
Available via International Nuclear Safety Centre, URL: http://www.insc.anl.gov/cgi-bin/sql_interface?view=reference&qvar=id&qval=7

[2] CHUL-UN LEE, et.al., *Operational Performance of Wolsong CANDU Nuclear Power Plants.* Twenty-third Annual Conference of the Canadian Nuclear Society in Toronto, Canada. June 2-5, 2002.

[3] KOREA ELECTRIC POWER RESEARCH INSTITUTE, 1997 KEPRI Annual Report, *Nuclear Power Generation*, January 1998.

[4] LEE, S-Y. AND JUNG, S.H., *The Development of PHWR Technology at KAERI*. Joint KAIF/CAN CANDU Seminar in Seoul, Korea, April 4th, 1995

[5] JEON, J-P., *Overview of Nuclear Projects and CANDU Status in Korea*. Joint KAIF/CAN CANDU Seminar in Seoul, Korea, April 4th, 1995

[6] KOREA HYDRO & NUCLEAR POWER COMPANY, SAP Customer Success Story, URL: http://www.sap.com/usa/industries/utilities/pdf/CS_Korea_Hydro.pdf, February 2004.

SESSION IV. REDUCTION IN OPERATING AND MAINTENANCE COSTS

CLEANUP AND RECOVERY OF HIGH TOC (TOTAL ORGANIC CARBON) D_2O AT PICKERING NGS

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Abstract

A process was developed in-house using readily available components from local vendors, pumps and storage tanks no longer used by the station. This process proved successful in bringing the total organic carbon (TOC) content of the D_2O to an average of 3 ppm, which was then fed through the ion exchange cleanup system (IXCU) for final polishing and sent to the station upgrader as feed. It is estimated that this process has saved the station several million dollars in processing costs alone, based on an estimate from a contract company to clean up a small volume of the TOC contaminated D_2O. The estimated value of the recovered D_2O is approximately $54 million.

1. Summary

Feed to the heavy water upgraders must have a TOC level of less than three ppm as it has been found that high levels of TOC will damage the copper oxide on the upgrader packing causing distributor plate plugging and gradually reducing upgrader efficiency. If left unchecked, it has the potential to damage the upgrader to a point where it will be rendered inoperable, requiring replacement.

Pickering had been collecting high volumes of TOC-contaminated heavy water that the station did not have facilities or processes to upgrader feed specifications. A large volume of D_2O was contaminated by oil from fuelling machine operations and other contaminants, and was written off as an asset and left stranded.

During reactor feeder scanning operations a glycol based coupling solution was used, which was then picked up by the reactor building (R.B.) dryers and collected as dryer condensate. This condensate was collected in drums for processing in the station upgraders. Sampling of these drums revealed extremely high TOC levels that were found to be caused by the feeder scanning operations, specifically from the glycol-based coupling solution.

A process was developed in-house using readily available components from local vendors, pumps and storage tanks no longer used by the station. This process proved successful in bringing the TOC content of the D_2O to an average of 3 ppm, which was then fed through the ion exchange cleanup system (IXCU) for final polishing and sent to the station upgrader as feed. To date this process has cleaned up and recovered over 2 700 drums of TOC contaminated D_2O and recovered D_2O that was previously written off as a station asset.

The process was developed and assembled by Common Services department staff (operators, maintenance and engineering) and is a testament to innovative thinking and station management allowing their own staff the freedom to put together their own solutions to problems that they clearly know the intimately.

It is estimated that this process has saved the station several million dollars in processing costs alone, based on an estimate from a contract company to clean up a small volume of the TOC contaminated D_2O. The estimated value of the recovered D_2O is approximately $54 million.

The team was recognised by the company by winning the 2007 Ontario Power Generation ‘Power Within Award’ for business excellence in the nuclear generating division.

2. Background

Over 250 megagrams of downgraded D_2O were trapped in tanks located at Upgrader A due to excessively high TOC contamination of the water. This was a result of contamination from fuelling machine oils, solvents and other chemicals.

In addition to this, drumming of dryer condensate that was contaminated by the glycol solution used in the feeder thinning inspection programme continued without the station having any ability to process it. TOC values averaged 300 ppm but some of the values were as high as 1400 ppm. The station recognized that the feeder inspection operations were producing this high TOC condensate, but had no option other than to continue its use as questions concerning feeder elbow erosion had forced the shutdown of the Pickering A operating units and until it could be proven that they were sound for restart.

Over 2300 drums of TOC contaminated D_2O were located in various areas of the Pickering units. Poor control over the usage of drums at the station further exacerbated the problem. At one point the total station inventory of empty drums fell to twelve and other operational practices had to be used to prevent further drum usage.

This storage of full drums of downgraded and TOC contaminated D_2O was not only a housekeeping issue, but also morale issue. In addition, it contributed to unnecessary dose to station staff due to high radiation fields from other contaminants in the D_2O.

In addition to this, the only process the station had for oil removal was made inoperable when a vent line to one of the oily water tanks in IXCU became blocked and the tank collapsed when its inventory was emptied. This gave the plant no other option than drumming TOC contaminated water, which over time became normal practice for the plant.

It was realised that the rate of production of the drummed D_2O was becoming untenable. Something had to be done to alleviate the situation. A decision was made to invite bids from contract organisations for a process to clean the water to acceptable upgrader-feed specifications.

An external company placed a bid that would purify approximately 100 Mg of the original 250 Mg of D_2O trapped in the tanks at the Upgrader A. The process submitted was at face value very simple, using high efficiency bag filters, ion exchange columns and an ultraviolet unit for final cleanup of the D_2O. The unit was to be provided on a portable skid. Total value of the contract was to exceed \$640 000 plus overtime and other extras. After this cleanup, the equipment was to be removed and returned to the vendor, so it was basically a ‘one shot deal’.

The proposal was reviewed by Common Services staff and a group discussion revealed that they believed they could build a similar process in-house at a fraction of the cost. Management gave permission and the funding to purchase equipment. A small experimental process was fabricated and they found that it could successfully clean a small batch of the TOC contaminated D_2O to an acceptable level.

After careful review of the results, management gave the team permission to move forward on a much larger scale with the goal of cleaning not only the D_2O in the upgrader tanks, but to also process all of the drums accumulated on site.

The team then designed and built a train of bag filters, polythene settling tanks, pumps, charcoal filters and ion exchange columns, somewhat resembling the process submitted in the original contract bid. The total cost for purchase of the new equipment was approximately \$20 000.

The team consisted of operators, maintenance, and engineering staff working together as a team to put together a workable solution. Their innovation and creative thinking was admirable and the teamwork shown was second to none. All involved showed immense pride in their achievements and were rightly recognised by the company by receiving the 2007 'Power Within Award' for business excellence in the Nuclear division of Ontario Power Generation.

The system developed consists of two similar trains of high-efficiency bag filters that empty into polythene collection tanks. D_2O is taken from the bottom of the tanks and the oils and other contaminants not removed by the filters settle on the top of the polythene tanks for easy removal.

From the tanks the D_2O is passed through charcoal filters and ion exchange columns into a collection tank. This tank is part of a system that is no longer in use. When the tank is full it is recirculated through the final elements of charcoal and ion exchange columns until it reached an acceptable specification to be sent to IXCU for final polishing of the D_2O. The process can be seen as a batch process as staff wait for the final tank to be filled before recirculating the D_2O through the final filters and ion exchange columns.

It should be noted that Common Services staff have focussed on the result rather than defining the actual physical processes that allows the system to successfully remove the TOC.

Some scepticism that the process cannot physically remove all TOC existed and it is probable that the system will not remove all 'species' of TOC; however, it has proved to be extremely successful for Pickering.

It is likely that pump heat and vapours expelled from the process in the final tank headspace have contributed to the success of the system.

Staff have never claimed to have invented a new process; rather, through innovative thinking they have put together a system from readily available components and equipment to overcome an issue that was previously thought to be unsolvable.

Although somewhat time consuming, the process has successfully cleaned the D_2O trapped in the tanks and has recovered and processed the D_2O from the drums stored in the units. To date over 2700 drums have been processed.

One of the biggest challenges that Common Services staff faced during this process was that as the empty drums accumulated we did not have adequate storage space for them. At one point this actually stopped movement of drums from the units. Management approved the purchase of six large land/sea containers that allowed staff to store empty drums and continue with the process.

Common Services also sold over 400 empty drums to the Bruce facility in a deal that benefited both parties.

The units are now free of the vast quantity of TOC contaminated D_2O drums, and it has made a huge improvement in the housekeeping of the units.

This has now allowed the Common Services department to implement a drum-control process to ensure that the station will not return to the same issues of uncontrolled drum storage.

It should be noted that operating experience has shown that the glycol based solution used in the feeder inspection process has now been reduced to a significantly lower concentration and TOC values from dryer condensate average in the 20 ppm range.

3. Drum control process

The drum control process gives each operating unit fifteen colour-coded drums (each unit has a different colour). Full drums are brought to IXCU and are returned empty to the unit operators by

Common Services staff. Common Services have agreed to process one drum from each unit per shift up to a total of eight drums per shift in total.

Although it was initially hard to get buy-in from the unit operators who have been used to having a free hand in drum usage, it has now become accepted practice. Good support from station management was crucial in achieving this buy-in.

There are approximately 200 drums of D_2O left to be processed. When this is done the drum control process will be effective in cleaning the D_2O through the normal IXCU process and the large-scale process developed by Common Services staff will be discontinued.

4. Key message

It is imperative that staff feel they have the ability to make and effect change. Common Services has promoted innovative thinking from its staff and has allowed experimental processes to be developed that have had a huge benefit in other areas such as the introduction of washable radiation-protection equipment and other consumables, and the conversion of liquid chemical waste into solid form.

Each success breeds further innovative thinking and allows Common Services to provide more efficient and timely services to the Pickering plant, and valuable operating experience is gained from unsuccessful initiatives.

Moreover, staff are engaged and proud of what they do. Teamwork is evident in everything that Common Services does.

It is understood that not all areas of plant operation can be allowed the freedom to be creative and innovative as do the areas owned by Common Services, but managers at all plants need to identify areas where the power of their people's imagination can flourish. Managers need to welcome creative ideas and ensure innovation is never stifled.

Note: This process has been reviewed by Darlington and Bruce sites as well as a representative from the CANDU Owner's Group (COG). Common Services has offered to process a small amount of TOC contaminated D_2O from both nuclear sites to identify if the Pickering system will work for them.

FUEL HANDLING BENCHMARKING

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Abstract

On-power fuelling is unique to the CANDU type of reactor. The systems and equipment used to handle the fuel from the time it enters the station to the time it is transferred to the spent fuel bay are designed, operated and maintained exclusively for the CANDU stations. Over the last ten years it was perceived by several CANDU utility executives and outside organizations that CANDU fuel handling (FH) performance was degrading. FH organizations were seen as insular from the rest of the station and did not appear to be working to the same standards of excellence as the rest of the industry. The concerns raised were common to the industry. In 2005, COG was requested by one of its members to undertake an industry wide fuel handling Benchmarking (FHB) exercise of CANDU fuel handling organizations. The COG members decided to 'Take the cape off fuel handling' allowing all CANDU stations to see: actual performance of FH organizations; i.e. based on performance not perception, FH best practices, and identification of stations with best practices available for widespread use. All COG members joined COG project JP 4207. Taken together, the FH Benchmarking Final Report and the station Reports provide a good picture of current CANDU FH best practices and performance.

1. Introduction

On-power fuelling is unique to the CANDU type of reactor. The systems and equipment used to handle the fuel from the time it enters the station to the time it's transferred to the Spent Fuel Bay (SFB) are designed, operated and maintained exclusively for the CANDU stations. Over the last ten years it was perceived by several CANDU utility executives and outside organizations that CANDU fuel handling (FH) performance was degrading. FH organizations were seen as insular from the rest of the station and did not appear to be working to the same standards of excellence as the rest of the industry. The concerns raised by the stations were common to the industry.

In 2005, COG was requested by one of its members to undertake an industry wide fuel handling Benchmarking (FHB) project (JP 4207) of CANDU fuel handling organizations. Since operating experience with this equipment does not exist outside CANDU stations, exchange of experience and good practices in this area amongst the CANDU stations is particularly beneficial and, in fact, necessary.

The COG members decided to 'Take the cape off fuel handling', allowing all CANDU stations to see:

- Actual performance of FH organizations; i.e., based on performance not perception,
- FH best practices, and
- Identification of stations with best practices available for widespread use.

COG believed that it was necessary to use eyes in the field to disclose performance of FH organizations and provide a clear picture of excellence in FH organizations. The best way to establish a roadmap of FH 'best practices' and performance standards is to review the actual use of 'Industry Standards of Excellence' in FH organizations; i.e. WANO POCs (Performance Objectives & Criteria). It was intended to document their implementation and effectiveness in top performance FH organizations. We also wanted to know if there were any characteristics, peculiar to a top performing FH organization that were unique to FH as opposed to station operations.

2. Project description

Project objectives for the JP 4207 CANDU fuel handling benchmarking project were identified as follows.

1) Benchmark FH and FH Organization Performance of CANDU stations and document the baseline; i.e. snapshot.
2) Develop a CANDU FH database of broad benchmarking information as a resource for FH organizations.
3) Identify CANDU FH best practices (& station strengths).
4) Identify FH common issues (& station specific issues).
5) Build FH peer network & relationships among CANDU FH organizations across stations to augment collaboration and build a sense of community and eventually tackle 'fleet' problems.
6) Provide focused self assessment tools; i.e. performance standards, Q&A database, etc. to support management efforts to achieve station integration and use of industry standards of excellence.
7) Provide stakeholders with a path forward using an industry workshop to discuss project results and build consensus to support development of an industry improvement plan and transfer ownership to utility executives.

3. Project results

The FH Benchmarking project objectives were met as follows.

1) *Benchmark FH Performanc*e*:* COG conducted FHB visits to twelve stations with teams, each composed of 150-200 years of FH experience and 1-2 WANO experts. COG issued a FH Benchmarking Station Report (including recommendations) to each station providing a baseline snapshot of station FH organization performance. Taken together, the FHB Station Reports provide a good summary picture of current CANDU FH performance over a wide range of areas.
2) *CANDU FH Database & On-line FHB Forum of Q & As:* Software using blog technology has been developed to provide on-line access to the 600 FHB and Mini-Benchmarking questions and station answers. The database is being updated on a regular basis by COG as FH organizations continue to initiate FH mini-benchmarking through the COG Information Exchange base programme.
3) *CANDU FH best practices*: A list of 61 FH best practices including station contacts has been prepared to help FH organizations identify specific improvement opportunities. It is arranged by importance to the larger FH community. This list is by no means exhaustive.
4) *CANDU FH Common Issues:* A list of 18 common issues has been prepared to help member target areas for improvement. They identified by the prevalence of FHB visit observations. An issue had to be common to two or more stations to be considered common. Overall,
5) *FH Peer Network & Communications*: A noticeable step increase in the development of the FH peer relationships and inter-site communications was observed over the eighteen-month project duration. This was measured by a ramp-up in the volume of technical exchanges including: active mini-benchmarking forum separate from the FHB questions (typically 1-2 questions a week with 5-10 replies within 1-2 weeks), eight known technical-exchanges visits (subsequent to the FHB visits), dozens of exchanges of procedures, innovations, performance data, etc.
6) *Focused Self Assessment Tools:* Performance standards – The FHB project identified a need to provide all CANDU FH organizations with a clear roadmap to FH excellence as demonstrated by top performing organizations. Upon conclusion of the site visits, a COG fuel handling Index (FHI) was drafted to provide guidance in the key FH Equipment Reliability area. It is based on a Bruce Power FHI template as well as the WANO Equipment Reliability Index (ERI). Actual performance data from all stations, gathered during the FHB visits was used to determine key parameters and 'Best-in-Class' performance benchmarks.FHB Q&A

Database Directory – The FHB Q&A directory is an Excel database of all FH benchmarking questions and station answers and was also developed for use as a self-assessment tool for FH organizations. It provides a large inventory of assessment material common to all FH organizations.

7) *Path Forward:CANDU FH workshop:* The CANDU fuel handling Workshop (1st FHB Working Meeting #1) was held in June 2008 in Toronto (and the Pickering station), assembled over seventy representatives including forty-five station personnel from the entire international CANDU FH community including eight countries, to discuss FHB Project results and a path forward. This included FH department managers and supervisors, as well as FH equipment designers and vendors. Over two and a half days, attendees participated in eighteen presentations and four problem-solving Breakout sessions dedicated to tackle nine key FH issues identified by the FH Benchmarking project. This input was used to complete project results and recommendations.

General feedback received at the industry workshop in June 2008 confirmed the buy-in of all FH organizations. FH Stakeholders were very satisfied with the FH Benchmarking workshop and project results.

4. General observations

All FH Benchmarking Visit teams were well received at the host station and they benefited from the full cooperation of station management and FH organization. FH personnel at all stations demonstrated a keen desire to share and learn from other stations. They appreciated the scrutiny accorded to them by the visit team peers, and proactively exchanged ideas, issues, strengths and questions with the peers. It was also apparent that every station was very keen to leverage the FH Benchmarking project results to make further improvements.

Stations contributed a total of 112 strengths to the industry pool from which 61 CANDU FH best practices were selected and documented in this report.

It was observed directly from the station visits that there exists a wide range of performance between stations and between performance areas at a given station. It also became evident that the problems identified by any station (i.e. Top-Ten List) were common to most stations. Less than half of the problems were station/technology specific. It also emerged during the project that the vast majority of problems had already been solved at another station if not more. It was also observed that most of the 60 issues identified overall by the FHB visit teams were shared by three or more stations.

It was observed by FH Benchmarking teams that Top performing FH organizations consistently demonstrated:

1) Effective communication between FH operations, maintenance and engineering ensure good coordination of work and a common understanding of equipment and plant status.
2) High standards of housekeeping and material condition throughout FH areas
3) Motivated and engaged work force throughout FH maintenance. FH maintenance is involved in and participates with the station maintenance organization to stay current with station improvement initiatives and avoid ‘silo situations’.
4) FH Engineering support and prompt response is focused on any emerging issues via daily FH meetings.
5) FH Engineering displays a significant field presence and an integrated team approach. System walk downs by system responsible engineering staff are emphasized.
6) Fuelling machine availability was dependant on implementing a living preventive maintenance programme using worker feedback and equipment failure analysis.

7) High station schedule adherence in fuel handling operation and maintenance was the result of clearly defining and reinforcing management expectations for schedule development and implementation and it could only be achieved in a culture where integration among station operations and fuel handling is a top priority

8) Pre-job briefings are often comprehensive and provide good insights to expected job performance and error-likely situations.

5. Conclusions and recommendations

Stakeholder concerns with the sharing of FH best practices and lack of independent assessment, were observed by the FH Benchmarking project to be generally valid Numerous CANDU FH best practices have been developed by FH organizations that demonstrate industry standards of excellence and merit widespread use by the industry. FH Benchmarking results provide the tools and the momentum for CANDU fuel handling organizations to move forward.

Recommendations have been developed to provide an overall roadmap for the typical FH organization and their current performance baseline. Some of the recommendations could be implemented as a joint initiative or on a fleet basis.

6. Path forward

The Path Forward/Project Follow-up was handed over to the utility executive champions.

Each station was to consider their individual station report as the prime source of recommendations and the FHB Final Report as a complementary report for the FH community.

7. Lessons learned

Lessons Learned – FH Benchmarking:

- CANDU Peers + WANO Peer Assist Visit Process = Effective, repeatable results at low cost
- Industry peer teams with 150-200 a of FH & WANO experience on each visit team was key to credibility
- Performance based not perception based
- Deliverables designed as tools for ongoing use i.e. FHB Forum (Newsgroup) + living Q&A Data base
- FHB Project Peer Assist Visits build the relationships to open communications & work together on common issues

Appendix. CANDU FH best practices

1) Integration of FH maintenance into station's 13 week rolling schedule (Darlington)
2) Long term F/M maintenance planning & schedule adherence i.e. well defined schedule for fuelling & FH planned maintenance 2-3 years in advance (Cernavoda/PLGS)
3) Setting/reinforcement of management expectations for station schedule adherence in FH operations & maintenance driving integration with station operations (PLGS)
4) F/M head change-out procedure and tools allow 1 ½ day turnaround (Cernavoda)
5) F/M Preventive maintenance programme optimization programme gives near 100% f/m availability (PLGS)
6) Procedural adherence; i.e. implementation of FH continuous use procedure (Darlington/Pickering A)
7) Use of event free tools (PLGS/Cernavoda/Pickering A)
8) F/M spare parts kit (Darlington)
9) Independent oversight of fuelling activities (RAPS/Pickering A)
10) Spent fuel bay management/F/ME (Pickering A)
11) Housekeeping& low tritium in key FH areas i.e. programme in place to systematically reduce leaks and releases to minimize contamination and dose (PLGS)
12) F/M material condition & housekeeping with aggressive cleaning, lighting and management presence (Cernavoda)
13) Weekly FH operator training on OPEX/plant configuration/technical issues (4 hours from senior FH operator) (KHNP)
14) Housekeeping Standards and general material condition reflect employee ownership of plant (TQNPC)
15) Fuel handling Index (Bruce A/B)
16) FH equipment status log and electronic status monitoring (Darlington)
17) Cross-training of FH personnel in FH maintenance and operations gives ownership and flexibility (Cernavoda/PLGS/RAPS)
18) Comprehensive pre-job briefings and checklists (Bruce A/B)
19) Continuing training of FH operators i.e. 40 hours/year of classroom technical & OPEX (NASA)
20) FH work management overview tool to facilitate tracking of work order backlog and removal of material holds (Pickering B)
21) Comprehensive tracking of FH spare parts procurement by FH Engineering including use of FH materials SPOC performing one-year look aheads (Darlington)
22) Computer stops programme & software to capture, trend and analyze cancels to reduce their frequency (Pickering B)
23) Tritium dose reduction modifications for fuelling machine, F/M locks and spent fuel bay (PLGS)
24) Conservative decision making for increasing ram overhaul interval in a stepwise fashion following post-operation equipment evaluation (NASA)
25) Worker owned FH maintenance procedures feature ongoing feedback and pictures of key items, notes and precautions from workers (KHNP)
26) FH system health reports produced every 6 months for all FH systems (Darlington)
27) Colour CCTV cameras for enhanced remote monitoring of F/M operations and maintenance (KHNP)
28) FH Department 'equipment reliability reset' performance indicator (TQNPC)
29) CANDU F/M Troubleshooting guide (RAPS)

30) FH maintenance procedures with clear work instructions, prerequisites & precautions, clear markings for place-keeping & data recording and expert reviewed (Pickering A/B)
31) On-line reactor physics prediction programme i.e. finer control of reactivity/flux levels (G2)
32) Crisp shift turnovers allow complete review of plant status and key focus items. Electronic logs facilitate cascade from control room to shop (Pickering A/B)
33) CANDU FH knowledge mgmt.(RAPS)
34) FH system troubleshooting process/procedure with on-line information i.e. Ops & maintenance logs (Darlington)
35) FH model work orders/model pressure boundary work packages (Pickering B)
36) FH Department Performance Reports (Quarterly/Monthly/Weekly) (Darlington)
37) FH field operator rounds procedure & checklists (Bruce A/B)
38) FH system notebooks created and maintained for all FH systems (Darlington)
39) Full scope simulator under development (Darlington)
40) Plant /FH areas aesthetics & worker comfort; i.e. attention to workplace appearance & 'top quality' worker facilities & change rooms, increase productivity and ownership (KHNP)
41) Desk top simulator used to train and qualify personnel on FH normal, abnormal and emergency operations (RAPS)
42) Current on-line radiological conditions available on local monitors (KHNP)
43) 3D Animated F/M Models for Training & Troubleshooting (Pickering B)
44) FH system performance monitoring plan details all monitoring requirements (Darlington)
45) FH event reset trending & criteria (Pickering B)
46) FME exclusion boxes with required items available to facilitate FME practices by workers (Bruce A/B)
47) Electronic log keeping of control room and field operations including historical data and notes are available to operators and for trending (Pickering B)
48) DIGMA & FH Log for CANDU 6 equipment trending (Cernavoda)
49) FH abnormal operations procedures (PLGS)
50) Display case of electrically damaged components for teaching of C&I safe work practices (Pickering B)
51) Use of shipping containers in Zone 2 to store radioactive tools and equipment (PLGS)
52) Use of benchmarking to develop CANDU FH policy. This tool effectively to poll all stations with respect to Keeping Fuel in F/M overnight-Station Policy/Guidance (FHB Forum Q2.2.5.2-1) (All Stations)
53) Online data acquisition system (DAS) provides real time access to F/M Field data of operating equipment for equipment monitoring & troubleshooting (Pickering B)
54) F/M bridge shaft encoder calibration procedure (Bruce A/B)
55) Development of FH PSA tools to establish FH reliability centred maintenance baseline. (in progress) (TQNPC)
56) Participation of FH operators in control room simulator training during emergency operating procedure training (Pickering A)
57) Management of dry spent fuel storage processes & equipment (PLGS)
58) Redundant D2O pressure supply control system with dual controllers reduces transients and risk to equipment and PHT system (KHNP)
59) Emergency operating & alarm response procedures (NASA)
60) Review of chemistry to reduce PHT & F/M radioactive source term (Bruce A/B)
61) Maintaining F/M leak-tight connections i.e. oil leaks (Darlington)

DEVELOPMENT OF CONSOLIDATED SPENT FUEL DRY STORAGE SYSTEM

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Abstract

The four units at the Wolsong site are 700 MW(e) class CANDU 6 reactors. Wolsong Unit 1 went into service in 1983 and Wolsong Units 2, 3 and 4 began commercial operation in 1997, 1998 and 1999, respectively. About 20 000 bundles of spent fuel discharged annually into the spent fuel bays, where they are cooled by water. As the capacities of the existing spent fuel storage bays are limited, the spent fuel dry storage (SFDS) facility was built in 1990. Since 1991, spent fuel from Wolsong 1 has been stored at the SFDS, which uses concrete canisters each holding 540 standard CANDU 6 fuel bundles. The spent fuels from Wolsong 2/3/4 have been stored in the SFDS facility since 2005/2006. The SFDS facility has accumulated 162 000 bundles in 300 canisters, and sufficient capacity is only available to store the spent fuels from four units up until the end of 2009. KHNP has performed an in-depth evaluation to select a new dry storage technology. Key considerations were nuclear and radiation safety, technological maturity and economic/space aspects. In 2001, KHNP/NETEC selected AECL as a partner for the joint development of the MACSTOR/KN-400 storage module. The MACSTOR/KN-400 is an enlarged version of the MACSTOR-200 that uses its proven features while doubling its capacity. The MACSTOR/KN-400 is a secure concrete module housing steel cylinders that hold sealed baskets of spent fuel bundles. The capacity of the MACSTOR/KN-400 is 400 fuel baskets in a 4 by 10 array instead of a 2 by 10 array used in the MACSTOR-200. Seven modules are initially planned for construction at Wolsong. A storage density of approximately 88 bundles per square metre is expected from the new module, about three times the density of concrete canisters and 1.3 times the density of the MACSTOR-200 module.

1. Introduction

KHNP has operated the Wolsong 1 CANDU 6 NPP since 1983, and has operated the similar Wolsong 2-3-4 units since 1997, 1998 and 1999, respectively. Since 1991, spent fuel from Wolsong 1 has been stored at the Wolsong 1 dry storage facility, which uses concrete canisters each holding 540 standard CANDU 6 fuel bundles. The facility consists of 300 concrete canisters having a total dry storage capacity of 162 000 fuel bundles. The four reactors now produce approximately 20 000 bundles per year, and spent fuel from Wolsong 2/3/4 has also been stored the dry storage facility.

The basic storage density offered by concrete canisters is approximately thirty bundles per square metre. This density is sufficient at certain storage sites having a single reactor, but is now too low for a site like Wolsong that now has four units. For their supplementary fuel storage needs, two other CANDU 6 stations use a larger and denser structure, the MACSTOR-200 storage module. The MACSTOR 200 module has been used at Gentilly 2 in Canada since 1995, and at Cernavoda in Romania since 2003. These modules hold 12 000 fuel bundles in 200 fuel baskets, with each module holding sixty bundles. The fuel baskets are stacked ten high in each of twenty vertical storage cylinders, which are arranged in a two-by-ten rectangular array. The MACSTOR-200 provides a storage density of approximately sixty-eight bundles per square meter, which is about 2.3 times better than concrete canisters. In order to best use the land within the Wolsong site, KHNP decided to develop a storage structure with a storage density higher than that offered by the MACSTOR-200 module, and to construct it by 2009.

In 2001, KHNP/NETEC selected AECL as a partner for the joint development of the MACSTOR/KN-400 storage module. The MACSTOR/KN-400 is an enlarged version of the MACSTOR-200 that retains its proven features while doubling its capacity. The MACSTOR/KN-400 is thus configured to store 400 fuel baskets in a four-by-ten array instead of a two-by-ten array. A storage density of approximately eighty-eight bundles per square metre is expected from the new module, about three times the density of concrete canisters and 1.3 times the density of the MACSTOR-200 module.

2. Configuration of MACSTOR/KN-400 module

The MACSTOR/KN-400 module is a concrete monolith made from regular density reinforced concrete. The storage cylinders are vertical and made from galvanized carbon steel. The storage cylinder penetrates the module's top slab and is laterally restrained at its base by two seismic restraints, anchored in the module's floor. Each storage cylinder is closed at its top by a shield plug that is made from reinforced concrete lined with galvanized steel. A weather cover made of stainless steel covers the top of each storage cylinder. The modules are normally built on a base slab that enhances the interface between the module's structure and the foundations. With its thick reinforced concrete, the module is designed to withstand natural and man-made hazards such as a seismic event, high winds, tornado winds and tornado missiles and impact from equipment. Modules are laid in arrays within a fenced storage site. Figure 1 shows the configuration of the module.

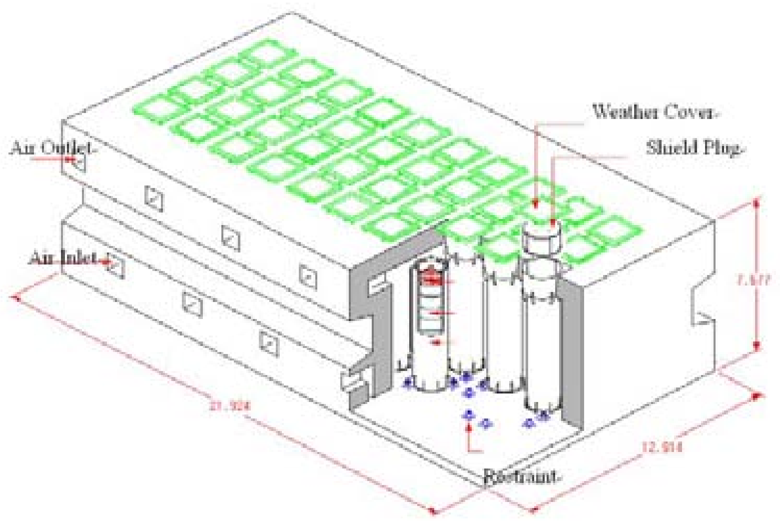

FIG. 1. Configuration of MACSTOR/KN-400 module.

The heat produced by the fuel bundles dissipates mainly by natural convection, infrared radiation and some conduction through the module's structure. The module is provided with an air circuit that is made of ten air inlets and of twelve air outlets laid as a labyrinth. The air circuit provides paths so that the cooling air, driven by its natural buoyancy, enters at the bottom air inlets and exits at the top air outlets. Fuel baskets are loaded into a storage cylinder using a transfer flask, a flask guide mechanism, a loading plug assembly and a gantry crane that are manually operated.

Each storage cylinder provides a confinement to the ten sealed fuel baskets stored into it, generating a safe, double barrier to the release of volatile radionuclides that could escape from non-leak tight fuel bundles. The module is designed to store both intact and non-leak tight (but mechanically sound) fuel bundles. Each storage cylinder is provided with a vent and drain line capped with valves located outside the lateral shielding walls. These lines provide means to periodically verify the integrity of the two confinement barriers using the storage cylinder monitoring system.

3. Design requirement of MACSTOR/KN-400 module

Like many other engineered dry storage systems, the safe operation and maintenance of MACSTOR/KN-400 basically depends on adopting adequate design requirements. The most important design targets for the module are those that provide the necessary assurance that spent fuel can be received, handled, stored and retrieved without undue risk to the health and safety of workers or the public.

To achieve these objectives, the design of the module incorporates features to remove spent fuel residual heat, to provide for radiation protection, and to maintain containment over the lifespan of the module as specified in the design specifications. The features also provide for all possible anticipated operational occurrences and design basis events in accordance with the design basis as guided by the designated regulations.

The general performance requirements of the MACSTOR/KN-400 are listed in Table 1. The parameters in the requirements have been primarily derived from the previous performance requirements of the MACSTOR-200 module. (Refer to Table 1)

The MACSTOR/KN-400 storage module is designed to store fuel bundles having the parameters listed in Table 2. Small increases/decreases in average burn-up (from small variations in the initial fuel bundle Uranium mass) with respect to the specified reference average may occur as the mass of bundles may vary from plant to plant. These variations are compensated for by a corresponding small site-specific increases/decreases in the reference-cooling period of six years,[1] as long as the average reference fuel bundle heat release is met. (Refer to Table 2).

The generic design basis events considered for the MACSTOR/KN-400 module are listed in Table 3. Shielding and radiological requirements of the module are listed in Table 4. (Refer to Tables 3, 4)

Table 1. General Performance and Capacity Requirements

PARAMETER	PERFORMANCE REQUIREMENT
• Reference dry storage period	50 years
• Reference fuel	CANDU 6 spent fuel bundles
• Bundle integrity	Intact and non-leak tight (mechanically sound)
• Location of fuel storage site	Co-located within the Exclusion Zone Boundary of Wolsong CANDU 6 station
• Quantity of fuel bundles per module • Quantity of fuel baskets per module • Quantity of storage cylinders per module • Quantity of fuel baskets per storage cylinder • Number of fuel bundles per basket	24 000 bundles per module 400 fuel baskets per module 40 storage cylinders per module 10 fuel baskets per storage cylinder 60 fuel bundles per fuel basket
• Air cooling circuit	10 air inlets (5 on each side) 12 air outlets (6 on each side)

[1] i.e. the minimum time that bundles must cool in the spent fuel bay before transfer to dry storage.

Table 2. Acceptance Criteria for Fuel in Storage Module

PARAMETER	PERFORMANCE REQUIREMENT
Reference fuel cooling period	6 years for reference fuel
Fuel age spread in module	None (All bundles are conservatively assumed to have minimum cooling period)
Reference average fuel burn up	7 800 MW/d/Mt U
Average bundle heat release	6.08 Watts
Reference maximum fuel burn up	12 083 MW/d/Mt U
Maximum bundle heat release	9.76 Watts
Irradiation period/bundle power	325.5 days/452.5 MW(th) per bundle
Fuel bundle initial Uranium contents	18.9 kgU – generic, 19.2 kg (Wolsong specific)
Reference hot basket configuration	53 average power basket 7 maximum power bundles in a cluster
Fuel basket heat release: Average basket Hot basket:	 364.8 Watts 390.6 Watts
Maximum initial fuel bundle temperature	160°C(168°C for allowable)

Table 3. Generic Design Basis Events Considered for MACSTOR/KN-400 Module

DESIGN BASIS EVENT	CRITERIA
Design Basis Earthquake (ground motion acceleration)	0.2g horizontal acceleration 0.133 g vertical acceleration
Wind caused by Typhoon or Hurricanes	144 km/h (Meteorological record in Korea)
Severe air flow blockage conditions	50% of air inlet circuit (at non-floodable site) 100% of air inlet circuit (at floodable site)
Fuel basket drop in storage cylinder	From transfer flask to bottom of storage cylinder
Drop of storage cylinder shield plug	From highest handling height
Drop of flask guide mechanism	From highest handling height
Transfer flask drop on module • If commercial transfer flask hoist (having a regular reliability) is used • If single-failure-proof transfer flask hoist is used	 Drop from maximum operational height Transfer flask drop is not an applicable Design Basis Event
Collision from land vehicle	Collision from transfer flask transporter at speed of 20 km/hr
Fires	Fire from transfer flask transporter fuel tank

Table 4. Shielding and Radiological Requirements

ITEM	CRITERIA
Contact dose rate on module	25 μSv/h
Temporary dose rate during fuel basket loading	250 μSv/h
Fence dose rate	2.5 μSv/h
Effective dose (occupational)	Less than 20 mSv per year
Effective dose (for public at Exclusion Zone Boundary) from normal operation	Less than 0.1 mSv yearly
Effective dose (for public at Exclusion Zone Boundary) following Design Basis Events	1 mSv

4. Design description of MACSTOR/KN-400

Storage cylinder

The storage cylinder is designed to contain ten baskets and is made of carbon steel. Zinc corrosion protection is applied to all internal and external storage-cylinder surfaces. Table 5 shows the design features of the storage cylinder.

The bottom of each storage cylinder is separated from the concrete base by 18 cm. This arrangement enables a uniform cooling of the cylinders by the airflow circulating underneath the storage cylinders, and absorbs thermal expansion due to temperature increases during long-term storage. The cylinder is designed to sustain impact loads due to basket drops during handling. Each cylinder constitutes an independent confinement structure against radioactive material escape, and thus forms a double-confinement boundary when considered together with the basket. Each storage cylinder is equipped with a vent and drainpipe terminating outside the shielding in a valve box. The monitoring system is composed of a small-capacity air pump, a particulate filter, a desiccant vessel, and the necessary fittings and tubes. Monitoring consists of connecting the air pump and filters to the storage cylinder cavity vent and drain valves, and recirculation the gases through the system. (Refer to Table 5)

Table 5. Design Parameters and Materials of Storage Cylinder

Storage Cylinder	Parameter
Number of storage cylinders	40
Size of storage cylinder Overall height External diameter of main body	 6.93 m (rounded) 1.14 m (rounded)
Storage cylinder power:	Initial: 3.6 kW At 50 years: 1.2 kW
Typical dose rate seen by the storage cylinder body	2000 to 3000 Rads per hour initially
Main storage cylinder materials: • Body • Top portion for seal welds with shield plug and weather cover	 • Zinc coated carbon steel • Stainless steel grade 304L, annealed

Thermal insulation panel

The MACSTOR/KN-400 is capable of dissipating the heat released from the spent fuel using the same air circuit as the MACSTOR 200, as it is equipped with thermal insulation panels (TIP) inside the module to protect the concrete. The thermal insulation panels consist of insulation material enclosed in a housing that is made of stainless steel sheet metal. The thermal insulation material performs only a thermal insulating function. The stainless steel box protects the insulation from damage, provides the necessary structural stability, and prevents the ingress of water. TIPs are used at two locations: at the ceiling of the module and at the superior portion of the walls. The ceiling's TIPs are embedded into the ceiling while the wall-mounted TIPs are anchored to the internal walls of the module. The anchor bolts of the wall mounted TIPs run through a stainless steel anchor-support pipe connecting the front and the back of the TIP. The glass type material used for the TIPs is chemically inert, impermeable to water and provides excellent resistance to high doses of gamma radiation. The material maintains its thermal insulation and mechanical properties under a wide range of temperatures, well above the MACSTOR/KN-400 operating temperature.

Re-verification tube

The MACSTOR KN-400 module is based on the MACSTOR- 200 design but has twice the capacity and thus twice the number of storage cylinders. In all, the new module contains 40 dry fuel storage cylinders, each of which houses ten spent fuel baskets. The storage cylinders are arranged in four rows of ten, with twenty-four located close to the periphery of the module and sixteen located internally at some distance from the peripheral walls.

Re-verification is an IAEA safeguard requirement, and involves measuring the gamma dose rate and spectrum of each irradiated fuel basket once the storage cylinders are loaded with spent fuel. This is required for confirming the presence of spent fuel in the storage cylinders. To achieve this on the existing MACSTOR-200, a re-verification tube running inside the module walls is provided for each storage cylinder. The gamma profile is read by lowering a detector inside the tube so that it can be registered at the level of each basket. For the twenty-four peripheral storage cylinders this method of measurement is retained on the MACSTOR KN-400 module. An alternate method is required, however, for the sixteen internal storage cylinders since they are located some distance from the module walls and are thus surrounded by storage cylinders. In order to be effective in this prescribed monitoring function, the re-verification column is designed so that the signal received at each of these forty detector positions originates primarily from only one fuel basket.

The detector signal is derived from the radiation source of the target fuel basket and background radiation. The background radiation will originate primarily from the numerous other fuel baskets contained in the storage cylinder. One way of maximizing the signal from an individual basket while simultaneously reducing the background signal is to have an unshielded path oriented toward an individual fuel basket through a thick shield. The unshielded paths in the re-verification columns are the collimators (or view tubes). Figure 2 shows the arrangement of collimators within the re-verification column. Over the 55.7 cm height of a fuel basket, there are four collimators vertically separated and aimed at four different fuel baskets in the four nearest storage cylinders. In this way, a single re-verification column can monitor forty nearby fuel baskets. The re-verification column is square in shape. It contains a central steel box six-inches square with a two-inch square central cavity. This steel box is enclosed within concrete thirty-inch square. The collimators penetrate through the steel and concrete structure of the re-verification column. This steel and concrete structure attenuates the background signal to increase the signal from the target basket.

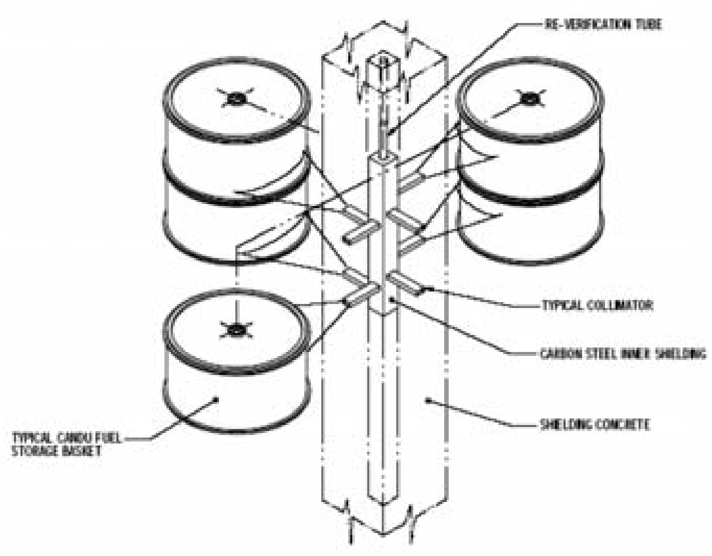

FIG. 2. Arrangement of collimators within MACSTOR/KN-400 Re-verification columns.

Gantry crane

The MACSTOR/KN-400 gantry crane shall be constructed and tested in accordance with the 'Single Failure Proof (SFP)' requirements to NUREG 0554. That is to say, the crane hoists shall be designed to maintain the loads in position or lower them in a controlled manner during a loss-of-power event. The crane is equipped with two (2) electric hoists, a main hoist and an auxiliary hoist, mounted on the same trolley. The auxiliary hoist will be used to remove and replace the shield and loading plugs on the concrete storage module. The main hoist will be used to lift the fuel transfer flask from the flask transporter and place it atop the concrete storage module. The hoist will also lower the flask back onto the flask transporter.

The gantry crane is used to lift the fuel basket transfer flask from the flask transporter, remove the shielding plug from the concrete storage module, place the flask on the top of storage module so that the fuel basket can be lowered into the storage module, put the shield plug back on the storage module, and return the flask to the transporter. The gantry crane is also used to place the fuel basket transfer flask guide mechanism at the top of storage module before the storage module is first loaded with baskets.

5. Conclusions

In summary, it can be concluded that the MACSTOR/KN 400 module satisfies the following requirements for safe storage of CANDU spent fuels.

- It can store CANDU 6 fuel baskets containing reference fuel bundles, passively dissipate heat generated by the stored fuel and maintain fuel bundles and storage module at acceptable temperatures.

- It can provide sufficient shielding to attenuate gamma and neutron radiation, keeping them below acceptable values.

- It provides confinement for the storage basket.

- It provides adequate structural integrity during construction, normal and abnormal operation, and during design basis events.

- It provides capability for periodic sampling of each storage cylinder cavity.
- It provides a basic intrusion resistance against removal of fissile material and receptacles for the installation of Safeguards monitoring equipment by the IAEA.

ROD-BASED GUARANTEED SHUTDOWN STATES (RBGSS) IMPLEMENTATION AT PICKERING B

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Abstract

During Pickering B unit outages, the over-poisoned guaranteed shutdown state (OPGSS) is used to keep the reactor deeply sub-critical. A neutron-absorbing material, namely gadolinium nitrate, is dissolved in the moderator. The over-poisoned moderator system is then isolated from D_20 transfer lines and other systems that can potentially alter the poison concentration of the moderator inventory, such as the moderator-purification and the poison addition systems. The overall OPGSS process is complex, time and labour intensive, has disadvantageous with respect to conventional, and radiation safety, and raises station-material-conditions concerns. Due to the large number of devices that must be placed in guaranteed positions, and also due to difficulties in accessing some of the required devices (because of their physical locations inside the reactor building), the application and removal of OPGSS takes at least twenty-four hours, thus causing delays in establishing a defined reactor state. As well, the approach to critical (ATC) procedure following an OPGSS is also a long process due to the required high concentration of Gadolinium dissolved in the moderator. This approach to reactivity hold down is unique to the CANDU design. Around the world, other nuclear-reactor designs utilize the installed control and shut-off rods to establish a guaranteed shutdown state. Inadvertent removal of the rods is prevented by physical barriers and by enforcing strict procedural controls. On completion of the guaranteed shutdown state, criticality is normally achieved by withdrawing the rods from the reactor core. Similarly, it has been demonstrated that for the Pickering B reactor design, sufficient sub-criticality margin can be achieved and maintained by guaranteeing all solid shutoff absorbers (SAs), control absorbers (CAs), and adjuster absorbers (AAs) are in core, and by applying appropriate physical barriers and procedural controls to prevent their inadvertent removal from the core. This new configuration is referred to as a Rod-Based GSS (RBGSS). Under RBGSS, with all rods guaranteed in the core, sufficient negative reactivity is still provided for any credible process failure; Shutdown system 1 (SDS1) can, therefore, be considered 'available.' In fact, an argument can be made that the 'in core' state of SAs is better than the nominal poised state, in that all possible failure modes of SDS1 have been removed by safe-stating of the SAs. In other words, the RBGSS is simply the 'pre-activation' of SDS1 ahead of any accident, and placing it in a state in which it has already been demonstrated that it is able to keep the reactor sub-critical. The 'pre-activation' eliminates the (very small) unreliability associated with the trip logic, clutches and SA rods. Furthermore, throughout the RBGSS, shutdown system 2 (SDS2) shall be maintained poised and available — providing additional protection, and fulfilling the defence in depth requirement.

1. Technical considerations

While no formal definition of the guaranteed shutdown state (GSS) is found in regulatory documents, a typical definition is found in the Pickering B Operating Policies and Principles (OP&Ps). Here, the GSS is defined as a state where the reactor remains sub-critical in the event of any process failure, and administrative safeguards are in place to prevent net removal of reactivity. This general GSS definition is detailed in terms of equipment configuration and procedures in station Operating Manuals.

1.1. Safety analysis

A detailed safety analysis has been completed by Nuclear Safety Solutions Limited (NSS) for Ontario Power Generation. The analysis accounts for all uncertainties and credible accident scenarios and has considered both the primary heat transport system (PHTS) hot and pressurized as well as cold and depressurized cases. In summary, the results of the analysis indicate that:

a) the most limiting accident while in RBGSS corresponds to a Loss of Coolant Accident (LOCA) with the loss of heat sink under cold moderator and heat transport system conditions;

b) the heat transport system may remain hot and pressurized;

c) there is sufficient sub-criticality margin in the RBGSS configuration to accommodate normal variations in operating parameters including moderator purity, shutdown fuelling, drained liquid zone compartments, and the removal of a single shutoff rod; and

d) for the transition to approach to critical, sufficient moderator poison must be added to ensure the reactor will remain sub-critical following re-poising of all SAs and CAs.

Note: the design of Pickering B reactor (very similar with the other CANDU plants) provides two independent, safety-grade shutdown systems: SDS1 with 28 shut-off rods (stainless steel or cobalt) that are dropped in the core when activated, and SDS2 that injects Gadolinium into the core (moderator) when activated.

1.2. Common mode failure analysis

The adequacy of the RBGSS scheme to cope with design-basis earthquake (DBE), fire, and harsh-environment (EQ) events has been assessed. It was concluded that RBGSS is DBE, fire and EQ qualified. The arguments for this conclusion are as follows:

The SDSI shut-off rod mechanism was designed to DBE category B in accordance with the guidelines for seismic qualification of safety related systems. This means the shut-off rods remain functional (i.e. capable of being inserted into the core) during and/or following a design basis earthquake, and remain in-core if inserted prior to a DBE.

In the case of a design-basis fire, the method of rod-based GSS deployment provides assurance that the GSS will not be affected. In particular, the power supply to the shut-off rod mechanisms is guaranteed to go open-circuit via devices in both the main control room (MCR) and at the motor control centres (MCCs). Therefore, it is not possible for any postulated fire to cause a hot-short condition at both the MCR and MCCs that could energize the shut-off rod circuits and facilitate spurious shut-off rod operation.

Furthermore, EQ harsh environment will not affect the reactor auxiliary bay (RAB) and MCR at the same time, so as to cause spurious energization of the shut-off rod mechanisms.

2. Regulatory perspective

The regulatory authority in Canada is the Canadian Nuclear Safety Commission (CNSC).

The only issued regulatory document to specifies GSS requirements is R-8, '*Requirements for Shutdown systems for CANDU Nuclear Power Plants*', which documents current regulatory expectations and thereby provides a point of reference for all reactor vintages.

Consultative document C-6 R1, '*Safety Analysis of CANDU Nuclear Power Plants*', only requires that the GSS be considered as a plant state for the systematic plant review, but otherwise imposes no requirements on GSS *per se*.

One other regulatory document, P-242, '*Considering Cost-Benefit Information*', is relevant to the regulatory considerations of a RBGSS. In summary, this policy allows CNSC staff to consider costs and benefits when making decisions involving a licence.

There are no regulatory requirements or expectations that would appear to be violated by adoption of a RBGSS. The RBGSS compliance with the regulatory requirements and expectations listed in R-8 are summarized below:

A. Procedures for putting the reactor in a guaranteed shutdown state shall be prepared and shall require approval by the (CNSC) prior to issuance of an operating licence. Compliance with this requirement is ensured by the issuance of an operating documentation (Operating Manual) for RGGSS application and removal. Revisions to this document require CNSC approval.

B. GSS procedures shall specify at least two independent means of ensuring the reactor remains subcritical. Compliance with this requirement is ensured by:

- Manual operation of the breakers/disconnects on the MCCs supplying power to the rod motors being prevented by the application of a lock.

- Disconnects on the SA and CA clutch power supplies being opened and tagged

- All rod control and rod-bank selector hand switches on the control panel in the MCR will be selected to ‘STOP’ and tagged.

C. A shutdown system shall not intentionally be made unavailable — except when the reactor is in an approved GSS. Compliance with this requirement is ensured by:

- SDS1 being pre-activated and locked in the core.

- SDS2 being poised and available.

D. When the reactor is in an approved GSS, not less than one shutdown system shall be available at all times when this is practical. Compliance with this requirement is ensured by having the SDS2 poised and available.

E. In the event one shutdown system operates, it shall be returned to the poised state as soon as practical without causing criticality, or the reactor shall be placed in an approved GSS. Compliance with this requirement is ensured by the operating provisions in the Operating Manual: ‘following a non-spurious SDS1 or SDS2 neutronic trip while in RBGSS, the unit shall be placed in over-poisoned GSS (OPGSS).’

3. Implementation

3.1. RBGSS application

The following steps are taken in order to apply RBGSS:

a) Enter RBGSS by placing the shut-off absorbers (SAs) into the core, either by a trip (manual or automatic) or by orderly driving them in.

b) Drive all of the control absorbers (CAs) and adjuster absorbers (AA) manually into the core, if they are out of core.

c) Apply RBGSS isolations as follows.

   - tag in the ‘STOP’ position all the MCR hand switches associated with the control of the SAs, CAs and AAs;
   - open, lock, and tag all of the motor power supplies for the SAs, CAs and AAs; and
   - open and tag power supplies to SAs and CAs clutches.

d) Place the moderator purification in by-pass mode.

e) Shift manager to verify the tagged devices.

f) Operations and maintenance director accepts the RBGSS.

3.2. Maintain RBGSS

During RBGSS the following conditions must be met.

a) All SAs, CAs, and AAs remain fully in core.

b) SDS2 is poised and available.

c) All guaranteed devices are verified and logged daily.

d) Heat transport and moderator isotopics remain within the normal operating range.

e) Maintenance activities that increase the risk of withdrawing the SAs, AAs or CAs from core or that make SDS2 unavailable are restricted.

Following a non-spurious SDS1 or SDS2 neutronic trip during RBGSS, the unit must be placed in OPGSS.

3.3. Exit from RBGSS

The RBGSS may be terminated when the approach to critical (ATC) process is about to start on the unit, or when the unit has been placed in OPGSS.

In order to exit RBGSS and start ATC, the steps to be taken are:

- determine the initial Gd poison concentration requirements for ATC,
- confirm moderator purification is on by-pass,
- add the required concentration of poison,
- surrender RBGSS,
- remove locks and tags from guaranteed devices,
- re-poise SAs and CAs,
- place moderator purification in service, and
- proceed with ATC by poison removal.

In order to exit RBGSS and enter OPGSS, the steps to be taken are:

- remove the moderator spool piece to isolate the moderator purification,
- add poison,
- establish OPGSS, and
- surrender RBGSS.

4. Conclusions

There are numerous benefits associated with the RBGSS when comparing to the OPGSS:

Reactor safety benefits:

- Fewer devices that must be in guaranteed positions (22 devices compared to 44);
- Transition times to and from GSS are shorter — 18 hours and 30 hours, respectively;
- Independence from moderator chemistry, and no concern of precipitation/dilution;

- Approach to critical performed under reactor regulating system control (Ion chamber signal remains rational throughout the RBGSS);

- Lower risk of D_2 excursions in the moderator cover gas.

Radiation safety benefits:

- Less dose used in the application of RBGSS compared to OPGSS (approx. 300 person-mrem; 3 p-mSv).

Economic benefits:

- Shorten outage duration by approximately two days.

The following steps were taken in the implementation of RBGSS:

- Technical and reactor safety analysis were performed to demonstrate the viability of the concept.

- Discussions with the CNSC regularly performed, in preparation for the final approval by the regulator.

- Operational decision making meetings held at Pickering B with participation from other stations to assess the readiness for the physical implementation of RBGSS.

- Operations instructions issued in order to document the new conditions

- Training programmes revised in order to include the new process.

- A trial of the concept performed in 2007, during P761 outage

- A demonstration of the concept to be performed during a planned/forced outage

- Final approval to permanently use RBGSS to be obtained from CNSC.

Pickering B is convinced of the advantages of RBGSS and it is committed to pioneering the new concept.

ABBREVIATIONS

AA	adjuster absorber
AECL	Atomic Energy of Canada, Limited
AERB	Atomic Energy Regulatory Board (of India)
ALARA	as low as reasonably achievable (economic and social factors considered)
ATC	approach to critical
BDBA	beyond design basis accident
CA	control absorber
CANDU®	Canada deuterium uranium[24]
CNCAN	National Commission for Nuclear Activities Control of Romania
CNEA	Comisión Nacional De Energía Atómica, República Argentina
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners' Group
CRE	collective radiation exposure
CRUD	Chalk River unidentified deposit
CSEN	State Committee for Nuclear Energy (of Romania)
DAC	derived air concentration (typically of tritium)
DAE	Department of Atomic Energy (of India)
DBA	design basis accident
DBE	design basis earthquake
DCC	digital control computers
DOM	director of operations and maintenance
ECCS	emergency core cooling system
EPRI	Electric Power Research Institute
ESL	environmental survey laboratory
F/M	fuelling machine
FH	fuel handling
FLR	forced loss factors
GSS	guaranteed shutdown state
HCLPF	high confidence low probability of failure
HPU	health physics units
HTS	heat transport system
HVAC	heating, ventilation, and air conditioning
HWR	heavy water reactor
I&C	instrumentation and control
ICET	Integrated Chemical Effects Test
ISI	in service inspection
ISOE	information system on occupational exposure
KAERI	Korea Atomic Energy Research Institute
KEPRI	Korean Electric Power Research Institute
KHNP	Korea Hydro & Nuclear Power Co. Ltd
LOCA	loss of cooling accident
LOECC	loss of emergency core cooling
LRF	large release frequency
MCR	main control room
MOV	motor operated valve
MPC	maximum permissible concentration (in air; typically of tritium)
NGS	nuclear generating station
NI	nuclear island
NPCIL	Nuclear Power Corporation of India Limited

[24] CANDU is a registered trademark of Atomic Energy of Canada, Limited (AECL).

OCTF	one cycle trouble free
OP&P	operating policies and principles
OPG	Ontario Power Generation
OPGSS	over-poisoned guaranteed shutdown state
PHT	primary heat transport (system)
PLGS	Point Lepreau Generating Station
PM	preventative maintenance
PSA	probabilistic safety analysis/assessment
QA	quality assurance
RATM	real time tritium monitor
RBGSS	rod-based guaranteed shutdown state
RE&E	risk estimation and evaluation
RIDM	risk-informed decision making
RSL	risk significance level
RTS	risk tolerability scale
SA	shutoff absorber
SCA	safety and compliance assurance procedure
SCDF	severe core damage frequency
scfm	standard cubic feed per minute
SDS1	(safety) shut-down system 1
SDS2	(safety shut-down system 2
SFCR	single fuel channel replacement
SFDS	spent fuel dry storage
SMA	seismic margin assessment
SPOC	single point of contact
SPV	single point of vulnerability
SSC	systems, structures and components
TOC	total organic carbon
TRF	tritium removal facility
UCLF	unplanned capacity loss factors
VRD	vapour recovery dryer
WANO	World Association of Nuclear Operators

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