**ALARA ENGINEERING
AT
DEPARTMENT OF ENERGY FACILITIES:
BIBLIOGRAPHY OF SELECTED READINGS
IN
RADIATION PROTECTION AND ALARA**

S.W. Daniel, E. Kaplan, B.J. Dionne, T.A. Khan, S.G. Lane and J.W. Baum

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PREFACE

This report is the first in the series of bibliographies supporting the efforts at the Brookhaven National Laboratory ALARA Center on dose reduction at DOE facilities. Abstracts for this bibliography were selected from proceedings of technical meetings, journals, research reports, and searches of the DOE Energy Data Base. The abstracts included in this report relate to operational health physics as well as other subjects which have a bearing on dose reduction. Facilities covered include: radioactive waste, uranium enrichment, fabrication, unirradiated fissile material storage, irradiated fissile material storage, reprocessing, decommissioning, recovery, hot laboratories, tritium production, reactors (research, test and production but not power reactors), and accelerators. We have also included material on improved design, materials selection, planning, and other topics which are related to dose-reduction efforts.

The report contains 68 abstracts as well as subject and author indices. These abstracts were obtained from searches conducted from May 2, 1988 to May 2, 1989.

Information that the reader feels should be included in the next volume of this bibliography may be submitted to:

BNL ALARA Center
Building 703M
Department of Nuclear Energy
Brookhaven National Laboratory
Upton, N.Y. 11973 U.S.A.

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ACKNOWLEDGMENTS

The production of this volume of the bibliography was simplified by the use of personal computers and electronic links, which enabled one to rapidly search remote data bases for appropriate projects and to carry out the time-consuming but necessary tasks of producing subject and author indices. We thank our project sponsors, Kenneth Ferlic and Anthony Weadock of the U.S. Department of Energy, for their support of this work. We would also like to acknowledge the advice and encouragement of Charles Meinhold of Brookhaven National Laboratory. Finally, we would like to thank our secretary, Maria Beckman, for her help in producing this report.

COMPILATION OF ABSTRACTS

1. NON-DESTRUCTIVE IDENTIFICATION OF UNIDENTIFIED RADIOACTIVE MATERIALS USING MULTIPLE DATA INPUTS. G.S. MIHALOVICH. (Westinghouse Materials Co. of Ohio, Cincinnati, OH (USA)). Report No. FMPC--2158, 1989, pp. 10.

The Feed Materials Production Center (FMPC), is a Department of Energy (DOE) facility, that produces high purity uranium for defense programs. It also serves as the DOE's repository for temporary storage of thorium materials. The containers storing the thorium materials have suffered from extensive environmental deterioration and with the passing of time records have been lost. The net result is that a significant number of containers have lost their identifying markings. The exact type of materials in some of the containers is unknown. The containers must be repackaged for Environmental Safety and Health (ES&H) reasons. Prior to repackaging the identity of the contents must be ascertained. However, these containers cannot be opened due to As Low As Reasonably Achievable (ALARA) and ES&H considerations. A method was developed to determine the content of the containers, without physical inspection, prior to the repackaging/overpacking. This method utilizes a computer based data base management system which incorporates statistical analysis of the available input data and assigns statistical weighting to the various inputs, based on the significance of the data. The data consists of the following: historical data on the containers, if it exists; the weight of the containers and materials; the gamma energies related to the various materials; the geometry of the container and materials; the absorption coefficients of the materials and containers; the self-shielding of the materials; external contamination data, if any; and any other data obtained either prior to or during the overpacking effort. This system will work not only for the thorium materials, but the system has the capability to perform similar functions at other facilities faced with similar concerns.

2. ON GIVING RADWASTE MANAGEMENT SOME STATUS. H.W. DICKSON, E.E. WALKER, and J.W. THIESING (Bechtel National, Inc., Oak Ridge, TN (US)). Proceedings of a Symposium - Topical conference on theory and practice in radiation protection, Knoxville, TN, US, April 22, 1987.

Radwaste management is receiving ever increasing attention in the nuclear industry. The reasons for this include limited allocations for burial, increasing costs of handling and disposal, increased regulatory attention, and ALARA requirements. These issues have lead to an increasing awareness of the disadvantages of running a 'dirty' plant and a variety of sophisticated systems have been proposed to fix the problem. Instead of these technologically difficult, and sometimes very expensive fixes, this paper focuses on several relatively simple 'low tech,' and inexpensive solutions. Much can be done with organizational alternatives and assigned responsibilities and authorities to improve the situation. Applying controls on the front end of a radiological task rather than attempting to reduce the magnitude at the back end is the only realistic method for proper radwaste management.

3. STATUS OF ACHIEVEMENTS REACHED IN APPLYING OPTIMIZATION OF PROTECTION IN DESIGN AND NORMAL OPERATION OF NUCLEAR FACILITIES. J. LOCHARD and J.R. CROFT (Centre d'Etude sur l'Evaluation de la Protection dans le domaine nucleaire, 92 - Fontenay-aux-Roses (France); National Radiological Protection Board, Leeds (UK)). Optimization of radiation protection, Paris, FR, March 14, 1988. Nuclear Energy Agency, 75 - Paris (France). Report No. INIS-XN--149, pp. 113-142, 1988.

The increased emphasis in recent years on the ALARA principle and its practical implementation has opened new perspectives in the organization of radiological protection. This change could be characterized as a shift from an 'a posteriori' control, (i.e. demonstrating that dose limits had not been exceeded), towards an 'a priori' management or dose predictive approach. Undoubtedly in both operation and

design conscious efforts are made to achieve ALARA. However, there are certainly differences in various organizations' perceptions of what this practically entails and there is scope for us all to learn. If the 'ALARA Procedure' remains only a tool used at the early stage of design or as an elegant means of justifying past choices, it will have missed its main objective which is to give more rational and coherent management of practical protection at both the design and operational stages. We feel that the integration of the structured approaches of the ALARA Procedure and the ALARA Audit, into radiation protection programmes provides the best way forward. This would require ALARA studies moving from the province of the 'experts' to the 'practitioners'. This in turn will require us to provide them with the tools to do the job. The methodology is secure but the areas of dose data collection and the costing of detriment need to be addressed further. Similarly, there is perhaps a need for the 'experts' to develop further experience of using the multi-attribute and multicriteria techniques for the higher level decisions.

4. INTEGRATION OF OPTIMIZATION IN THE REGULATORY PROCESS. R.E. CUNNINGHAM, D.A. COOL, and R.L. O'CONNELL (US Nuclear Regulatory Commission, Washington, DC (USA)). Optimization of radiation protection, Paris, FR, March 14, 1988. Nuclear Energy Agency, 75 - Paris (France). Report No. INIS-XN--149, 1988, pp. 87-112.

The purpose of this paper is to describe how optimization fits in a complex regulatory regime aimed at protection of the public health and safety. The comments and observations about optimization in the regulatory process are based mainly on the authors' experience in a single agency, the US Nuclear Regulatory Commission (NRC). The NRC regulates a broad spectrum of nuclear activities ranging from nuclear power reactors and the nuclear fuel cycle to the use of radioisotopes in industry, research and medicine.

5. OPTIMIZATION AND DECISIONS IN RADIOLOGICAL PROTECTION - A REPORT OF THE WORK OF AN ICRP TASK GROUP. G.A.M. WEBB (National Radiological Protection Board Chilton (UK)). Optimization of radiation protection, Paris, FR, March 14, 1988. Nuclear Energy Agency, 75 - Paris (France). Report No. INIS-XN--149, pp. 37-80, 1988.

In 1984 the International Commission on Radiological Protection (ICRP) established a Task Group of Committee 4 to produce a report on methods for optimization of protection other than cost-benefit analysis. As the work of the task group progressed it became clear that it would be more useful to produce a report on the entire field of application of optimization, mainly to show how the various techniques including cost-benefit analysis could be applied appropriately to problems at different levels of complexity. This paper reports on the main ideas that have been developed by the task group. It must be emphasized that these ideas have not been endorsed by Committee 4 nor approved by the Commission so they can not yet be considered as recommendations.

6. COST BENEFIT ANALYSIS OF RADIATION SHIELDING IN A 60 MEV ALPHA ACCELERATOR. P.K. SARKAR (Variable Energy Cyclotron Centre, Calcutta (India). Health Physics Unit). Indian Journal of Physics, Part A (India), Vol. 62, January 29, 1987, pp. 618-622.

Cost benefit analysis of radiation shielding for a medium energy accelerator with 60 MeV alpha ions is carried out. Cost optimum shield thicknesses are estimated for different operating conditions of the accelerator.

7. OFF-GAS FROM BETA-GAMMA LOW-LEVEL WASTE INCINERATION: ITALIAN TREATMENT SYSTEMS. E. MASOTTI. ENEA, Rome (Italy). Report No. ENEA-RT-DISP--86-2, Mar 1987, pp. 60.

The incineration process for radioactive wastes permits the maximum reduction in their volume and chemical reactivity, ensures safe disposal and contributes decisively to the protection of

man and the environment. The report indicates the regulatory requirements and limits for gaseous effluents and reviews the treatment systems for the off-gas produced by Italian experimental plants for the incineration of low-level beta-gamma wastes, including ion-exchange resins, radioactive mineral oils and solvents. Those plants which do not provide for off-gas treatment are excluded. Particular attention is given to the CRE Casaccia incineration plant because of the validity of the process and its high level of automation, which makes it easy to operate and follow the ALARA principles recommended by the ICRP. The plant also provides exceptional flexibility for experimentation. A critical examination of the described processes and plant concept and construction is also included.

8. MIXED WASTE CHARACTERIZATION AND CERTIFICATION AT THE NEVADA TEST SITE. T.A. KAWAMURA, R.L. DODGE, and P.K. FITZSIMMONS (Reynolds Electrical and Engineering Co., Inc., Las Vegas, NV (USA)). Report No. CONF-881054-42, 1988, pp. 8.

The Radioactive Waste Management Project at the Nevada Test Site (NTS) was recently granted interim status by the state of Nevada to receive mixed waste. The RCRA Part B permit application has been revised and submitted to the state. Preliminary indications are that the permit will be granted. In conjunction with revision of the Part B permit application, pertinent DOE guidelines governing waste acceptance criteria and waste characterization were also revised. The guidelines balance the need for full characterization of hazardous constituents with ALARA precepts. Because it is not always feasible to obtain a full chemical analysis without undue or unnecessary radiological exposure of personnel, process knowledge is considered an acceptable method of waste characterization. A balance of administrative controls and verification procedures, as well as careful documentation and high standards of quality assurance, are essential to the characterization and certification program developed for the NTS.

9. RIMS/SUP TM/ - RADIOLOGICAL INFORMATION MANAGEMENT SYSTEM: SOFTWARE PACKAGE EI-029-S86. Environment International (USA), Vol. 14, 1988. RIMS/sup tm/ has been developed for health physics record keeping and reporting. It provides for the management needs relating to radiological information control at a nuclear facility. The program is comprised of the following modules: Personnel Radiological Information, Radiological Work Permit, Radiation Survey Records, Access Control, ALARA Reporting, and Respirator and Survey Instrument Inventory Modules.

10. CHEMICAL DECONTAMINATION OF COMPONENTS AND SYSTEMS. H. WILLE and H.O. BERTHOLDT (Siemens A.G., Erlangen (Germany, F.R.)). Nuclear Europe (Switzerland), Vol. 8, Oct 1988, pp. 41-42.

Decontamination is becoming more and more important within the framework of procedural development for inspection and repair. To meet ALARA requirements, increasing dose rates of nuclear power plants with age make efforts to reduce occupational radiation exposure by decontamination essential. Steadily increasing attention is therefore being given to plant components or, better still, entire systems and loop decontamination. Furthermore, decontamination of a whole system or of a complete primary loop is sometimes easier to perform than cleaning of a single component. Since Siemens offers several decontamination processes there is always a single process or a combination of processes available to meet the specific requirements of a decontamination task. For decontamination of a complete system it is obvious that the process to be applied should be 'soft', i.e., low concentration. Use of a high chemical concentration 'hard' process would lead to too large amounts of waste. Due to problems and costs associated with waste handling for the decontamination of dismantled components, soft decontamination processes have now also been developed and applied.

11. DECISION AIDING TECHNIQUES FOR RADIOLOGICAL PROTECTION. G.A.M. WEBB and J. LOMBARD (National Radiological Protection Board, Chilton (UK); CEA, Fontenay-aux-Roses (France). Centre d'Etude sur l'Evaluation de la Protection dans le Domaine Nucleaire) International Atomic Energy Agency, Vienna (Austria). Proceedings of a Symposium - International conference on radiation protection principles, Sydney, AU, April 18, 1988.

As input to the work of a task group set up by the International Commission on Radiological Protection on optimization of protection, the authors reviewed a wide range of decision aiding techniques. The results of the review are summarized, with particular emphasis placed on the ability of the different techniques to deal with the various factors relevant to optimization of protection. Four quantitative decision aiding techniques are reviewed: simple cost-benefit analysis, extended cost-benefit analysis, multi-attribute utility analysis and multi-criteria outranking analysis. All can be used for optimization, but the choice of technique depends on the number of protection factors to be included in the optimization study, the complexity of the problem being studied, and the degree to which the factors considered can be quantified in monetary terms. If the problem being studied involves a number of factors but is treated using a simple technique, then the output from the quantitative analysis will be incomplete. However, it is shown that the overall result is dependent on the factors included and the criteria adopted, but not on the technique used.

12. SIMPLE METHOD FOR ISOLATING A SOURCE OF TRITIATED WATER VAPOR. J.H. METCALF (Sandia National Labs., Albuquerque, NM (USA)). Report No. SAND--88-2816C, 1988, pp. 15.

A source of tritiated water vapor existed in G tunnel at the Nevada Test Site which resulted in internal, whole-body exposure of personnel working in this complex. In an effort to apply the ALARA concept in this situation, it was necessary to identify the source of the tritium. Attempts were made to locate the source by

taking air samples on drierite for laboratory analyses, but this technique was not sensitive enough to pinpoint the source. A procedure will be described using the exchange of tritium in water which was effective in identifying the source of the tritium and allowing corrective actions to be taken.

13. BEST AVAILABLE TECHNOLOGY (ECONOMICALLY ACHIEVABLE) GUIDANCE DOCUMENT FOR THE HANFORD SITE. Westinghouse Hanford Co., Richland, WA (USA). Report No. WHC-EP--0137, Jul 1988, pp. 41.

This document provides Westinghouse Hanford Company (Westinghouse Hanford) and the US Department of Energy (DOE) with a step-by-step procedure for the identification and documentation of the Best Available Technology (BAT) economically achievable for treating liquid effluents on the Hanford Site. The BAT determination is a key element in the DOE strategy to eliminate use of the soil column for contaminated effluents disposal. Following application of BAT, a liquid effluent is considered suitable for discharge to the environment, including the soil column. Liquid effluents on the Hanford Site are currently disposed of in accordance with DOE orders that require protection of public health and safety, and to the extent possible, minimize adverse impacts on the environment. The determination of BAT on a liquid effluent will only occur after the effluent meets all applicable release limits. As a result, the application of BAT may involve an additional level of control, as well as contribute to the overall Hanford Site as low as reasonably achievable (ALARA) program.

14. SOLID LOW-LEVEL WASTE CERTIFICATION PROGRAM AT ORNL. M.C. WHEELER and L.D. BATES (Automated Sciences Group, Inc., Oak Ridge, TN (USA)). 1987 Oak Ridge model conference: Proceedings: Volume I, Part 2, Waste Management, Oak Ridge, TN, US, October 13, 1987. Analysis Corp., Oak Ridge, TN (USA). Report No. CONF-871075--Vol.1-Pt.2, pp. 149-168, 1987.

Over the past several years increased awareness and concern has developed regarding the proper generation, packaging, transporting and storing of all forms of waste matter, including Solid Low-Level Radioactive Waste (SLLW). Recently, special attention has been given to the proper segregation of waste types to prevent unauthorized and unregulated storage or disposal of waste matter. In April, 1986, the Oak Ridge National Laboratory (ORNL) embarked on a rigorous program to upgrade its SLLW management program. This waste certification effort included such elements as waste generator identification, training, procedure development, quality assurance, safety analysis and As Low As Reasonably Achievable (ALARA) studies. This paper provides an overview of this effort and identifies the techniques employed to develop a SLLW certification at ORNL.

15. ORNL IMPLEMENTATION OF NEW HEALTH AND SAFETY REQUIREMENTS (DOE ORDER 5480.11). J.S. ABERCROMBIE (Oak Ridge National Lab., TN (USA)). Report No. CONF-881054--11, 1988, pp. 8.

New mandates in radiological protection outlined in DOE Order 5480.11, include changes in the methodology for determining total radiation dose, ALARA program accountability, monitoring requirements, and standards for public entrance into controlled areas. The new order places distinct requirements concerning training at all DOE facilities. Radiation protection training requirements are addressed, including the effective communication of operations changes to all employees. This paper details the endeavors underway at ORNL in designing, developing, and delivering the training required by the new mandates. Strategies taken to reach the intended goals are explained. Efforts involve the design and implementation of the above mentioned radiation protection programs, a job-specific ALARA instructional package, and a Risk-Based Philosophy program matched to operational changes.

16. DISPOSAL/STORAGE CONTAINER DEVELOPMENT EXPERIENCE. R.W. MORROW, Jr., S.D. VAN HOESSEN, E. FOWLER, J.G. BARREIRA, and R.W. EMMETT (Oak Ridge National Lab., TN (USA)). Report No. CONF-881054--16, 1988, pp. 5.

Developmental work is currently underway at the Oak Ridge National Laboratory to design and manufacture a radioactive waste container suitable for both storage and disposal of radioactive wastes. The container is designed to fulfill the Department of Energy and Nuclear Regulatory Commission requirements for on-site storage, as well as the Nuclear Regulatory Commission's requirements for high integrity containers. The project also involves meeting the strict design and manufacturing ANSI/ASME NQA-1 guidelines. Special provisions of the container include a double containment system, with the inner barrier being corrosion resistant, the capability to monitor the internal cavity of the container, and off-gas venting capability. Further, yet related developmental work includes evaluating the cask for other varied uses, such as a processing cask, an ALARA shield, and even the possibility of Department of Transportation approval for an over-the-road transport cask.

17. ALARA: PROGRESS AND PROSPECTS. T.A. KHAN and J.W. BAUM (Brookhaven National Lab., Upton, NY (USA)). Report No. BNL-NUREG--41635, 1988, pp. 14.

This paper has two main topics. The first part describes the establishment and work of the ALARA Center; the second part presents some results of studies at the Center with international data on doses at PWR plants. This data then is used to reach a preliminary understanding of some of the factors that are causing high doses at PWRs. This approach should help in reducing occupational exposures in a more effective manner.

18. DEMONSTRATION OF NEW HANDLING EQUIPMENT CONFIGURATIONS FOR THE OVERPACKING OF THORIUM MATERIALS. G.S. MIHALOVICH and S.W. HEISLER, Jr.

(Westinghouse Materials Co. of Ohio, Cincinnati (USA)). Feed Materials Production Center. Report No. FMPC--2141, 1988, pp. 12.

A remote handling system has been designed for the removal and overpacking of thorium materials currently stored at the Feed Materials Production Center (FMPC). The FMPC is the Department of Energy (DOE) designated interim storage location for thorium materials. The nature of thorium materials indicated that a system to reduce exposure, assure environmental protection, and provide positive identification of the materials be devised. The result of this was the development and implementation of a remote handling system utilizing existing technologies but integrating them into a system that had previously not been used. The equipment used involves: remote controlled cranes, conveyers, sensing equipment and overpacks; a computer based central control and processing system that also provides cross data correlation; a shielded handling unit with multiple articulation capability and a environmentally isolated work area; and significant shielding at the operators station while the operator works in an enclosed, but mobile environment. These measures were prompted by both As Low As Reasonably Achievable (ALARA) considerations for the employees and protection for the environment. The overpacking is required due to the environmental deterioration that is occurring to both the building containers resulting in the need for this operation to be initiated to assure the continued safe storage of these materials.

19. HEALTH PHYSICS MANUAL OF GOOD PRACTICES FOR REDUCING RADIATION EXPOSURE TO LEVELS THAT ARE AS LOW AS REASONABLY ACHIEVABLE (ALARA). W.N. HERRINGTON, D.P. HIGBY, R.L. KATHREN, S.E. MERWIN, and G.A. STOETZEL (Pacific Northwest Lab., Richland, WA (USA)). Report No. PNL--6577, Jun 1988, pp. 169.

A primary objective of the US Department of Energy (DOE) health physics and radiation protection program has been to limit radiation exposures to those levels that are as low as

reasonably achievable (ALARA). As a result, the ALARA concept developed into a program and a set of operational principles to ensure that the objective was consistently met. Implementation of these principles required that a guide be produced. The original ALARA guide was issued by DOE in 1980 to promote improved understanding of ALARA concepts within the DOE community and to assist those responsible for operational ALARA activities in attaining their goals. Since 1980, additional guidance has been published by national and international organizations to provide further definition and clarification to ALARA concepts. As basic ALARA experience increased, the value and role of the original guide prompted the DOE Office of Nuclear Safety (ONS) to support a current revision. The revised manual of good practices includes six sections: 1.0 Introduction, 2.0 Administration, 3.0 Optimization, 4.0 Setting and Evaluating ALARA Goals, 5.0 Radiological Design, and 6.0 Conduct of Operations. The manual is directed primarily to contractor and DOE staff who are responsible for conduct and overview and ALARA programs at DOE facilities. The intent is to provide sufficient guidance such that the manual, if followed, will ensure that radiation exposures are maintained as low as reasonably achievable and will establish the basis for a formally structured and auditable program.

20. INEL HOT SHOP CONTAMINATION CONTROL DURING TMI-2 CORE DEBRIS HANDLING. L.D. CROFT and J.A. YEAZEL. Transactions of the American Nuclear Society (USA), Vol. 55, November 15, 1987, pp. 677-678.

The Idaho National Engineering Laboratory (INEL) houses unique hot cell/hot shop facilities used for remote handling and processing of highly contaminated and radioactive nuclear fuels and materials. These facilities are currently being used to handle, examine, and temporarily store Three Mile Island Unit 2 (TMI-2) core debris. These processes have created difficult but achievable challenges to control radioactive contamination spread within these facilities and maintain improving as-low-as-reasonably-achievable

(ALARA) trends. This paper presents approaches and resolution to radiation exposure reduction and contamination control challenges incurred as part of the processing of TMI-2 core debris.

21. INITIAL OPERATION AND MAINTENANCE TURNAROUND AT THE NEW WASTE CALCINING FACILITY. T.H. WAITE. Transactions of the American Nuclear Society (USA), Vol. 55, November 15, 1987, pp. 636.

The New Waste Calcining Facility (NWCF) has successfully processed a large quantity of radioactive liquid waste. The design concept of remote operation, maintenance, and repair has proven successful for both plant operating efficiency and maintenance capability. The system design and lessons learned from the system operation have been used to incorporate additional desirable system design changes and have resulted in a facility that will perform its objective with minimal personnel exposure and operational downtime. The NWCF processes high-level radioactive liquid waste by-products from reactor fuel reprocessing into solids using a process known as calcining. The solids provide an eight-to-one volume reduction and allow easier and safer long-term storage. Due to the high radioactivity of the liquids being processed and the severe operating environment, the design of the NWCF includes many specially designed features to allow remote handling and maintenance of the equipment. This capability was used to maintain and repair equipment both during the initial hot run, which began in September 1982 and ended in June 1984, and during the maintenance period that took place afterward. The remote capabilities have successfully allowed radiation exposures to permit achievement of reasonably achievable goals.

22. RADIOACTIVE DISCHARGES AND DECISION-MAKING ON RADIOLOGICAL PROTECTION OF THE PUBLIC IN THE UNITED KINGDOM AND FRANCE. D. HART (East Anglia Univ., Norwich (UK). School of Environmental Sciences). Assessing

and managing health and environmental risks from energy and other complex industrial systems. Proceedings of a workshop held in Paris, 13-17 October 1986, Paris, FR, October 13, 1986. International Atomic Energy Agency, Vienna (Austria) ; United Nations Environmental Programme ; World Health Organization. Report No. IAEA-TECDOC--453, pp. 97-106, Mar 1988.

The institutional arrangements for controlling marine discharges of radioactivity in the UK and France are outlined. The level of discharges from nuclear fuel reprocessing plants in the two countries are compared (namely the discharges from Sellafield, La Hague and Marcoule). A review is given of the changes made in discharge practices at Sellafield over the years, and the reasons for those changes. Difficulties in the use of the ALARA principle and of risk cost-benefit analysis for decision-making are discussed, including the problem of setting a monetary value for health detriment and for loss of human life. The relevance of the factor of public acceptance is discussed, and mention is made of the attempt to address this through a 'willingness-to-pay' survey in the UK. Finally, the difficulties of having an investment strategy which is formulated to take account of the sensitivity of public opinion are discussed.

23. COST-EFFECTIVENESS APPROACH TO COMPARISON OF NUCLEAR WASTE MANAGEMENT ALTERNATIVES. A REVIEW. S. VUORI and T. VIENO (Technical Research Centre of Finland, Espoo (Finland). Nuclear Engineering Lab.). Assessing and managing health and environmental risks from energy and other complex industrial systems. Proceedings of a workshop held in Paris, 13-17 October 1986, Paris, FR, October 13, 1986. International Atomic Energy Agency, Vienna (Austria) ; United Nations Environmental Programme ; World Health Organization. Report No. IAEA-TECDOC--453, pp. 309-316, Mar 1988.

There is an obvious need to consider the achievable level of safety of waste management in view of the costs involved. The feasibility of the cost-effectiveness approach for this purpose

has been studied in the framework of practical case studies. The analysis indicates that such an approach has clear benefits, but it also reveals several issues and ambiguities in its application. The waste management alternatives considered include various concepts for the disposal of low- and intermediate-level reactor and decommissioning wastes as well as of the unprocessed spent fuel versus disposal of high-level waste from reprocessing. The employed impact indicators describe both the individual and collective risks. In addition, indicators simultaneously giving a perspective into other risks in the society and a means to make a rank ordering of the alternative options should be considered. The cost-effectiveness ratios for collective risks vary in the range of ten to hundreds of millions US \$ per manSv. The examples considered also indicate that increased costs do not necessarily improve safety. Especially in the case of spent fuel and/or high-level waste disposal the extremely long time ranges bring about large uncertainties and ambiguities in applying the conventional concept of collective dose commitment. However, the application of cost-effectiveness approach is not fixed to any specific impact indicator and consequently the maximum individual risk can be used as a basis for decision-making as recently proposed by the NEA and ICRP.

24. ANALYSIS OF FACILITY DESIGN AND OPERATION FOR MAINTAINING PERSONNEL EXPOSURES AS LOW AS REASONABLY ACHIEVABLE. L.M. COCO (Westinghouse Electric Corporation, Pittsburgh, PA (USA)). Operational health physics: Proceedings, Denver, CO, US, February 9, 1976. Health Physics Society, Laramie, WY (USA). Proceedings Committee of the Central Rocky Mountain Chapter. Report No. CONF-760202--, pp. 647-653, 1976.

Increasing NRC emphasis on maintaining personnel exposures as low as reasonably achievable (ALARA), has resulted in much confusion and discussion of the subject. Health physicists in operating facilities and designers of new facilities are faced with the responsibility of developing and/or implementing programs

which will maintain personnel exposures ALARA, but have been given little regulatory guidance. Although several cost-benefit approaches have been recommended and general regulatory guidelines put forth in Regulatory Guide 8.8, the actual mechanics of situation analysis has not been clarified. The author has developed an ALARA program which, when used in conjunction with facility design and/or operation, anticipates potential problem areas and can contribute in reducing the total plant man-rem dose. The program provides the mechanics for analysis and evaluation of potential hazards, shielding and process design, and personnel usage.

25. TECHNICAL SAFETY APPRAISAL OF THE PLUTONIUM FINISHING PLANT, WESTINGHOUSE HANFORD COMPANY. L.C. BRINKERHOFF. USDOE Assistant Secretary for Environment, Safety and Health, Washington, DC. Office of Nuclear Safety. Report No. DOE/EH--0065, Apr 1988, pp. 232.

The Plutonium Finishing Plant (PFP) is located in the 200 West Area of the Hanford Site in Southeastern Washington State. The PFP is operated for DOE by the Westinghouse Hanford Company (WHC). Program direction is provided by DOE's Assistant Secretary for Defense Programs, Office of Nuclear Materials. Sixty-eight concerns are contained in this report which pertain to achieving compliance with some aspect of a DOE requirement or to achieving a greater level of safety. None of these concerns address a situation that presents a "clear and present danger" or requires expedited attention by the contractor. Thus, all are categorized as Category III for seriousness. Each concern has been characterized by relative hazard and compliance considerations. The results are indicated to the left of each concern by an alphanumeric designation giving the assigned relative hazard level (H) and compliance level (C) (e.g., H2/C3). These characterization are also summarized in Part 1 of Appendix B to this report. The concerns are tabulated in Part 2 of Appendix B. The reader is cautioned that to understand the full intent of any concern, it is necessary also to read its supporting bases,

which are found in Section III of the report. A total of 90 Performance Objectives in 13 subject areas have been addressed by this appraisal. Concerns associated with 42 of these Performance Objectives have been developed.

26. RESIDUAL RADIOACTIVITY GUIDELINES: A COST-BENEFIT CONSIDERATION. C. YU, J.M. PETERSON, and Y. YUAN (Argonne National Lab., IL (USA)). Transactions of the American Nuclear Society (USA), Vol. 55, November 15, 1987, pp. 112-114.

The US Dept. of Energy (DOE) has established radiological protection guidelines for cleanup of residual radioactive material located at sites administered under the Formerly Utilized Sites Remedial Action Program and Surplus Facilities Management Program. The methods and data for deriving site-specific guidelines and procedures for implementing DOE policy for reducing residual radioactivity to levels that are as low as reasonably achievable specify that, under reasonably foreseeable conditions, the annual radiation dose received by a member of the general public from residual radioactivity at a given site must not exceed 100 mrem (the basic dose limit) if the site is to be released for unrestricted use. This paper examines the relative effects of annual dose limits and cleanup costs on the allowable concentration of residual radioactivity. Two sites are analyzed: site A, a small site (1.4 ha) contaminated with uranium (an alpha emitter), and site B, a large site (77 ha) contaminated with Cs-137 (a beta-gamma emitter). For this analysis, uranium is assumed to consist of U-238, U-234, and U-235 in their naturally occurring activity concentration ratio.

27. HEALTH PHYSICS MANUAL OF GOOD PRACTICES FOR PLUTONIUM FACILITIES. L.W. BRACKENBUSH, K.R. HEID, W.N. HERRINGTON, L. KENOYER, L.F. MUNSON, L.H. MUNSON, J.M. SELBY, K.L. SOLDAT, and G.A. STOETZEL (Pacific Northwest Lab., Richland, WA (USA)). Report No. PNL-6534, May 1988, pp. 251.

This manual consists of six sections: Properties of Plutonium, Siting of Plutonium Facilities, Facility Design, Radiation Protection, Emer-

gency Preparedness, and Decontamination and Decommissioning. While not the final authority, the manual is an assemblage of information, rules of thumb, regulations, and good practices to assist those who are intimately involved in plutonium operations. An in-depth understanding of the nuclear, physical, chemical, and biological properties of plutonium is important in establishing a viable radiation protection and control program at a plutonium facility. These properties of plutonium provide the basis and perspective necessary for appreciating the quality of control needed in handling and processing the material. Guidance in selecting the location of a new plutonium facility may not be directly useful to most readers. However, it provides a perspective for the development and implementation of the environmental surveillance program and the in-plant controls required to ensure that the facility is and remains a good neighbor. The criteria, guidance, and good practices for the design of a plutonium facility are also applicable to the operation and modification of existing facilities. The design activity provides many opportunities for improvement of features to promote more effective protection and control. The application of "as low as reasonably achievable" (ALARA) principles and optimization analyses are generally most cost-effective during the design phase.

28. WORLDWIDE ACTIVITIES ON THE REDUCTION OF OCCUPATIONAL EXPOSURE AT NUCLEAR POWER PLANTS. T.A. KHAN and J.W. BAUM. (Brookhaven National Lab., Upton, NY (USA)). NUREG/CR--5158-Vol.1, Jun 1988, pp. 38.

This report is based on analysis of an informational data base set up at the Brookhaven National Laboratory ALARA Center. It is part of a project sponsored by the US Nuclear Regulatory Commission to monitor and evaluate research on dose reduction at nuclear power plants in the US and abroad. The main benefits to be expected from reducing occupational exposures are highlighted in the report, the chief causes of elevated doses are identified, and effective approaches to minimize radiation exposures are proposed. A wide range

of research activity is covered, including plant chemistry, cobalt reduction techniques, stress corrosion cracking, decontamination, remote tools and devices, and robotics. Advanced reactors, which are designed for low radiation exposures, are examined, and health physics technology programs which have been effective in reducing occupational exposure at various utilities are discussed. The highlights of the programs on dose reduction conducted by a number of countries are described, and comparisons are made of the collective occupational radiation dose equivalents for selected countries. The short and long term trends such studies are pointing to are evaluated. It is concluded that the efforts to improve dose reduction, both in the US and abroad, remain in healthy state but require continuing encouragement and further development.

29. RADIATION PROTECTION AND DOSIMETRY IN A REPROCESSING PLANT (in French). P. HENRY, C. CLECH, and C. LAFFAILLE (Compagnie Generale des Matieres Nucleaires (COGEMA), 78 - Velizy; Villacoublay (France)) Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris. Proceedings of a Symposium - International conference on nuclear fuel reprocessing and waste, Paris, FR, August 24, 1987, pp. 1079-1085.

In this paper the authors show how COGEMA conceive and exercise the radiation protection in the La Hague reprocessing plant, to limit the exposure of workmen according to the ALARA principle.

30. MONITORING FOR INTERNAL CONTAMINATION IN THE WAK-RE-PROCESSING PLANT. W. KRAUT, H.P. WICHMANN, and H.O. WILLAX (Wiederaufarbeitungsanlage Karlsruhe, Betriebsgesellschaft, Eggenstein-Leopoldshafen (DE)) Societe Francaise d'Energie Nucleaire (SFEN), 75 - Paris. Proceedings of a Symposium - International conference on nuclear fuel reprocessing and waste, Paris, FR, August 24, 1987, pp. 1073-1077.

By choosing suitable dosimeters individual monitoring of external radiation causes no

problems. The risk of exposure due to internal radiation, however, seems to us by far more difficult to control, especially in the case of unexpected events. Besides the general guideline of keeping any exposure as low as possible, routine monitoring for internal contamination is required by a German BMI recommendation whenever 5% of the annual limits of intake (ALI) are supposed to be exceeded. In the WAK ca. 500 workers (staff and contractors) with working permits for contaminated areas are to be observed using the monitoring program.

31. KEY ISSUES IN THE IMPLEMENTATION OF ALARA IN OPERATIONS. J. LOCHARD and J.R. CROFT (Centre d'Etude sur l'Evaluation de la Protection dans le Domaine Nucleaire, 92 - Fontenay-aux-Roses (France); National Radiological Protection Board, Leeds (UK)). Journal of Radiological Protection (UK), Vol. 8, Jun 1988, pp. 87-96.

This paper focuses on how the ALARA concept can be practically implemented in operational situations and in particular on the value of using a structured approach and the problem of the collection of relevant dosimetric data. The roles of ALARA audits, quantitative decision aiding techniques and predictive ALARA plans are discussed and some practical examples provided. A basis requirement in the pursuance of optimization is to be able to quantify the doses predicted to be associated with the various components of a person's work. Many dosimetric systems in use simply cannot provide these data and the need to provide task specific dosimetry with a feedback mechanism into dose modelling is discussed.

32. HEALTH PHYSICS MANUAL OF GOOD PRACTICES FOR ACCELERATOR FACILITIES. W.R. CASEY, A.J. MILLER, J.B. McCASLIN, and L.V. COULSON (Stanford Linear Accelerator Center, Menlo Park, CA (USA); Brookhaven National Lab., Upton, NY (USA)); SLAC-327, Apr 1988, pp. 140.

It is hoped that this manual will serve both as a teaching aid as well as a useful adjunct for program development. In the context of appli-

cation, this manual addresses good practices that should be observed by management, staff, and designers since the achievement of a good radiation program indeed involves a combined effort. Ultimately, radiation safety and good work practices become the personal responsibility of the individual. The practices presented in this manual are not to be construed as mandatory rather they are to be used as appropriate for the specific case in the interest of radiation safety. As experience is accrued and new data obtained in the application of this document, ONS will update the guidance to assure that at any given time the guidance reflects optimum performance consistent with current technology and practice. The intent of this guide therefore is to: define common health physics problems at accelerators; recommend suitable methods of identifying, evaluating, and managing accelerator health physics problems; set out the established safety practices at DOE accelerators that have been arrived at by consensus and, where consensus has not yet been reached, give examples of safe practices; introduce the technical literature in the accelerator field; and supplement the regulatory documents listed in Appendix D. Many accelerator health physics problems are no different than those at other kinds of facilities, e.g., ALARA philosophy, instrument calibration, etc. These problems are touched on very lightly or not at all. Similarly, this document does not cover other hazards such as electrical shock, toxic materials, etc. This does not in any way imply that these problems are not serious.

33. ESTIMATING COLLECTIVE DOSE IN NUCLEAR FACILITIES, WITH EMPHASIS ON THE DESIGN PROCESS. S. COHEN and B. MANN (Atomic Industrial Forum Bethesda, MD), 1987.

The report presents a more accurate, systematic method than has been available previously for predicting worker doses which might be incurred during routine and non-routine work in radioactive areas. Besides assisting regulators with an analysis of the 'potential impact on radiological exposures of facility employees' now required under the new backfit rule (10 CFR

50.109c), this predictive model will also help licensees conserve dollars as well as dose because it can be employed very early in the engineering design phase of a modification, when adjustments can still be made easily to change orders. Such early estimates make good business sense because they will facilitate planning, labor loading, costing, resource and equipment scheduling, and overall coordination of both single and repetitive projects. Also, with the support of corporate management, radiation protection coordinators can introduce the model into training programs to acquaint design engineers and others with dose calculation techniques. The importance assigned by nuclear industry senior management to the principle of ALARA and the reduction of collective worker dose is measured, in large part, by demonstrated efforts to integrate the control of radiation exposure fully into the overall planning function of nuclear facility management. That integration will be fostered through the use of this approach.

34. DECONTAMINATION AND DECOMMISSIONING OF THE WEST VALLEY REPROCESSING PLANT. H.F. DAUGHERTY and R. KEEL. Transactions of the American Nuclear Society (USA), Vol. 52, June 15, 1986, pp. 34-35.

This paper addresses the methods being used to decontaminate and decommission the various cells and operating spaces of the former West Valley Fuel Reprocessing Facility, located in West Valley, New York. Topics addressed include the application of innovative techniques to permit decontamination and decommissioning (D & D) of processing cells with initial radiological conditions including radiation levels > 100 R/h and extremely high levels of both alpha and beta/gamma airborne concentrations exceeding 10,000 times the maximum permissible concentration.

35. ENVIRONMENTAL MONITORING PROGRAM AT THE FEED MATERIALS PRODUCTION CENTER. K.A. MALONE. EC Design and Technology, Inc., Fairfield, OH (USA). Report No. FMPC/Sub--009, Mar 1988, pp. 72.

The environmental monitoring program at the Feed Materials Production Center (FMPC) consists of monitoring and surveillance activities which support facility regulatory compliance. These activities include environmental sampling, data management and analysis, and data reporting to support air compliance, water compliance, and solid waste compliance at the FMPC. The Environmental Compliance subsection of the Environment, Safety and Health Department is responsible for planning, implementing, and reporting environmental monitoring at the FMPC. Program management is supported by quality assurance requirements, standard operating procedures, chain of custody procedures, data management and analysis strategies, and personnel training. The development and implementation of environmental monitoring and surveillance programs at the FMPC is guided by DOE Orders and federal and state regulations which specify radiological and non-radiological parameters for facilities such as the FMPC. These parameters and associated exposure limits support the ALARA principle of maintaining exposures to the public and the environment to as low as reasonable achievable. Monitoring and surveillance activities at the FMPC are grouped into three categories: air compliance, water compliance, and solid waste compliance. These categories consist of individual sampling programs designed to effectively monitor the various media within each category. Sampling locations, schedules, and analytical parameters are specified in this report for each program when applicable.

36. DRYING OF ION-EXCHANGE RESIN AND FILTER MEDIA: [FINAL REPORT]. R.E. BURNHAM and D.S. HILLSTROM (Nuclear Packaging Corp., Federal Way, WA (USA)). Report No. DOE/ID/12639--T1, 1987, pp. 35.

Ion exchange resins, filter media and sludges are currently either dewatered or solidified, for stabilization, prior to disposal at a low level waste facility. The Nuclear Packaging, Inc. (NuPac) Resin Drying System was developed and placed in commercial service to provide a

system which meets the regulatory requirements for free standing water, with a relatively short process duration, requiring no chemical or material addition, and utilizing more efficiently the available container volume than was previously achievable. The volume reduction of 10 to 14% associated with this system coupled with reductions in operator radiation exposure provide a significant economic advantage over other options a utility may have for processing resins and sludges.

37. THE UNIVERSITY OF MISSOURI RESEARCH REACTOR FACILITY CAN MELTER SYSTEM. C.B. JR. EDWARDS, O.L. OLSON, R. STEVENS, and R.M. BRUGGER (University of Missouri, Columbia (USA)). Research Reactor Facility. Report No. DOE/ID/12638--T1, 30 Apr 1987, pp. 81.

At the University of Missouri Research Reactor (MURR), a waste compacting system for reducing the volume of radioactive aluminum cans has been designed, built and put into operation. In MURR's programs of producing radioisotopes and transmutation doping of silicon, a large volume of radioactive aluminum cans is generated. The Can Melter System (CMS) consists of a sorting station, a can masher, an electric furnace and a gas fired furnace. This system reduces the cans and other radioactive metal into barrels of solid metal close to theoretical density. The CMS has been in operation at the MURR now for over two years. Twelve hundred cu ft of cans and other metals have been reduced into 150 cu ft of shipable waste. The construction cost of the CMS was \$4950.84 plus 1680 man hours of labor, and the operating cost of the CMS is \$18/lb. The radiation exposure to the operator is 8.6 mR/cu ft. The yearly operating savings is \$30,000.

38. DECONTAMINATION TO ACHIEVE SITE RELEASE CRITERIA. A.R. SCHULMEISTER (General Electric Co., Shippingport, PA (USA)). Report No. DOE/SSDP--0041, 1988, pp. 8.

The Shippingport Station Decommissioning Project presented unique opportunities in the management and performance of component

and area decontamination to meet established site release criteria. Decontamination activities were performed to meet the following overall project objectives (criteria): (1) internal decontamination of radioactively contaminated components to meet low specific activity (LSA) shipping limits; (2) reduction of component radiation levels to as low as reasonably achievable (ALARA) for subsequent dismantlement, packaging and off site disposal; (3) reduction of surface contamination to qualify steel structurals for unrestricted off site release; (4) reduction of surface contamination to qualify concrete structurals for unrestricted on site release.

39. IMPLEMENTATION OF SITE RELEASE CRITERIA AT SHIPPINGPORT. K.J. EGER, M.J. GALGOUL, and J.B. CARR (USDOE Shippingport Station Decommissioning Project Office, PA). Report No. DOE/SSDP--0040, 1988, pp. 4.

Criteria for the unconditional release of the site, and materials left thereon following decommissioning of the Shippingport Atomic Power Station (SAPS) Site were issued by the Department of Energy (DOE) in January 1987. The criteria were "100 mrem/year via the worst case scenario" and "ALARA." Subsequently, GE Nuclear Energy, the Decommissioning Operations Contractor (DOC), generated a "Final Consolidated Implementation Plan." This document defined the scenarios by which the final dose would be calculated, the limiting conditions for the "as decommissioned" site and the methodology for demonstrating the achievement of these conditions. The plan is being used concurrently with the decommissioning operations to demonstrate that the site will meet the criteria issued.

40. WORKER DOSE AFTER 10CFR61. M.T. RYAN (Chem-Nuclear Systems, Inc., P.O. Box 726, Barnwell, SC 29812 (USA)). Proceedings of a symposium - waste management '87, Tucson, AZ, US, March 1, 1987.

With the execution and implementation of 10CFR61 into low-level waste management practices at Chem-Nuclear's facility in Barnwell,

particular attention has been paid to occupational radiation exposure. The concern for occupational radiation exposure arose from the segregation of higher activity waste as required in 10CFR61. The implementation of separate waste burial trenches (for Class A, B, C wastes) at Barnwell included operational review, procedure modification, ALARA review, and operational health physics monitoring to ensure the effective control. In this paper a detailed statistical assessment of occupational radiation exposure in a variety of work groups is presented. In general, the implementation of 10CFR61 has not had an adverse impact. There are no statistical significant increases in overall radiation exposure. This paper provides a detailed statistical assessment of radiation exposure and accounts for waste volumes and total radioactivity contained in buried wastes. The experience at Chem-Nuclear is compared to engineering low-level waste burial facilities in Europe.

41. SOLVING HIGH-LEVEL WASTE DISPOSAL PROBLEMS. S.D. LANDSMAN (Pacific Northwest Lab., Richland, WA (USA)). Report No. PNL-SA--15438, Dec 1987, pp. 5.

Techniques developed for disposing of high-level waste from a hot cell proved to be highly successful in reducing the radiation exposure of the workers involved. The people responsible for developing these techniques were primarily from the radiation protection profession. This paper describes the problems encountered and solutions developed during the 9-month period of this operation.

42. AN APPROACH TO INNOVATIVE REMOTE SPENT FILTER HANDLING. L.J. DERITIS, M. BROWNSTEIN, and A. CORRAL (ATCOR Engineered Systems, Inc., Avon, CT 06001 (USA)). Proceedings of a symposium - waste management '87, Tucson, AZ, US, March 1, 1987, pp. 351-356.

Reducing both man-rem exposure and manpower expenditure during normal spent filter handling and maintenance operations is essential to routine nuclear power plant operation. Unfortunately, lack of either attention

or operating experience on the part of equipment designers and engineers frequently prevails. This in turn prevents the use of standardized components, integrated systems, and simplified approaches to accomplishing the necessary task of remote spent filter handling in accordance with ALARA principles. ATCOR recently completed a successful acceptance test and supply of a Spent Filter Handling System for the Georgia Power Company, Vogtle Nuclear Power Station. This paper discusses ATCOR's design plan and implementation for achieving a totally integrated system for efficient, remote handling of spent filters with maximum protection to operating personnel. The paper also includes a description of the approach followed for pre-delivery performance testing. The resulting impact of this on the final design, which ultimately led to repeatable successful plant operation, is also discussed. All problems encountered, along with their corresponding solutions are discussed in detail. Future designers and operators of these types of interface equipment and interfaced common pitfalls are presented, thereby increasing their expertise and likelihood of successful plant operations.

43. LARGE-SCALE SPENT-FUEL CASK RECEPTION AND DRY UNLOADING AT LA HAGUE. C. HUTCHISON and P. LEMAISTRE. Transactions of the American Nuclear Society (USA), Vol. 54, June 7, 1987, pp. 60-61.

The two new commercial reprocessing plants at La Hague, France, must be able to receive, unload, store, and reprocess a combined total of 1600 tonne U of spent fuels each year when they begin full-scale active operation at the end of 1988 and in 1991. This rate of reception and production required several innovative design concepts to meet the goals of safety and high availability. The TO facility includes a receiving and shipping building, a cask preparation cell, and a cask unloading and loading cell, which features by unloading of casks with a maximum residual heat release of 85 kW. A major requirement of the design was remote and automatic operation of the system with high

equipment reliability in a centralized process control mode. The reduction of individual dose rates to < 0.5 rem/yr was also taken into consideration. The guaranteed production capacity is 800 tonne U in a period of 215 days.

44. SAMPLING PLAN FOR THE UNCONDITIONAL RELEASE OF BUILDING MATERIALS AND SIMILAR LARGE COMPONENTS ENCOUNTERED IN DECOMMISSIONING. D.L. GARDNER and K.J. EGER (General Electric Co., Shippingport, PA). Transactions of the American Nuclear Society (USA), Vol. 54, June 7, 1987, pp. 48-50.

The most efficient sampling plan for the unconditional release of building materials and similar large components encountered in decommissioning can be developed without requiring 100% coverage by measurements on all surfaces. This plan is being implemented in the decommissioning of the Shippingport Atomic Power Station. The backbone of the plan is the checking and rechecking of the candidate material so that there is a high degree of assurance that it is releasable before the final survey is performed. This sampling plan process is summarized.

45. ALARA CONSIDERATIONS AND RADIOACTIVE WASTE DISPOSAL IN THE UNITED KINGDOM. H. BEALE and A.W. DAVIES (Nuclear Industry Radioactive Waste Executive, Harwell, UK) Institution of Nuclear Engineers, London (UK). Proceedings of a Symposium - Symposium on ALARA - quantitative optimization techniques London, GB, September 18, 1986, J.R.A. Lakey and J.D. Lewins (eds.), pp. 126-139.

The implications of the ALARA principle when applied to the disposal of solid radioactive wastes in the UK are discussed. Following a brief outline of the regulatory aspects, an introduction to UK Nirex Ltd is given, the facilities Nirex wishes to develop briefly discussed and the methodology of site selection outlined. The radiological principles to be applied by the regulatory bodies are discussed. This leads to the introduction of the concept of risk

and is followed by ALARA optimization and repository development.

46. ROLE OF QUANTITATIVE OPTIMIZATION TECHNIQUES IN ASSESSMENT OF BEST PRACTICABLE ENVIRONMENTAL OPTIONS FOR RADIOACTIVE WASTE MANAGEMENT. P.D. JOHNSTON

(Department of the Environment, London, UK) Institution of Nuclear Engineers, London (UK). Proceedings of a Symposium - Symposium on ALARA - quantitative optimization techniques, London, GB, September 18, 1986, J.R.A. Lakey and J.D. Lewins (eds.), pp. 118-125.

The interpretation of the Best Practicable Environmental Option (BPEO) and ALARA concepts in radioactive waste management is given. The quantitative analysis of the financial and radiological impacts of different options for waste management is discussed. Finally, the role of quantitative multi-attribute analysis in the DOE's assessment of BPEOs for radioactive waste is described.

47. ALARA IN PRACTICE. P.W. MUMMERY and A.R. ANDERSON (British Nuclear Fuels Ltd., London) Institution of Nuclear Engineers, London (UK). Proceedings of a Symposium - Symposium on ALARA - quantitative optimization techniques, London, GB, September 18, 1986, J.R.A. Lakey and J.D. Lewins (eds.), pp. 104-114.

A range of practical examples of ALARA in practice concerning radioactive liquid discharges from the Sellafield Reprocessing Plant are given. These include the use of ion-exchange measures to reduce radio-caesium discharges, the decision to construct the enhanced actinide removal plant (EARP), the influence of changing habits and parameters of critical group exposure on considerations of reduction in liquid discharges and finally safe plant operation considerations.

48. METHODS FOR REDUCING OCCUPATIONAL EXPOSURES DURING THE DECOMMISSIONING OF NUCLEAR FACILITIES. REPORT OF A TECHNICAL COMMITTEE MEETING HELD IN

PRAGUE 21-25 OCTOBER 1985. IAEA (Vienna, Austria), October 21, 1985, pp. 42.

The decommissioning and decontamination of nuclear facilities is a topic of great interest to many member states of the International Atomic Energy Agency (IAEA) because of the large number of older facilities which have been or soon will be retired from service. This report is a review of the current state of knowledge concerning methods for reducing occupational exposures during the decommissioning of nuclear facilities. This report focuses on water cooled nuclear power plants but, in addition, other major nuclear facilities are briefly discussed to determine how they differ from nuclear power plants in this regard. The information presented should be useful to those responsible for or interested in designing or constructing nuclear facilities or in the planning or implementing of the decommissioning of such installations.

49. BNFL, SELLAFIELD: REVISION OF AUTHORIZATION TO DISCHARGE LIQUID RADIOACTIVE WASTE. TECHNICAL APPRAISAL OF INFORMATION PROVIDED BY BNFL. Ministry of Agriculture, Fisheries and Food (London, England), 1986, pp. 49.

The proposed authorization covers the reprocessing programme at Sellafield. The site ion exchange effluent treatment plant and a salt evaporator which have recently come into operation enable substantial reductions in radionuclide discharges to be made. This has made it possible to revise the current authorization to reduce the permitted discharge levels. This document embodies a technical appraisal of the information submitted by British Nuclear Fuels Ltd. to the authorizing authorities. BNFL were required to review the waste streams which contribute to the discharges to sea from the Sellafield site and to justify their radioactive content. The principles of BPM (best practical means) and ALARA (as low as reasonably achievable) have to be followed within the numerical limits set. The derivation of these limits proposed in the revised authorization are given. The following points are covered: the

radiological protection principles of the International Commission on Radiological Protection, the technical factors (such as fuel characteristics, reprocessing rates and availability of treatment plant) that have been taken into account in this appraisal, an explanation as to how the limits on discharges of these radionuclides have been derived, an assessment of effects of discharges allowed under this authorization in terms of the radiation dose to critical groups and to the general population and finally, the possible dose consequences of the proposed authorized limits.

50. DECOMMISSIONING PLAN DEPLETED URANIUM MANUFACTURING FACILITY. D.E. BERNHARDT, J.D. PITTMAN, and S.V. PREWETT (Rogers and Associates Engineering Corp., P.O. Box 330, Salt Lake City, UT 84110-0330). Proceedings of a Symposium - waste management '87, Tucson, AZ (US), March 1, 1987, pp. 533-538.

Aerojet Ordnance Tennessee, Inc. (Aerojet) is decommissioning its California depleted uranium (DU) manufacturing facility. Aerojet has conducted manufacturing and research and development activities at the facility since 1977 under a State of California Source Materials License. The decontamination is being performed by a contractor selector for technical competence through competitive bid. Since the facility will be released for uncontrolled use it will be decontaminated to levels as low as reasonably achievable (ALARA). In order to fully apply the principles of ALARA, and ensure the decontamination is in full compliance with appropriate guides, Aerojet has retained Rogers and Associates Engineering Corporation (RAE) to assist in the decommissioning. RAE has assisted in characterizing the facility and preparing contract bid documents and technical specifications to obtain a qualified decontamination contractor. RAE will monitor the decontamination work effort to assure the contractor's performance complies with the contract specifications and the decontamination plan. The specifications require a thorough cleaning and decontamination of the facility, not

just sufficient cleaning to meet the numeric cleanup criteria.

51. WASTE HOIST TECHNOLOGY. Morrison-Knudsen Engineers, Inc., San Francisco, CA (USA) ; Fluor Technology, Inc., Irvine, CA (USA). Report No. DOE/CH/46656--06, Sep 1987, pp. 119.

This report evaluates unique applications of current hoisting design practice that satisfy the waste package transfer functional requirements for a nuclear waste repository in salt. Hoisting system configurations that could provide the desired service for this important function are identified. Implications on hoist design for both shielded and unshielded waste package casks are evaluated. The report also presents specially engineered features which limit exposure levels to as low as is reasonably achievable (ALARA) criteria.

52. ADVANCED RADIOACTIVE WASTE ASSAY METHODS: FINAL REPORT. J.E. CLINE, D.E. ROBERTSON, and S.E. DEGROOT. Science Applications International Corp., Rockville, MD (USA) ; Pacific Northwest Lab., Richland, WA (USA) ; Stock Equ. Report No. EPRI-NP--5497, Nov 1987, pp. 281.

This report describes an evaluation of advanced methodologies for the radioassay of low power-plant low-level radioactive waste for compliance with the 10CFR61 classification rules. The project evaluated current assay practices in ten operating plants and identified areas where advanced methods would apply, studied two direct-assay methodologies, demonstrated these two techniques on radwaste in four operating plants and on irradiated components in two plants, and developed techniques for obtaining small representative aliquots from larger samples and for enhancing the Ce-144 activity analysis in samples of waste. The study demonstrated the accuracy, practicality, and ALARA aspects of advanced methods and indicates that cost savings, resulting from the accuracy improvement and reduction in sampling requirements can be significant.

53. DECOMMISSIONING OF A MIXED OXIDE FUEL FABRICATION FACILITY. S. BUCK and A.P. COLQUHOUN (British Nuclear Fuels Ltd., Sellafield) Institution of Mechanical Engineers, London (UK) ; British Nuclear Energy Society, London ; Institution of Nuclear Engineering. Proceedings of a symposium - IMechE international conference on decommissioning of major, London, GB, October 11, 1988, pp. 67-75.

A lead project in the decommissioning of plant heavily contaminated with plutonium and other actinides is being used to test in-situ assay and size reduction methods, and improvements in temporary containment and contamination control. The project is also yielding data on manpower requirements, personnel radiation uptake, and waste arisings to help in the planning of future schemes.

54. SAFETY APPRAISAL OF THE PROGRAM TO CONTROL AND MONITOR WORKER INTERNAL RADIATION EXPOSURE AT THE FEED MATERIALS PRODUCTION CENTER FERNALD, OHIO (WITH TRANSMITTAL MEMORANDUM). USDOE Assistant Secretary for Environment, Safety and Health, Washington, DC (USA). Office of Safety Policy and Standards Report No. DOE/EH--0086, Mar 1989, pp. 149.

The purpose of this Appraisal was to determine the safety adequacy of the Feed Materials Production Center (FMPC) radiological protection program as it related to the control and monitoring of worker internal exposure to radioactive materials. The appraisal was prompted by the concern of a Federal agency that the plant was unsafe for its workers based on preliminary indications that uranium was present in the urine of one of its inspectors after a short visit to the site. The concern led to the termination of further visits by the Agency's inspector. The inspector did not enter process areas of the site where internal exposures, if any, would be most likely to occur. In addition, no FMPC employees in the same area at the same time as the inspector showed any positive urine results. Findings of the appraisal team do not

support the plausibility of an actual intake by the inspector during a site visit.

55. COLLECTION AND REMOVAL OF RADIOACTIVE WASTE, COMING FROM UNIVERSITIES, HIGH SCHOOLS AND ACADEMIC HOSPITALS. P.H. DIGNUM, A. ZWIGT, and LAVM - WERKGROEP STRALINGSHYGINE COLLABORATION. Interuniversitaire Adviescommissie Veiligheids- en Milieuwetgeving (LAVM), Amsterdam (Netherlands). Onderwerpgröe. Report No. LAVM--9, 1985, pp. 58 (in Dutch).

In radionuclide laboratories of universities, radionuclides are employed for biomedical, chemical and physical research. In university hospitals, radionuclides are employed in vivo for diagnostic and therapeutic purposes and in vitro in tracer methods in chemical analysis. In general it concerns radioactive materials in dispersive form such as gases, fluids or powders (open sources). During operation with open sources radioactive waste originates with low specific activity. Regulations and recommendations concerning the collection and separation, transport, processing and eventual storage of this low-level radioactive waste are dealt with.

56. TAMING WINDSCALE'S PILES. A.L. ADAMS (UKAEA Windscale Nuclear Power Development Labs.). Professional Engineering (UK), Vol. 2, Jan 1989, pp. 10-12.

The options as to what to do with the Windscale piles are being assessed before a final decision on decommissioning is made. Both piles were shutdown in 1957 following the fire in the pile number 1. Pile 1 still contains 22 tons of natural uranium fuel. The details of graphite moderator content, biological shielding and other components and containment are given. The fuel and isotope channels in pile 2 have been examined and the air and water ducts have been inspected. The chimneys of both piles are contaminated and all entrances have been sealed. Before any work starts the air outlet ducts will be sealed from the chimney and a ventilation system installed. A manipulator is being prepared to remove the remaining fuel

elements from both piles. The proposed decommissioning programme for both piles is outlined. (U.K.).

57. DECOMMISSIONING OF EIGHT GRAPHITE-MODERATED REACTORS. H.F. DAUGHERTY and C.E. MILLER (Westinghouse Hanford Co., Richland, WA (USA); Department of Energy, Richland, WA (USA). Richland Operations Office) Institution of Mechanical Engineers, London (UK) ; British Nuclear Energy Society, London ; Institution of Nuclear Engineering. Proceedings of a symposium - IMechE international conference on decommissioning of major, London, GB, October 11, 1988, pp. 11-20.

Eight graphite-moderated plutonium reactors and associated support facilities were constructed on the Hanford Site in Washington state between 1942 and 1955 for the production of plutonium. Four alternatives are presented for evaluation in decommissioning these reactors and ancillary structures.

58. DECOMMISSIONING OF A 5 MW RESEARCH REACTOR. R.G. STRUSS (Ames Lab., IA (USA)) Institution of Mechanical Engineers, London (UK) ; British Nuclear Energy Society, London ; Institution of Nuclear Engineering. Proceedings of a symposium - IMechE international conference on decommissioning of major, London, GB, October 11, 1988, pp. 51-53.

The complete decommissioning of a research reactor is described. Planning and execution of all activities, including schedules, budgets, waste management, health physics and subcontracted operations are presented. Flexibility in operations was obtained by using the operating staff as the decommissioning progressed. Totals for waste shipments and costs are given. Final site conditions are presented along with a description of the subsequent use of the facility.

59. DISMANTLING OF ACTIVE PLANT AT THE PETTEN HIGH FLUX REACTOR. J.A. COLLIS-SMITH, M.R. CUNDY, and R.J. SWANENBURG DE VEYE (GEC Energy Systems Ltd., Leicester (UK); Commission of

the European Communities, Petten (Netherlands). Joint Nuclear Research Center; Netherlands Energy Research Foundation, Petten) Institution of Mechanical Engineers, London (UK) ; British Nuclear Energy Society, London ; Institution of Nuclear Engineering. Proceedings of a symposium - IMechE international conference on decommissioning of major, London, (UK), October 11, 1988, pp. 33-42.

The reactor vessel, support structure and other highly irradiated components at the Petten High Flux Reactor in the Netherlands were decommissioned over a period of seven months as part of an extensive upgrading programme. Following preparation of a detailed vessel dismantling scenario, special tools for underwater cutting, debris control and visual aid were developed in a dummy pool and were proved by dismantling a spare vessel. Radiological exposure during dismantling was minimized by preliminary studies, underwater gamma-ray measurements during maintenance outages and detailed pre-planning. Structural steel in the pool walls adjacent to the beam tubes and thermal columns turned out to be more highly activated than expected and operational radiation doses were kept to acceptable levels by a combination of permanent and temporary shielding. Almost six tonnes of medium-active waste were generated during dismantling. Part of the original thermal column has been stored intact; other waste was cut, packaged and put into long-term storage. The time spent in development of the decommissioning activities paid dividends in the successful completion of the project. Much useful data and experience has been accumulated which would be of benefit to similar projects elsewhere.

60. DECOMMISSIONING IN A MIXED OXIDE FUEL FABRICATION PLANT. J. DRAULANS (Belgonucleaire, Dessel (Belgium)) Institution of Mechanical Engineers, London (UK) ; British Nuclear Energy Society, London; Institution of Nuclear Engineering. Proceedings of a symposium - IMechE international conference on decommissioning of

major, London (UK), October 11, 1988, pp. 95-102.

Information is given concerning the different ways and techniques applied during the decommissioning of glove boxes and equipment inside the BELGONUCLEAIRE Mixed Oxide fuel fabrication plant. Ways and techniques have been selected on the basis of the glove box dimensions, the capability of being removed out of the building and/or being transported on the public road, the inner contamination level etc.... Different techniques applied are explained.

61. NUCLEAR DECOMMISSIONING - PRACTICAL EXPERIENCE OF THE PRIVATE SECTOR. A.W. BRANT (WasteChem Ltd., Bramhall (UK)) Institution of Mechanical Engineers, London (UK) ; British Nuclear Energy Society, London ; Institution of Nuclear Engineering. Proceedings of a symposium - IMechE international conference on decommissioning of major, London (UK), October 11, 1988, pp. 103-110.

There is a growing requirement for decommissioning of redundant nuclear facilities. This has led to a number of opportunities for private sector organizations to carry out the work. This paper, based on two actual projects in the United Kingdom, outlines the input required from private sector contractors in executing such work.

62. DECOMMISSIONING PROGRAMME EXPERIENCE BY THE UNITED STATES DEPARTMENT OF ENERGY'S SURPLUS FACILITIES MANAGEMENT PROGRAMME. W.E. MURPHIE (USDOE, Washington, DC) Institution of Mechanical Engineers, London (UK) ; British Nuclear Energy Society, London ; Institution of Nuclear Engine. Proceedings of a symposium - IMechE international conference on decommissioning of major, London (UK), October 11, 1988, pp. 43-49.

The U.S. Department of Energy's Remedial Action Programs are identified and the role of the Surplus Facilities Management Program (SFMP) highlighted, including the SFMP long range plan. The paper focuses on programmatic

experience within the SFMP that has influenced program policy, including influences due to environmental compliance and inadequate project planning. This is provided from the programmatic versus project specific perspective. Finally, the relevance of current SFMP projects to future large scale commercial decontamination and decommissioning activities are discussed, specifically highlighting the relevance of the Shippingport project.

63. PRELIMINARY SITE SURVEY REPORT FOR THE SACANDAGA SITE, GLENVILLE, NEW YORK. W.D. COTTRELL, R.D. FOLEY, and R.F. CARRIER. Oak Ridge National Lab., TN (USA). Report No. ORNL/RASA--88/102, Mar 1989, pp. 28.

Of the numerous original buildings, only Building P, the Critical Assembly Building, remains standing. Front and rear views of the small, concrete block building are shown. The remaining structure is situated on approximately 51 acres of land, most of which is covered by a heavy growth of trees and brush. An abundance of rubbish and dismantled structural materials is strewn throughout the site. It is estimated that the total inventory of radioactive materials in the site included approximately 300 kg of uranium in the form of solid metal discs, approximately 8 mg of plutonium on platinum discs covered with plastic, and approximately 3 mg of radium sealed in a radium/beryllium source. Both U-238 and U-235 in solid metal form were used in experiments for breeder reactor studies. This type of assessment typically involves a two-stage process. The first part consists of a site visit and preliminary scoping survey to verify that no radiological hazard exists at a specific site, and to collect sufficient data to plan the second part of the process--the more comprehensive radiological survey. The preliminary radiological scoping survey discussed in this report was conducted. Compared to the large size of the property, the survey was limited in scope, including only the 5 to 6 acres enclosed within the fence and concentrating on those areas having the greatest possibility of containing residual radioactive material.

64. RESULTS OF THE RADIOLOGICAL SURVEY AT METPATH INCORPORATED, 1 MALCOLM AVENUE, TETERBORO, NEW JERSEY (TJ003). R.D. FOLEY and L.M. FLOYD. Oak Ridge National Lab., TN (USA). Report No. ORNL/TM--11118, Feb 1989, pp. 15.

A radiological survey of the commercial property at 1 Malcolm Avenue, Teterboro, New Jersey, was conducted during 1986 and 1988. The survey and sampling of the ground surface over a major portion of the property were carried out. An unused parking area was surveyed. Conversations with property owners revealed that originally this site was part of a single property of approximately 107 acres owned entirely by the Bendix Aerospace Corporation. During this period of total property ownership, Bendix was licensed by the Nuclear Regulatory Commission to use thorium in an on-site Navy/Bendix process. Around 1976, the property was subdivided into three parcels, and one parcel of about 30 acres was purchased by Metpath Incorporated. Subsequent to this purchase, Metpath leased the General Engineering Test Building back to Bendix Aerospace, the present occupants of the building.

65. EFFECTIVE TRANSITION FROM PANEL BOARD TO CRT-BASED PROCESS CONTROL. N.E. MAYFIELD and P.A. BAYNES. Westinghouse Hanford Co., Richland, WA (USA). Report No. WHC-SA--0459, Jan 1989, pp. 14.

Nuclear process operators have controlled process operations using Panel-Board Mounted Instrumentation at Hanford since 1943. The decision was made to upgrade the B Plant Defense Waste Processing facility to state-of-the-art CRT-based process control, commonly referred to as a Distributive Control System to improve operations safety, plant efficiency, and product quality. The safe and effective transition from PBMI to a DCS was the result of a cooperative effort between engineering, operations, and training, with full plant management support. Many elements of the upgrade had to be addressed: operator apprehension, vendor courses for the instructor

and engineers, training interface with engineering and procurement, closed-loop communications, and structured implementation. Plant management considers this project a model example of effective implementation of new technology. This was a proactive and cooperative approach on a major facility upgrade. All too often training is left out of the planning stages on such projects. This more often than not creates reactive and unproductive training responses because of a lack of technical expertise and program development time. Other nuclear processing facilities are or will be facing this challenge. This paper presents the successes encountered during the transition to CRT-based process control. The intent of this presentation is to better prepare other training organizations for this type of transition.

66. EXPERIENCE SO FAR IN THE MELTING OF LARGE AMOUNTS OF RADIOACTIVE SCRAP (in German). M. SAPPOK (Siempelkamp Giesserei G.m.b.H. und Co., Krefeld (Germany,F.R.)). Reuse of residual materials from repair and dismantling of nuclear facilities, Juelich, DE, November 10, 1987. Kernforschungsanlage Juelich G.m.b.H. (Germany, F.R.) ; Siempelkamp Giesserei G.m.b.H. und Co., Krefeld (Germany,F.R.). Report No. Juel-Conf--62, Jul 1988, pp. 207-245.

The author describes the development and application of a method for the melting of scrap consisting of removed components from decommissioned nuclear facilities. The melting process under partial vacuum and the three-stage filtering system are explained, together with the regulatory aspects such as licensing procedure, radiological protection, and documentation of radiological measurements. With the melting method and equipment, more than 1.000 tonnes of material have been recovered so far, which were used to produce shielding slabs or doors and containers for nuclear use. Examples of practical efficiency are presented.

67. RESULTS OF PREPARATORY PLANNING FOR ASSESSMENT OF WASTE FROM DECOMMISSIONING OF THE WAK (in German). G. ENGELHARDT, G. SCHNEIDER, and R. MARX (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projektbereich Anlagen; IFM, Linkenheim-Hochstetten (Germany, F.R.)). Reuse of residual materials from repair and dismantling of nuclear facilities, Juelich, DE, November 10, 1987. Kernforschungsanlage Juelich G.m.b.H. (Germany, F.R.). Report No. Juel-Conf--62, Jul 1988, pp. 92-125.

From the preparatory planning work for the decommissioning of the WAK, this report explains the assessments with regard to decontamination and dismantling of contaminated components, the decontamination of hot cells and other structural members in the areas with high-level radioactivity, as well as the relevant waste management, including disposal and probable recycling. Keeping the radiation exposure of the personnel as low as low possible, and minimizing the accruing waste are two major goals. The planning results assessing the radioactivity and materials mass inventory, the radiological categorization of the waste materials, and the relevant maximum permissible contamination levels are explained.

68. ANNUAL REPORT ON OPERATION, UTILIZATION AND TECHNICAL DEVELOPMENT OF RESEARCH REACTORS AND HOT LABORATORY, FROM APRIL 1, 1987 TO MARCH 31, 1988. Japan Atomic Energy Research Inst., Tokyo. Report No. JAERI-M--88-211, Oct 1988, pp. 178 (in Japanese).

Activities of the Department of Research Reactor Operation in fiscal year 1987 are described. The department is responsible for operation and maintenance of JRR-2, JRR-4, Research Reactor Development Division which performed upgraded JRR-3 and other R D, and Hot Laboratory. In the above connection various other work has also been performed, such as technical management of fuel and coolant, radiation control, irradiation technique, etc. In

Hot Laboratory, we have performed post-irradiation examinations of fuels and materials, and also development of examination procedures, too.

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