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Occupational Dose Reduction at Nuclear Power Plants: Annotated Bibliography of Selected Readings in Radiation Protection and ALARA

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ABSTRACT

The ALARA Center at Brookhaven National Laboratory publishes a series of bibliographies of selected readings in radiation protection and ALARA in the continuing effort to collect and disseminate information on radiation dose reduction at nuclear power plants. This is volume 7 of the series. The abstracts in this bibliography were selected from proceedings of technical meetings and conferences, journals, research reports, and searches of the Energy Science and Technology database of the U.S. Department of Energy. The subject material of these abstracts relates to radiation protection and dose reduction, and ranges from the use of robotics, to operational health physics, to water chemistry. Material on the design, planning, and management of nuclear power stations is included, as well as information on decommissioning and safe storage efforts.

Volume 7 contains 293 abstracts, an author index, and a subject index. The author index is specific for

this volume. The subject index is cumulative and lists all abstract numbers from volumes 1 to 7. The numbers in boldface indicate the abstracts in this volume; the numbers not in boldface represent abstracts in previous volumes.

In addition, all information in this and previous volumes of the series is available through our on-line information system called ACE (ALARA Center Exchange). ACE is accessible through fax machines or personal computers interfaced with modems. The bibliography database and other databases are kept current with new abstracts, information on research projects, and recent news of international events related to ALARA at nuclear power plants. Access to the system is provided at no cost to the ALARA community. For password certification, a manual, and other information about our system, please contact the ALARA Center, Building 703M, Brookhaven National Laboratory, Upton, NY 11973, or call (516) 282-3228.

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ABSTRACTS

1614. CLEAN-CUT REMOVAL SYSTEM FOR CONCRETE DECONTAMINATION. FUNAKAWA, N.; DEMURA, H.; KINOSHITA, T.; TANAKA, T. (Sato Kogyo Co. Ltd., Tokyo (Japan)). The 1st JSME/ASME Joint International Conference on Nuclear Engineering, Japan Society of Mechanical Engineers, Tokyo (Japan), 4-7 November 1991, CONF-911135, 1991, pp. 109-114.

This report addresses the Clean-Cut Removal System (CCRS), which can accurately scrape the contaminated concrete surface by the rotation of special cutters and simultaneously collect the scraped debris in a drum. The techniques up to this time for removing contaminated concrete surfaces have some problems, such as irregular depth of removal, difficulty of collecting the debris, and so on. As a solution to these problems, this new method, with sophisticated cutters and capable vacuum collecting systems, has been developed. The cutters used in this system can scrape the concrete surface to a few millimeters depth accurately in one pass. The debris is a uniform powder and is collected almost 100% by a vacuum-collecting system. This method has many advantages, such as radioactive waste reduction, prevention of internal exposure of workers, recontamination prevention, and easy measurement of residual radioactivity after decontamination. In this report, it is shown that this system is very useful and effective for removing contaminated concrete in the nuclear facilities.

1615. DATA BASE ON DOSE REDUCTION RESEARCH PROJECTS FOR NUCLEAR POWER PLANTS. KHAN, T.A.; VULIN, D.S.; LIANG, H.; BAUM, J.W. (Brookhaven National Laboratory, Upton, NY). Nuclear Regulatory Commission, Washington, DC, Division of Regulatory Applications, NUREG/CR-4409, Volume 4, August 1992, (219 p).

This is the fourth volume in a series of reports that provide information on dose-reduction research and health physics technology for nuclear power plants. The information is taken from a data base maintained by Brookhaven National Laboratory's ALARA Center for the Nuclear Regulatory Commission. This report presents information on 118 new or updated projects, covering a wide range of activities. Projects including steam generator degradation, decontamination, robotics, improvement in reactor materials, and inspection techniques, among others, are described in the

research section of the report. The section on health physics technology includes some simple and very cost-effective projects to reduce radiation exposures. Included in this volume is a detailed description of how to access the BNL data bases which store this information. All project abstracts from this report, as well as many other useful documents, can be accessed, with permission, through the on-line system, ACE. A computer equipped with a modem, or a fax machine, is all that is required to connect to ACE. Many features of ACE, including software, hardware, and communications specifics, are explained in this report.

1616. MULTI-POINT DUST MONITORING SYSTEM WITH SSD. KOBAYASHI, H.; AOYAMA, K.; ISHIKURA, T. (Fuji Electric Co. Ltd., (Japan)). FAPIG (Japan), Volume 130, March 1992, in Japanese, pp. 27-33.

The radiation exposure administration in the nuclear facilities such as nuclear power stations is still more important recently. One of them is the environmental radiation monitoring system collecting radiation particles that is useful for the safety control during work. Fuji Electric Co. has developed the multi-point dust monitoring system with large area SSD. This system housing the digital equipments can simultaneously collect from multiple points and measure these samples, so this system can continuously measure without interruptions. And this system has many advantages: a good operation, many self-check functions, and easy data processing for example data storage, data analysis and so on. This system also provides the estimating function to enable earlier detection of unusual change of measured values.

1617. GERMAN ATOMIC ENERGY LAW IN THE INTERNATIONAL FRAMEWORK: PROCEEDINGS. Edited by PELZER, N. (Goettingen University (Germany)), Inst. fuer Voelkerrecht, INLA 3. Regional Meeting on the German Atomic Energy Law and its International Implications, 6-7 June 1991, Nomos Verlagsges, CONF-9106397, 1992, in German, (293 p).

The regional conference was devoted to the legal problems that ensue from German reunification against the background of the integration of German atomic energy law within international law. The elements of national atomic energy legislation

required by international law and recent developments in international nuclear liability law were discussed from different perspectives. The particular problems of the application of the German Atomic Energy Act in the five new Laender (the territories of the former GDR) were presented and discussed, namely: The continued validity of old licenses issued by the GDR; practical legal problems connected with the construction of nuclear power plants in the five new Laender; the legal issues connected with the final repository for radioactive wastes at Morsleben; and the new developments in radiation protection law following from the Unification Treaty and the new ICRP recommendations. All 14 lectures have been abstracted and indexed individually.

1618. STEAM GENERATOR CHANNEL HEAD DOSE RATES AT BABCOCK & WILCOX REACTORS. GRIGG, D.F. (B&W Nuclear Service Co., Lynchburg, VA). Electric Power Research Institute, Palo Alto, CA, EPRI-TR-100348, July 1992 (52 p).

This report describes the results of a program to collect historical OTSG dose rate data from the five Babcock & Wilcox operating plants. Data is presented for Arkansas Nuclear One, Crystal River, Davis-Besse, Oconee, and Three Mile Island. Also included are dose-reduction measures employed at each site.

1619. ORGANIZATION AND OPTIMIZATION OF NUCLEAR POWER PLANT INSPECTIONS: HIGH SERVE '90 - NUCLEAR ENGINEERING SERVICES. LANGETEPE, G. (Kernkraftwerk Philippsburg GMBH (Germany)), Edited by BAUER, K.G. (Deutsches Atomforum e.V., Bonn (Germany)), 23-24 October 1990, INFORUM Verl., CONF-9010517, 1991, in German, pp. 105-120.

Inspection of a nuclear power plant takes between four and six weeks. Highest demands are made on preliminary planning, execution, coordination, monitoring, quality assurance, and observance of schedules. In order to take into account both safety-related and economic aspects, at the Philippsburg Nuclear Power Plant, for instance, a complementary maintenance system is used with planned and unplanned work programs. The requirements of industrial safety and radiation protection are overriding.

1620. DECOMMISSIONING TECHNIQUES DEMONSTRATED WITH THE HELP OF THE GUNDREMMINGEN NUCLEAR POWER PLANT, UNIT A. HIGH SERVE '90 - NUCLEAR

ENGINEERING SERVICES. EICKELPASCH, N. (Kernkraftwerke Gundremmingen Betriebsgesellschaft mbH (Germany)); STANG, W. (Versuchskraftwerk Kahl GmbH (Germany)); Edited by BAUER, K.G. (Deutsches Atomforum e.V., Bonn (Germany)), 23-24 October 1990, INFORUM Verl., CONF-9010517, 1991, in German, pp. 327-337.

For the KRB A a decommissioning concept was worked out which, in the first place, provided for the dismantling of no longer required systems in the turbine building. During those activities, as little as possible radioactive wastes were produced. After intensive decontamination, all the steel scrap of the turbine building was to be further processed in industry. In the meantime, that has been done. In a second decommissioning stage, the license for which was granted about one year ago, comparably higher contaminated systems of the reactor building and reprocessing facility will be dismantled and further treated. This work is in progress and will be finished presumably by the end of 1993. For the last decommissioning stage then it is intended to dismantle the activated parts of the reactor, that is the pressure vessel and its internals.

1621. DEVELOPMENT OF DECONTAMINATION MACHINE FOR CONTAMINATED WASTE WEAKLY FIXED WITH RADIOACTIVITY. KAWAHARA, M.; TAMANO, M.; MIWA, K.; TOGASAKI, M. (Shikawajima-Harima Heavy Industries Co. Ltd., Tokyo (Japan)); Morikawa, K.; SHIMIZU, Y., Ishikawajima-Harima GIHO (Japan), Volume 32, Number 2, March 1992, in Japanese, pp. 106-111.

Most of the radioactive waste resulted from the periodical inspection and maintenance works at every nuclear power plant has been crammed in the drums and stored in each plant because it is practically difficult to decontaminate it. Consequently, a number of drums stored in each plant are increasing year after year. On the other hand, the necessity of the decontamination of the nuclear power plant, including removing the radioactive contaminant fixed on some equipment and tools is increasing for the protection of the personnel and workers from the radiation exposure dosage so the improvement of plant cleanup is increasing also. To cope with these circumstances, the decontaminating machine has been manufactured for trial based on the result of the experimental study on the decontamination of actual radioactive wastes.

1622. OBJECTIVES FOR INTERNATIONAL CO-OPERATION IN THE FIELD OF NUCLEAR

SAFETY: A JOINT CHALLENGE. TOEPFER, K., International Atomic Energy Agency Bulletin (Austria), Volume 34, Number 2, 1992, pp. 2-5.

The article discusses specific strategies and measures to be taken for securing nuclear safety and radiological protection under the future development of the peaceful use of nuclear energy.

1623. FUNDAMENTAL PRINCIPLES OF PROTECTION AND SAFETY FOR NUCLEAR POWER. GONZALEZ, A.J. (International Atomic Energy Agency, Div. of Nuclear Safety, Vienna (Austria)). International Atomic Energy Agency Bulletin (Austria), Volume 34, Number 2, 1992, pp. 10-14.

The purpose of this paper is to present a draft of a coherent set of fundamental principles of protection and safety for nuclear power, which might be considered as a basis for developing formal principles for an eventual de jure adoption by governments. The objective is to facilitate agreements for a global harmonization on criteria to offset perceived threats from nuclear power to health and the environment.

1624. ASSESSMENT OF SHUTDOWN MANAGEMENT. MARION, A. (Technical Div., Nuclear Management and Resource Council, Washington, DC). Proceedings of the American Power Conference, Chicago, IL, 13-15 April 1992, CONF-920432, 1992, pp. 929-933.

Over the past several years, there have been a number of events that have occurred during nuclear plant outages. These events included losses of AC power, losses of decay heat removal capability, reductions in shutdown margin, and losses of reactor coolant system inventory. Individually, these events have not posed nor indicated an undue risk to public health and safety. Collectively however, they contributed to a perception that outage activities are not being controlled effectively. This paper reports that for many of these same reasons, events that occur during outages have also been of concern to the industry. These events can have a significant economic impact on a company in addition to their being disruptive to the conduct of an efficient outage. And while we have expended industry resources reviewing these events, we have not been fully effective at addressing the root cause of the problem.

1625. THE CANADIAN APPROACH TO NUCLEAR SAFETY: THE SAFETY OF ONTARIO'S NUCLEAR POWER REACTORS. A SCIENTIFIC AND TECHNICAL REVIEW.

VOLUME 2: APPENDICES. ROBERTSON, J.A.L.; HURST, D.G., HARE, F.K. (Ontario Nuclear Safety Review, Toronto, Ontario (Canada)), INIS-mf-13341, February 1988, pp. 4.1-4.7.

The exemplary safety record for the generation of nuclear electricity in Canada has been attributed to both the inherent design features of the CANDU reactor and the professional competence and safety consciousness of the utilities that operate them. An important factor that is often overlooked is the particular approach to reactor safety adopted in Canada and the fact that it was essentially in place before the first CANDU reactor was built. The Atomic Energy Control Act, enacted in 1946, declared atomic energy to be a federal responsibility and established the Atomic Energy Control Board (AECB) as the agency responsible for controlling atomic energy in Canada. The public has access to all relevant information on the licensing and regulation of nuclear power, and is consulted during the development of new policies. Responsibility for the safe operation of nuclear facilities rests on the utilities that operate them; the regulatory agency sets safety objectives and performance requirements. The primary safety objective is that the nuclear generation of electricity should be at least as safe as comparable industries and activities. From this stem other objectives: the prevention of pollution and other harm to the environment, and the protection of the interests of future generations. In assessing the risk from radiation the AECB closely follows recommendations developed by the International Commission on Radiological Protection. No exposure to radiation is justified unless there is a compensating benefit, and exposures should be as low as reasonably achievable. Safety objectives are achieved through general principles that govern the designers and operators of a nuclear power plant and are used by the regulatory body in assessing the safety (procedural means), and through the design of the plant and the way it is operated (technical and administrative means).

1626. AN EVALUATION OF OPERATIONAL SAFETY IN ONTARIO HYDRO NUCLEAR GENERATING FACILITIES: THE SAFETY OF ONTARIO'S NUCLEAR POWER REACTORS. A SCIENTIFIC AND TECHNICAL REVIEW. VOLUME 2: APPENDICES. KEOUGH, W.J., HARE, F.K., (Ontario Nuclear Safety Review, Toronto, ON (Canada)), INIS-mf-13341, February 1988, pp. 3.1.1-3.1.27.

The objective of this study was to evaluate the safety culture within Ontario Hydro's nuclear operations, particularly relating to activities that bear directly on the operation of nuclear facilities that can potentially contribute to radiation in the environment, and to assess the effectiveness of systems and procedures used to monitor operations and to respond to deviations in normal operating standards. In its statement of corporate direction Ontario Hydro places a high value on safety. In addition, the Nuclear Generation Division has established a philosophy of defence-in-depth, which permeates the operating staff. Improvements are suggested in communications, training, vigilance in maintaining radiation protection standards, and in planning, control and direction of maintenance work. Ontario Hydro's injury rate is much higher than that at the three best-performing Canadian chemical companies. It is recommended that new, lower targets be established for personal injuries, and that programs used in other industries be evaluated. The survey of practices and procedures did not identify any deficiencies that would compromise the safe operation of the nuclear generating facilities, although improvements are recommended in operating manuals and incident reporting.

1627. REDUCTION OF RADIATION EXPOSURE AND RADWASTE GENERATION BY IMPROVING WATER QUALITY CONTROL FOR BOILING WATER REACTOR. OTOHA, K.; AMANO, O. (Tokyo Electric Power Co., Inc., Kashiwazaki-shi (Japan)); MASUDA, T.; MORIKAWA, Y.; YOKOMI, M.; OHSUMI, K. Karyoku Genshiryoku Hatsuden, Thermal and Nuclear Power (Japan), Volume 42, Number 12, December 1991, in Japanese, pp. 1734-1747.

Reduction of both occupational radiation exposure and radwaste generation is one of the most important issues in stable and reliable operation of nuclear power plants. It is well known that the primary radiation sources responsible for the radiation exposures are such radioactive corrosion products as ^{60}Co , ^{58}Co and ^{54}Mn in such a case there is little fission products release due to fuel failures, as recent Japanese BWR (Boiling Water Reactor) plants experience has shown. For the reduction of radiation exposure, it is required to strengthen cleanup operation to remove corrosion products before they are activated by neutron irradiation. The reduction of radwaste generation is the other important requirement for BWR plant operations. This should be achieved without increasing radiation exposure. Measures to address both of these requirements were developed

and applied to recent Japanese BWRs, realizing the excellent results together with very low occupational radiation exposure and radwaste generation. Many of these measures were fitted back to the older plants also showing satisfactory results. The present paper shows a brief explanation of each measures developed. The results from the adoption of these measures are also shown.

1628. LICENSING-SPECIFIC ASPECTS OF THE DECOMMISSIONING PHASE OF NUCLEAR INSTALLATIONS. MAIN VOLUME. ANNEXES A-F. GOERTZ, R.; GRAF, R.; KISTINGER, S.; KNAUP, A.G., Brenk Systemplanung Ingenieurbuero fuer Wissenschaftlich-Technischen Umweltschutz, Aachen (Germany), BMU-1991-324/1/2/3, November 1990, in German, (828 p).

The project aim was to compile and evaluate, and indicate improvements to the available methods of evaluating concepts and techniques for the decommissioning of nuclear facilities, with a view to creating a database for establishment of future licensing requirements. Scenarios for the assessment of the radiation exposure of the personal and the radiological effects on the environment are set up. A study has been made into the impact of incidents or accidents in a facility on subsequent decommissioning work and conditions. The risk emanating from decommissioned facilities has been assessed in accordance with the Paris Convention on Third Party Liability. The assessment of radioactive wastes accrued by decommissioning and their conditioning is reported. The non-hazardous recycling of contaminated steel and iron scrap from decommissioned nuclear power plant, using ^{60}Co as the leading nuclide for calculation and modeling was investigated.

1629. TRUTH IN RADIATION: A MATTER OF ACCURATE MEASUREMENT. GILL, D.; SCHUTZ, D.F. (Teledyne Isotopes, Westwood, NJ). American Chemical Society (ACS) National Meeting, 26-31 August 1990, CONF-900802, 1990, p. 6, Paper CHAS 18.

Radiation protection begins with accurate measurement. Without respect for radiation based on a solid scientific measurement free reign is given to those who would create harm through over exposures resulting from carelessness and apathy as well as to those who would deny the benefits of nuclear technology by seeking to eliminate all exposure to nuclear radiation. Teledyne Isotopes (TI) has been a leader in accurate radiation measurement since 1955, providing

measurements of all significant radionuclides in environmental media from the vicinity of nuclear power plants, nuclear material production facilities and research facilities. TI also provides thermoluminescent dosimetry (TLD) for environmental radiation and human exposure monitoring. Instrumentation ranges from manual readers for small facilities and research applications to automatic readers for large-scale badge service operations.

1630. DOSE EXPANSION FROM USING RESPIRATORS. GREENWOOD, R.A.; O'DOU, T.J. (Toledo Edison Co., OH). Radiation Protection Management, Volume 9, Number 4, July 1992, pp. 22-29.

Internal radiation exposure control programs depend on the identification, control, containment, or elimination of radioactive material that could enter the body, and the use of protective measures to minimize introduction of the material into the body. This article relates specifically to the current liberal use of respiratory protection equipment in the nuclear power industry without regard for total dose. It also discusses operating costs of the respirator program, reasons for overuse of respirators, and the cost per person-rem saved. Additionally, dose to the wearer due to respirator use is estimated, and the net saving (excess) in dose is evaluated.

1631. DETERMINATION OF ABSORBED BETA DOSE IN MATERIALS. BROWN, D.D. (Chem-Nuclear Geotech, Inc.) Radiation Protection Management, Volume 9, Number 4, July 1992, pp. 63-68.

It is necessary during construction and subsequent operation of a commercial nuclear power reactor to evaluate the impact of the absorbed dose on the useful life of equipment and components. Routinely, the gamma contribution to this absorbed dose is evaluated, and major vendors provide information pertaining to equipment qualification for use in a radiation environment. When evaluating the absorbed beta dose to facility equipment, the source term is readily available during construction from the vendor's radiation design guide. This source term may be used to project potential post-loss of coolant accident (LOCA) equipment dose. After reactor startup, the actual source term may be determined by laboratory analysis of the affected system components. This article clarifies existing practices and calculational methods, and provides a straight-forward method for evaluating the absorbed beta dose in materials.

1632. THE DEVELOPMENT OF A DIRECT GAMMA-RAY MONITORING SYSTEM FOR NUCLEAR PLANTS. MURATA, Y. (Dept. Of Electrical and Electronic Engineering, Musashi Institute of Technology, Tamazutsumi, (Japan)); HORIUCHI, N. (Atomic Energy Research Laboratory, Musashi Institute of Technology, Kawasakishi-Asaoku (Japan)). IEEE Transactions on Nuclear Science (Institute of Electrical and Electronics Engineers), Volume 39, Number 3, June 1992, pp. 457-460.

This paper reports that an environmental monitoring system using a source modulation technique was experimentally developed for the measurement of direct gamma rays emitted from nuclear plants. It has a rotary collimator with which the gamma component coming in a detector is modulated as a sine-wave function of time. The intensity (dose rate) of direct gamma rays is determined from analyzing the cross correlation power spectrum density (CCPSD) of the detector output. The sensitivity and angular resolution of the system were examined by using a ^{137}Cs source: (1) The background component was reduced to less than 4% in comparison with the usual collimator, and then (2) the direct gamma of $0.3 \mu\text{R/h}$ ($0.79 \times 10^{-10} \text{ C/kg/h}$) could be detected with an excellent angular characteristic.

1633. IMPLEMENTING NEW RADIATION RULES IN FINLAND. VILKAMO, O.; PALTEMAA, R. (Finnish Centre for Radiation and Nuclear Safety Helsinki (Finland)). Nuclear Europe Worldscan (Switzerland), Volume 12, Numbers 5-6, 1992, pp. 80-81.

The article deals with the description of the radiation protection in the environment of Finnish nuclear power plants, improvements at the nuclear power plants and the radiation protection of the nuclear power plant staff.

1634. DEVELOPMENT OF AN ADVANCED ROBOT MANIPULATOR SYSTEM. OOMICHI, T.; HIGUCHI, M.; SHIMIZU, Y.; OHNISHI, K. (Mitsubishi Heavy Industries Ltd., Tokyo (Japan)). Transactions of the 11th International Conference on Structural Mechanics in Reactor Technology. Edited by SHIBATA, H. (Tokyo University, Institute of Industrial Science (Japan)), 18-23 August 1991, Atomic Energy Society of Japan, CONF-910817, 1991, pp. 343-349.

A sophisticated manipulator system for an advanced robot was developed under the "Advanced Robot Technology Development" Program promoted and

supported by the Agency of Industrial Science and Technology of MITI. The authors have participated in the development of a fingered manipulator with force and tactile sensors applicable to a master slave robot system. Our slave manipulator is equipped with four fingers. Though the finger needs many degrees of freedom so as to be suitable for skillful handling of an object, our fingers are designed to have minimum degree of freedom in order to reduce weight. Each fingertip was designed to be similar to a human finger which has flexibility, softness, and contact feeling. The shape of the master finger manipulator was so designed that the movement of the fingers is smoother and that the constraint feeling of the operator is smaller. We adopted a pneumatic pressure system for transmitting the tactile feeling of the slave fingers to the master fingers. A multiple sensory bilateral control system which gives an operator a feeling of force and tactility reduces his feeling of constraint in carrying out work with a robot system.

1635. PROPERTIES OF HIGH-DENSITY CONCRETE. MUKHERJEE, P.K. (Ontario Hydro Research Center, Toronto, Ontario (Canada)). Journal of Testing and Evaluation, Volume 20, Number 1, January 1992, pp. 78-86.

This paper reports that high-density concrete is used extensively in nuclear power plants for radiation shielding against biological hazards. Apart from the basic physical properties (i.e., compressive strength, density, and absorption), knowledge of thermal properties of such concrete is required to assess its performance under service conditions. This paper describes investigative work carried out on a high-density concrete. Also, because aggregates are the most important and critical component of the high-density concrete, properties of the aggregates and the criteria for their use in concrete have been discussed. In addition, cores were obtained from a shielding wall of a 20-year-old nuclear plant and selective tests were carried out on the core samples for comparison. The mix proportions of the concrete from the wall were different from those of the laboratory-made concrete. Thermal properties, such as conductivity, diffusivity, specific heat, emissivity and coefficient of thermal expansion, were determined, and the values were compared with a normal-density dolomite-limestone based concrete.

1636. GEORGIA TECH VIDEO-BASED MS PROGRAM IN HEALTH PHYSICS/RADIOLOGICAL ENGINEERING. ABDEL-KHALIK, S.I.; KAHN, B. (Georgia Institute

of Tech., Atlanta, GA). 1991 Winter Meeting of the American Nuclear Society (ANS) Session on Fundamentals of Fusion Reactor Thermal Hydraulics, 10-15 November 1991, Transactions of the American Nuclear Society, Volume 64, CONF-911107--, November 1991, pp. 50-51.

For the past several years, the health physics/radiation protection field has experienced a significant shortage of qualified professionals. The shortage is expected to continue for the foreseeable future given the continued demand by both nuclear and medical facilities and the expected growth in the areas of waste management and environmental remediation. In response to such a shortage, beginning in the fall of 1984, Georgia Institute of Technology (Georgia Tech) established a video-based instruction program that enables professionals in the nuclear field to earn a master of science degree in health physics/radiological engineering while working at a distant nuclear facility. The admission criteria and curricular requirements for the program are identical to those for the resident (on-campus) students (except that weekly attendance at departmental seminars is excused). The program is designed for students with undergraduate degrees in health physics, engineering, or appropriate sciences such as physics, chemistry, or biology. A total of 50 quarter credit hours is required, so that a student who takes one course per quarter can complete the program in four years.

1637. MASTER'S DEGREE IN NUCLEAR ENGINEERING BY VIDEOTAPED COURSES. CORRADINI, M.L.; VOGELSANG, W.F., 1991 Winter meeting of the American Nuclear Society (ANS) session on fundamentals of fusion reactor thermal hydraulics, 10-15 November 1991, Transactions of the American Nuclear Society, Volume 64, CONF-911107, November 1991, pp. 47-48.

In 1986, a group of northern midwest utilities met with faculty from the nuclear engineering department at the University of Wisconsin (UW) to discuss the possibility of offering graduate courses by videotape for academic credit and earning a master's degree. Four years later, two utility employees from Northern States Power (NSP) and Wisconsin Electric Power Companies (WEPCO) graduated from the University of Wisconsin with master's degrees earned entirely by taking videotape graduate courses at their individual nuclear power plant sites. Within these 4 years, more than a dozen videotaped graduate courses were developed by the faculty of the department in a formalized master's degree program in nuclear

engineering and engineering physics. This paper outlines the program's development and its current features.

1638. ON-SITE LOW-LEVEL RADWASTE STORAGE. BOUCHARD, V.F. (Bechtel, Oak Ridge, TN); RODDY, F.M. (Bechtel, Gaithersburg, MD). 1991 Winter Meeting of the American Nuclear Society (ANS) Session on Fundamentals of Fusion Reactor Thermal Hydraulics, 10-15 November 1991, Transactions of the American Nuclear Society, Volume 64, CONF-911107--, November 1991, p. 99.

With the impending closure of the radwaste burial sites on January 1, 1993, and the delay of state regional radwaste compacts, it has become necessary to look at other viable alternatives for radwaste storage. One of the most cost-effective methods, and the one having fewest difficulties with the regulatory agencies in local government, is to construct an on-site radwaste storage facility. Bechtel has produced an on-site radwaste storage facility design that can be adapted to serve any utility and/or nuclear station. The Bechtel radwaste storage facility will serve until the compacts are available and ready to receive and store radwaste. The present layout design has a storage capacity of 5 to 6 years and may be extended should the need arise.

1639. REVISED DOSE LIMITS FOR PERSONS OCCUPATIONALLY EXPOSED TO IONIZING RADIATION: PROBLEMS INVOLVED. RADIATION RISKS DUE TO RADON. THE NEW ICRP PUBLICATIONS. TRANSPLANTATION OF BONE MARROW. INCIDENCE OF LEUKAEMIA AMONG CHILDREN NEAR THE SELLAFIELD PLANT. PFOB, H. (Badenwerk AG, Karlsruhe, Abt. Arbeitsmedizin (Germany)); 32nd Annual Meeting of Vereinigung Deutscher Strahlenschutzärzte e.V., 30 May - 1 June 1991, Volume 33, Edited by REINERS, C. (Essen University, Gesamthochschule (Germany)), Klinik und Poliklinik fuer Nuklearmedizin); STREFFER, C. (Essen University (Gesamthochschule, (Germany)) Institute fuer Medizinische Strahlenbiologie); MESSERSCHMIDT, O. Published by G. Fischer, CONF-9105363--, 1992, in German, pp. 157-173.

The latest ICRP recommendations give the effective dose limit for occupational exposure to be 20 mSv per year on the average. In addition, the ICRP recommends annual dose limits of equivalent doses to the eye, lens, the skin, and the limbs. The dose limits applicable to occupationally exposed pregnant women are clearly lower. The paper discusses primarily the

problems involved in practical implementation of the new standards, e.g., for personal dose monitoring in the medical field, and in nuclear power plant operation.

1640. HUMAN FACTORING ADMINISTRATIVE PROCEDURES. GRIDER, D.A.; STURDIVANT, M.H., 1991 Winter Meeting of the American Nuclear Society (ANS) Session on Fundamentals of Fusion Reactor Thermal Hydraulics, 10-15 November 1991, Transactions of the American Nuclear Society, Volume 64, CONF-911107--, November 1991, pp. 190-191.

In non-nuclear business, administrative procedures bring to mind such mundane topics as filing correspondence and scheduling vacation time. In the nuclear industry, on the other hand, administrative procedures play a vital role in assuring the safe operation of a facility. For some time now, industry focus has been on improving technical procedures. Significant efforts are under way to produce technical procedure requires that a validated technical, regulatory, and administrative basis be developed and that the technical process be established for each procedure. Producing usable technical procedures requires that procedure presentation be engineered to the same human factors principles used in control room design. The vital safety role of administrative procedures requires that they be just as sound, just as rigorously formulated, and documented as technical procedures. Procedure programs at the Tennessee Valley Authority and at Boston Edison's Pilgrim Station demonstrate that human factors engineering techniques can be applied effectively to technical procedures. With a few modifications, those same techniques can be used to produce more effective administrative procedures. Efforts are under way at the U.S. Department of Energy Nuclear Weapons Complex and at some utilities (Boston Edison, for instance) to apply human factors engineering to administrative procedures: The techniques being adapted include the following.

1641. RADIATION SAFETY IN DECOMMISSIONING OF NUCLEAR POWER PLANTS. PIETRIK, I.; MELICHAR, Z. (Vyskumny Ustav Jadrovych Elektrarni, Trnava (Czechoslovakia)). Spravodajca - Vyskumny Ustav Jadrovych Elektrarni (Czechoslovakia), Volume 8, Number 4, 1991, in Slovak, pp. 1-4.

General principles and guidelines are described securing radiation safety of the personnel, population and environment during the decommissioning of nuclear power plants. This includes assessment of the

quantity of sources of "residual" radioactivity, assuring adequate monitoring of working rooms and individuals, identification of the development of exposure and of the most effective way of its control, and adaptation of conventional working procedures for the facility decommissioning.

1642. MEASURES TO BE TAKEN IN CASES OF RADIOACTIVE CONTAMINATION OF THE SKIN. RECOMMENDATION OF THE COMMISSION ON RADIOLOGICAL PROTECTION, APPROVED AT ITS 92ND MEETING ON SEPTEMBER 22, 1989. Bundesministerium fuer Umwelt, Naturschutz und Reaktorsicherheit, Bonn (Germany), Volume 15, Published by G. Fischer, 1991, pp. 77-90.

This recommendation refers to those technical, medical, and scientific facilities in which unsealed radioactive substances are handled. These facilities include, among other things: nuclear power plants, facilities of the fuel cycle, scientific and industrial laboratories, doctor's practices, hospitals and medical laboratories. Any unnecessary contamination of individuals shall be avoided. If contamination does occur, the resulting radiation exposure shall be kept as low as possible, even below the dose limits of the Radiological Protection Ordinance, taking into consideration that state of the art and also taking into account all circumstances pertaining to each individual case. Preventative measures to avoid or limit contamination represent the most effective form of protection. Preventative measures therefore include regular checks on the condition of the skin and constant personal skin care. Specific decontamination instructions must be provided for each individual plant and also, where applicable, for each individual work place. Contaminated items of clothing must be removed before decontamination is commenced. Low levels of contamination can usually be eliminated in one step by washing with water. There is no need to carry out further decontamination steps where the decontamination effect is less than 10% and the remaining surface-related activity is less than 10 Bq/cm² (averaged over 100 cm² where the contamination is largely distributed over the entire area).

1643. RADIOLOGICAL ASSESSMENT OF PLUTONIUM. RECOMMENDATION OF THE COMMISSION ON RADIOLOGICAL PROTECTION, APPROVED AT ITS 88TH MEETING ON DECEMBER 7-9, 1988. RECOMMENDATIONS OF THE COMMISSION ON RADIOLOGICAL PROTECTION 1988/1989. Bundesministerium fuer

Umwelt, Naturschutz und Reaktorsicherheit, Bonn (Germany), Volume 15, Published by G. Fischer, 1991, pp. 67-75.

The methods for the assessment of the radiation exposure to plutonium and of the radiation risk involved do not differ from those used with respect to other radionuclides. For occupationally exposed persons, a careful planning and implementation of monitoring measures can ensure that the dose limits are not exceeded in the case of plutonium. The long-term observations at nuclear facilities in the Federal Republic of Germany show that the low plutonium emissions lead to a radiation exposure of the population in the environment of nuclear power plants which is far lower than that caused by other radionuclides. The total dose as a result of all radionuclides from nuclear facilities is small as compared with the range of variation of natural radiation.

1644. THE USE OF PRACTICALS IN HEALTH PHYSICS TRAINING: OCCUPATIONAL RADIATION PROTECTION. PRICE, L.; DUTCH, R.S. (Nuclear Electric PLC. (United Kingdom)). British Nuclear Energy Society, London (United Kingdom), Conference on Occupational Radiation Protection, 29 April - 3 May 1991, CONF-910429--, 1991, pp. 165-167.

The authors are health physicists who, having worked at nuclear power stations, are now experienced tutors employed at Nuclear Electric's Nuclear Training Centre at Oldbury-on-Severn. The Centre provides training at professional and technician levels in all aspects of nuclear and associated technologies to operating and support staff throughout Nuclear Electric. For many years the authors have lectured on radiological protection to groups with widely differing backgrounds: engineers, scientists and industrial staff, including contractors. This paper will discuss the authors' experiences in using practicals and exercises to reinforce traditional lectures.

1645. TRAINING OF MAINTENANCE PERSONNEL IN THE CETIC: OCCUPATIONAL RADIATION PROTECTION. CENTRE FOR PWR NUCLEAR STEAM SUPPLY SYSTEMS' MAINTENANCE TECHNIQUE EXPERIMENTATION AND VALIDATION. RICHEUX, G. (CETIC (France)). British Nuclear Energy Society, London (United Kingdom), Conference on Occupational Radiation Protection, 29 April - 3 May 1991, CONF-910429--, 1991, pp. 161-164.

The CETIC (Maintenance Preparation and Qualification Center for Pressurized Water Reactor Steam Supply Systems) was created in 1986 for validation and maintenance procedures, related toolings qualification and personnel training using such equipment in the Nuclear Power Plants. The CETIC utilization advantages are the improvement of the safety of the operation, the reduction of the collective dose of the maintenance team and the reduction in unit availability.

1646. ISOE: AN INTERNATIONAL CONTRIBUTION TO OCCUPATIONAL DOSE CONTROL IN A NUCLEAR POWER PLANT. OCCUPATIONAL RADIATION PROTECTION INFORMATION SYSTEM ON OCCUPATIONAL EXPOSURE. ILARI, O.; VIKTORSSON, C. (Organization for Economic Co-operation and Development (OECD), Paris (France)). British Nuclear Energy Society, London (United Kingdom), Conference on Occupational Radiation Protection, 29 April - 3 May 1991, CONF-910429--, 1991, pp. 39-44.

One of the conditions for achieving an efficient ALARA program in a nuclear facility is the availability of a broad data base on exposure data and dose reduction techniques. This would be facilitated by the existence of a mechanism which could aggregate and consolidate into a unified data base the experience available at individual plants and in different countries. In response to this need, the NEA promoted an international information system on occupational exposure (ISOE), whose features are briefly described in the paper. A pilot project to develop and test the system was completed in 1990 and negotiations are under underway to set up ISOE on a permanent basis.

1647. DOSI-ANA: A SOFTWARE PACKAGE FOR MAINTENANCE OPERATIONS DOSE MANAGEMENT AT PWRs. OCCUPATIONAL RADIATION PROTECTION. VESSIERES, G. (Électricité de France (France)), BERARD, S. (Framatome, Lyon (France)); LEFAURE, C. (Centre D'Etude sur L'Evaluation de la Protection dans le Domaine Nucleaire, Fontenay-aux-Roses (France)). British Nuclear Energy Society, London (United Kingdom), Conference on Occupational Radiation Protection, 29 April - 3 May 1991, CONF-910429--, 1991, pp. 29-32.

DOSI-ANA is a microcomputer software comprised of a set of easy-to-use program modules which can operate independently or in conjunction with each other, to plan, follow up and evaluate operations in nuclear power plants in an occupational dose control

perspective. This software, which is in use in French PWRs, has been developed by CEPN for ÉLECTRICITÉ DE FRANCE and FRAMATOME as a tool to support health physicists in their approach to reduce occupational exposures. The PREDICTION module allows the documentation of data bases integrating all relevant information related to the operation. It allows potential collective exposures to be calculated according to various criteria such as the technical phases of the operation and the categories of workers. The FOLLOW UP module integrates the fractions of individual dose and time associated with the various elementary tasks to be performed during the operation. The input data are collected manually or by means of an automatic dose and job time tracking system. The EVALUATION module allows complementary calculations to be performed as variations between different alternatives at the predictive level or between predicted and actual results and classical statistical analyses. The paper presents the methodological principles on which the software is based.

1648. OUTAGE RISK MANAGEMENT. KALRA, S.P., EPRI Journal (Electric Power Research Institute, Palo Alto, CA), Volume 17, Number 3, Apr-May 1992, pp. 34-36.

EPRI is developing risk assessment and management tools that will support utility efforts to ensure safety and maximize efficiency during refueling and other outages at nuclear plants. These tools include comprehensive guidelines and associated software to help utility personnel conduct safety assessments of outage plans, evaluate the risk impacts of proposed changes in outage schedules, monitor safety during a shutdown, and plan and implement contingency actions.

1649. FUEL HANDLING GRAPPLE FOR NUCLEAR REACTOR PLANTS. ROUSAR, D.L., Patent Number US 5104612 A, General Electric Co., San Jose, CA, April 1992.

This patent describes a fuel-handling system for nuclear reactor plants. It comprises: a reactor vessel having an openable top and removable cover and containing therein, submerged in water substantially filling the reactor vessel, a fuel core including a multiplicity of fuel bundles formed of groups of sealed tube elements enclosing fissionable fuel assembled into units, the fuel handling system consisting essentially of the combination of: a fuel bundle handling platform movable over the open top of the reactor vessel; a fuel bundle handling mast extendable downward from the platform with a lower end

projecting into the open top reactor vessel to the fuel core submerged in water; a grapple head mounted on the lower end of the mast provided with grapple means comprising complementary hooks which pivot inward toward each other to securely grasp a bail handle of a nuclear reactor fuel bundle and pivot backward away from each other to release a bail handle; the grapple means having a hollow cylindrical support shaft fixed within the grapple head with hollow cylindrical sleeves rotatably mounted and fixed in longitudinal axial position on the support shaft and each sleeve having complementary hooks secured thereto whereby each hook pivots with the rotation of the sleeve secured thereto; and the hollow cylindrical support shaft being provided with complementary orifices on opposite sides of its hollow cylindrical and intermediate to the sleeves mounted thereon whereby the orifices on both sides of the hollow cylindrical support shaft are vertically aligned (prob)Ming a direct in-line optical viewing path downward there-through and a remote operator positioned above the grapple means can observe from overhead the area immediately below the grapple hooks.

1650. RADIOLOGICAL CONSEQUENCES OF PROPOSED LANDFILLING OF LOW-LEVEL RADIOACTIVE WASTE. DRAKE, P. (Ringhalsverket, Vaerobacka (Sweden)). Environmental Consequences of Hazardous Waste Disposal, Volume 2. Edited by JOHANSSON, G. (Swedish Radiation Protection Board, Stockholm (Sweden)). Joint International Symposium on the Environmental Consequences of Different Waste Disposal, 27-31 May 1991, Volume 57, Published by Allmaenna Foerlaget, CONF-910508--, 1991, pp. 183-192.

A proposal for landfilling of low-level radioactive waste (100 GBq in 10,000 m³ at Ringhals Nuclear Power Plants) was sent to the Swedish Radiation Protection Institute (SRPI) in 1989. In 1990, slight changes were made to the proposal to overcome the conventional risks of landfilling. In the proposed method, most of the low-level wastes is compacted and put into sealed plastic packages. The waste is then placed on a sand-moraine bed above a level-blasted rock surface. The area above and between the waste packages is filled with a sand-seashell mixture for pH adjustment of any water infiltrating into the site. The whole deposit is covered with a least 50 cm of moraine in order to attenuate most of the radiation from the waste and to prevent rainwater from reaching the waste. Downstream from the deposit, there is a retention bed made of seashells

and seaweed. Any small quantities of water from the deposit passing through the retention bed, as well as rainwater, will continue out to the sea. Use of this method will most probably not lead to an increase in the radiation dose to people outside the site. In the rather improbable case of intrusion into the deposit after 50 years, a maximum dose of 10 µSv per year would be received. If all the radioactivity were to be transported to the sea in one year, as a result of the breakdown of all the barrier, an individual eating fish caught in the vicinity of the site would receive less than 25 µSv during the following year. In the worst case, if the waste ignites prior to, or during land-filling, individuals living at a distance of 1 km from the fire could receive a dose of less than 20 µSv.

1651. LOW-LEVEL RADIOACTIVE WASTE MANAGEMENT IN EDF NUCLEAR POWER PLANTS. LOW-LEVEL RADIOACTIVE WASTE MANAGEMENT. BOUSSARD, C. (Électricité de France (EDF), Paris (France)). Societe Francaise de Radioprotection, Fontenay-aux-Roses (France). Meeting of the Societe Francaise on Low-Level Radioactive Waste Management, 5 March 1991, CONF-9103249--, 1991, in French, pp. 20-24.

This paper shows some recent examples of low-level radioactive waste management in EDF nuclear power plants: - Radioactive liquid wastes proceeding from steam generators leaching (NOGENT SUR SEINE-1 REACTOR) - Thermal insulation proceeding from heat exchanger and blower (CHINON-2 REACTOR) - Old iron from reactor dismantling (CHINON-3 REACTOR, MARCOULE G1 REACTOR, MARCOULE G2-G3 REACTORS) - fresh air filter and fire detector - CHINON-2 REACTOR breaker chambers.

1652. QUALITY ASSESSMENT WITHIN THE OSART PROGRAM. SVA FURTHER EDUCATION COURSE: QUALITY MANAGEMENT IN THE NUCLEAR POWER PLANT. MOORE, B.M. (International Atomic Energy Agency, Div. of Nuclear Safety, Vienna (Austria)). SVA-Vertiefungskurs: Qualitätsmanagement im Kernkraftwerk, 22-25 April 1992, Schweizerische Vereinigung fuer Atomenergie (SVA), CONF-9204161--, 1992, pp. 9.1-9.13.

For the past ten years, the International Atomic Energy Agency (IAEA) has been conducting Operational Safety Reviews (OSARTs) at nuclear power plants around the world. OSARTs are three week in-depth reviews of the operational safety practices at nuclear power plants. The OSART program is a widely known IAEA activity and is a highly visible

part of the IAEA's safety program. As of April, 1992, 56 OSART missions to 50 nuclear power plants in 25 countries have taken place.

1653. METHODS FOR IMPROVING SAFETY OF NUCLEAR POWER ENGINEERING. ARTEMCHUK, V.V.; PLOKHII, T.G., Bezopasnost Truda v Promyshlennosti (USSR), Volume 5, May 1991, in Russian, pp. 9-11.

Disadvantages of the current approaches to the safety of nuclear power plants (NPP) are discussed. It is shown that NPP safety level is an indeterminate parameter due to (1) uncertainty about the technical conditions of equipment; (2) excess and contradictoriness of safety manuals and regulations; (3) uncertainty about the readiness of the personnel to fulfill his functions and (4) lack of legal differentiation of responsibility. A number of proposals for improving NPP safety level is made.

1654. Q.A. NECESSITY IN PERSONNEL DOSIMETRY. PERSONNEL DOSIMETRY STUDY SESSION: PROCEEDINGS. BENCO, A.; DE FRANCESCHI, L., CAVALLINI, A.; (Pisa University (Italy)) KLAMERT, V., (ENEA, Bologna (Italy)). Area Energia Ambiente e Salute, Giornate di Studio Sulla Dosimetria Personale, 28 May 1987, ENEA-RT-AMB-90-24, CONF-8705421--, 1991, in Italian, pp. 11-20.

The authors put forward a series of considerations in favor of the incorporation in international radiation protection guidelines (e.g., those of the ICRP, International Commission on Radiation Protection) of quality assurance provisions. The various aspects discussed include: difficulties in the choice of the most suitable dosimeters for specific working environments; difficulties in the interpretation of readings; accuracy in routine monitoring; requirements in Italian law relative to personnel dosimetry; the early history of personnel monitoring programs in Italy and in other countries; the adaptation of quality assurance principles employed in other sectors such as nuclear power plant components fabrication to personnel dosimetry; quality assurance requirements in the case of the provision of monitoring services by one agency and the interpretation of the results by an independent consultant; employer responsibility to comply with personnel monitoring standards; the problem of compatibility of monitoring data obtained from different types of environments (hospitals, research institutions, small industrial firms, etc.).

1655. RECOMMENDATIONS FOR REDUCING POSITIVE WHOLE BODY COUNTS. FARRELL, W.E. (E.I. Du Pont De Nemours and Co., Aiken, SC); GARNER, R.E.; HARDING, P.C. (Carolina Power and Light Co., Hartsville, SC). Radiation Protection Management, Volume 5, Number 4, August 1988, pp. 53-56.

This paper reports that the number of positive whole body counts (+WBCs) is often used as a performance indicator for evaluating contamination control and respiratory protection programs at nuclear power plants. For evaluation purposes, +WBCs are generally grouped into two categories -- those between 1% and 5% of a Maximum Permissible Organ Burden (MPOB), and those above 5% MPOB. The +WBCs in the first category are insignificant (in terms of dose commitment) and are commonly assumed to be caused mostly by external contamination or counting anomalies such as incorrect calibrations or variable background (i.e., some may be false positives). Position WBCs above 5% MPOB are more significant and are well within the reliable sensitivity of most whole body counting systems. Thus, +WBCs in the 1-5% MPOB range may be difficult to assign to specific causes, but those above 5% MPOB can be assumed to have been caused by internal contamination; that is, uptakes resulting from a contamination control or respiratory protection related incident.

1656. RADIOBIOLOGICAL INVESTIGATIONS IN NUCLEAR POWER-PLANT WORKERS. GENSICKE, F.; SOLLE, M.; LAUSSMANN, D.; FENDER, H.; BRUENING, I.; CONRADY, J.; KLATT, R.; ARNDT, D. (Bundesgesundheitsamt, Berlin (Germany)). Klinisch-Diagnostischer Bereich. Bundesgesundheitsblatt (Germany), Volume 34, Number 11, November 1991, in German, pp. 513-518.

Two hundred nineteen clinically healthy persons working in a nuclear power plant were examined under the aspect of radiation protection. One hundred ninety-nine of them were particularly exposed persons working in the controlled area. In addition to the hematological data, information on the following parameters was collected: frequency of chromosome aberrations and micronuclei in lymphocytes, intensity of ultraviolet fluorescence in peripheral lymphocytes and the ratio of the intensity of red/green fluorescence of leucocytes in fluorochromed by acridine orange. The results indicated an increase with increasing total accumulated dose especially in the case of dicentric chromosomes and the intensity of ultraviolet fluorescence. The total accumulated dose of the

occupational exposed persons obtained by film dosimetry was between 0 and 240 mSv. The findings were evaluated by multivariate statistical analysis and the results were discussed.

1657. NUCLEAR MEDICINE AND THE NUCLEAR POWER PLANT. BURING, M.R. (Pennsylvania Power and Light Co., Susquehanna Steam Electric Station, Berwick, PA); BRILL, D.R. (Geisinger Medical Center, Danville, PA). Radiation Protection Management, Volume 2, Number 3, April 1985, pp. 80-82.

There were over 9,000,000 diagnostic and therapeutic administrations of radiopharmaceuticals in the United States in 1981. In Pennsylvania alone, there were approximately 650,000 diagnostic and 77,000 therapeutic administrations in 1981. With a state population of 12,000,000, this means that 1 in every 16 persons underwent some type of nuclear medicine treatment in a single year's time. These figures show that the chances of a nuclear power plant worker undergoing nuclear medicine treatment are far from remote. In fact, the chance are increasing each year as the use of nuclear diagnostic and therapeutic techniques continues to rise. Nuclear medicine patients present special problems to power plant health physicists. Some of these are discussed in this paper.

1658. A PORTABLE BREATHING AIR CONTROL AND PURIFICATION UNIT. PICKETT, R.L. II (Numanco, Inc., Barrington, RI). Radiation Protection Management, Volume 1, Number 1, October 1983, pp. 55-64.

The use of an installed plant air system as a breathing air supply poses several problems, including the potential for contamination of the air, the chance for rapid loss of air without warning, inconvenient locations of outlet stations, and poor control over air quality, flow, and pressure. It is possible to solve these problems without requiring major modifications to the installed air system by examining the breathing air supply route as a whole and utilizing an add-on adaptor to convert the output of the plant air system into a reliable, pure, and convenient source of breathing air for respirator users. The breathing air control and purification unit described in this article functions as just such an adaptor by purifying plant service air, controlling the distribution of this purified air to up to four respirator users, monitoring for the presence of carbon monoxide, and providing respirator users, monitoring for the presence of carbon monoxide, and

providing alarms and a reserve air supply to protect against a loss of service air pressure.

1659. ESTIMATES OF AVERAGE BETA/GAMMA ENERGIES IN REACTOR SYSTEMS. HUDSON, C.G. (Carolina Power and Light Co., H.B. Robinson Nuclear Plant, Hartsville, SC). Radiation Protection Management, Volume 1, Number 1, October 1983, pp. 25-30.

This paper reports on reactor health physics programs which require some knowledge of the beta/gamma spectra that may be encountered by personnel so that dosimetry and survey instrumentation may be properly calibrated and used. Sophisticated spectral mappings can be performed, but they tend to be dose intensive and difficult to apply to actual working conditions. An alternative approach is to use the radiochemistry results from routine sampling of plant systems to estimate the average beta and gamma energies that may be present within plant components. Such an approach was tested at the H.B. Robinson Plant by taking samples from plant systems over a nine-month period, estimating the average beta and gamma energies present, and comparing the estimates to personnel dosimetry results from the same period. The average gamma energy was found to be similar to Cobalt-60 or Cesium-137, while the average beta energy was comparable to that of Thallium-204.

1660. RADIOACTIVE CONTAMINATION CONTROL CRITERIA AND MONITORING PRACTICES AT NUCLEAR POWER STATIONS. BUNKER, A.S. (Techrite Co., Marietta, GA). Radiation Protection Management, Volume 2, Number 2, January 1985, pp. 19-32.

In October 1984, Radiation Protection Management conducted a survey of contamination control programs at 27 nuclear power stations. This article presents the results of the survey in three main areas: the criteria chosen and implemented at the stations as limits and action levels for beta-gamma contamination on personnel, plant surfaces, and materials for unrestricted release, the contamination monitoring methods and types of instrumentation used, and relevant experience with occurrences of personnel contamination, including frequency, levels, and causes. In addition, three areas for improvement are discussed: a more realistic view of the actual detection capabilities of frisking, the use of appropriate units of measurement, and developing criteria that will permit the use of more sophisticated instrumentation.

1661. GUIDANCE FOR SPECIFYING PROTECTIVE CLOTHING, RESPIRATORY PROTECTION, AND RADIATION MONITORING MEASURES. LEWIS, L. (Duke Power Co., Charlotte, NC). Radiation Protection Management, Volume 2, Number 2, January 1985, pp. 59-62.

This paper reports that as the radiation protection programs at nuclear power stations become more complex, and as the number of health physics personnel required to run these programs increases, the need for standard methods for prescribing and implementing radiation protection measures also increases. The value of a standardized approach becomes readily apparent during outages when large numbers of new personnel unfamiliar with station practices must enter and work in controlled areas and radiation protection coverage must be maintained on jobs that span several work shifts and, therefore, several different health physics work teams. Without some type of guidance, the protective clothing, respiratory protection, and radiation monitoring measures specified on Radiation Work Permits and enforced for work in radiologically controlled areas may vary undesirable between jobs, work groups, and work shifts, leading to confusion on the part the workers and their supervisors.

1662. MEASUREMENT OF BETA DOSE AND DOSE RATE AT NUCLEAR POWER STATIONS. SORBER, R.E.; TWIGGS, J.A. (Duke Power Co., Charlotte, NC). Radiation Protection Management, Volume 5, Number 4, August 1988, pp. 41-47.

In this paper, a beta dose rate and dose measurement program based on the determination of plant-specific half-value layers (HVLs) for mixed beta-gamma fields is described. The program has been implemented at Duke Power Company's three nuclear power plants by characterizing the plant radiation fields using attenuation measurements, selecting appropriate reference sources, calibrating the reference sources, determining correction factors for portable survey instruments and personnel dosimeters, and making allowances for attenuation due to protective clothing and materials.

1663. NEW DATA ON THE SELF-ABSORPTION OF BETAS IN COBALT-60 HOT PARTICLES. LANTZ, M.W.; STEWARD, J.B. (Arizona Nuclear Power Project, Palo Verde Nuclear Generating Station, Phoenix, AZ). Radiation Protection Management, Volume 5, Number 4, August 1988, pp. 57-58.

The authors demonstrated that standard dose calculation methods for hot particles could seriously overestimate the beta dose rate component to skin. The reason-self-absorption within an activated satellite particle that has a finite thickness can lead to dramatic reductions in beta output, as compared to that predicted by calculation models that assume the particle has zero thickness. In this paper, the authors demonstrate the self-absorption effect with a particle model and confirmed it with measurements on two high-activity Co-60 particles found at the Palo Verde Nuclear Power Station. The authors then described a method for using an Eberline RO-2 ion chamber survey instrument to estimate the beta dose rate reduction related to self-absorption within a particle. This method relied on the comparison of the uncorrected beta/gamma ratio ((open window-closed window) ÷ closed window) for a particle expected of exhibiting self-absorption to the ratio obtained for a particle of zero thickness.

1664. ALARA: WORKING LEVEL TO MANAGEMENT -- AN UPDATE. MCARTHUR, W.C.; KNIAZEWCYCS, B.G.; ANDERSEN, R.L.; BIRD, D.Y.; PULEO, F.J. (KLM Engineering, Inc., Walnut Creek, CA). Radiation Protection Management, Volume 1, Number 2, January 1984, pp. 15-24.

This article is an updated version of a paper on the same subject that was presented at the 25th Annual Meeting of the Health Physics Society in 1980. Opportunities that can exist for exposure reduction at nuclear power plants during outages and normal operations are reviewed with special emphasis on the problems caused by compressing outage schedules. Various factors that tend to undermine efforts to maintain personnel radiation exposures as low as reasonably achievable are examined, and recommendations are presented on ways to overcome these factors. Two techniques for encouraging worker involvement in ALARA programs are introduced: an ALARA Training - ALARA Awareness Program and quality circles.

1665. CALCULATING GAMMA DOSE FACTORS FOR HOT PARTICLE EXPOSURES. MURPHY, P. (Palo Verde Nuclear Generating Station, Phoenix, AZ). Radiation Protection Management, Volume 7, Number 3, June 1990, pp. 38-42.

For hot particle exposures to the skin, the beta component of radiation delivers the majority of the dose. However, in order to fully demonstrate regulatory compliance, licenses must ordinarily provide reasonable bases for assuming that both the gamma compo-

ment of the skin dose and the whole body doses are negligible. While beta dose factors are commonly available in the literature, gamma dose factors are not. This paper describes in detail a method by which gamma skin dose factors may be calculated using the Specific Gamma-ray Constant, even if the particle is not located directly on the skin. Two common hot particle exposure geometries are considered: first, a single square centimeter of skin lying at density thickness of 7 mg/cm² and then at 1000 mg/cm². A table provides example gamma dose factors for a number of isotopes encountered at power reactors.

1666. THE NEED FOR DOSIMETRY MULTIBADGING AT NUCLEAR POWER PLANTS. HUDSON, C.G. (Carolina Power and Light Co., H.B. Robinson Nuclear Plant, Hartsville, SC). Radiation Protection Management, Volume 1, Number 2, January 1984, pp. 43-50.

There are many situations that arise in the inspection and maintenance of nuclear power plants that call for workers to be exposed to high dose rates from anisotropic sources. In these situations, dosimetry that is placed at the normal chest or waist location may not measure the true whole body dose equivalents received by the workers, since other body locations may be exposed to dose equivalents that are significantly higher. These situations call for careful evaluation by health physics staffs to determine the need for multibadging, that is, for monitoring other body locations with additional dosimetry. This article outlines the basic factors that cause multibadging to be needed and examines the technical bases for selecting multibadging criteria. Multibadging results from recent steam generator work at the H. B. Robinson Plant are reviewed to demonstrate that multibadging is indeed necessary. Some practical criteria and recommendations are presented, including when to consider multibadging, how to implement multibadging, when to reduce the number of dosimeters used, and when to discontinue multibadging.

1667. INDUSTRY PERCEPTIONS ON QUALIFICATION CRITERIA FOR POWER REACTOR HEALTH PHYSICISTS. LACEY, L.R. (Quadrex Corp., Tulsa, OK). Radiation Protection Management, Volume 1, Number 1, October 1983, pp. 91-98.

This paper reports on a critical shortage of professional health physicists that has developed in the power reactor industry. One step that can be taken towards alleviating this shortage is to better define the qualification criteria for entry-level and senior power reactor health physics personnel and thereby establish

clearer goals for education and training programs. A survey was undertaken utilizing the Delphi technique to determine practicing power reactor health physicists' perceptions of the education, training, and experience requirements for entry-level positions and the position of Radiation Protection Manager. The survey yielded recommendations for criteria for general degree requirements, specific areas of technical education, and plant-specific orientation and training for entry-level power reactor health physicists and criteria for additional education requirements and experience necessary for the position of Radiation Protection Manager.

1668. THE PERSONNEL CONTAMINATION MONITOR. AN ALTERNATIVE TO HAND-PROBE FRISKING. GEIGER, E.; LITTLE, J.; SHEPARD, M. (Eberline Instrument Corp., Santa Fe, NM). Radiation Protection Management, Volume 1, Number 1, October 1983, pp. 102-105.

The primary method for detecting contamination on workers at a nuclear power plant as they exit from a contaminated area has been hand-probe frisking. The Health Physics Appraisal Program of the U.S. Nuclear Regulatory Commission found that personnel exiting from contaminated areas often monitored themselves too fast, too little, or not at all (NUREG-0855). Both the U.S. NRC and the Institute of Nuclear Power Operations have encouraged the industry to develop an alternative to hand-probe frisking. Eberline has responded by developing a family of instruments for personnel contamination monitoring. One of these instruments, recommended for replacement of hand-probe frisking, is described.

1669. STATIONARY SYSTEM FOR MONITORING LOCAL DOSE RATES WITHIN NUCLEAR POWER PLANTS. Kerntechnischer Ausschuss (KTA), Koeln (Germany), Published by Heymanns, KTA-1501, June 1991, in German, (8 p).

The rule contains requirements to be met by the system. It determines the measurement problem and stipulates requirements concerning measuring points, measuring layout, breakdown, and danger notification system, recording and registration of measuring values as well as documentation, special constructions of equipment for measurements during and after incidents, maintenance and repair, as well as inspections. The rule applies to the stationary system for monitoring local dose rates in controlled areas.

1670. ROBOTIC CLEANING OF A SPENT FUEL POOL. ROMAN, H.T.; MARIAN, F.A.

(PSE&G Research Corp., Newark, NJ); SILVERMAN, E.B.; BARKLEY, V.P. (ARD Corp., Columbia, MD). Radiation Protection Management, Volume 4, Number 3, May 1987, pp. 31-36.

Spent fuel pools at nuclear power plants are not cleaned routinely, other than by purifying the water that they contain. Yet, debris can collect on the bottom of a pool and should be removed prior to fuel transfer. At Public Service Electric and Gas Company's Hope Creek Nuclear Power Plant, a submersible mobile robot - ARD Corporation's SCAVENGER - was used to clean the bottom of the spent fuel pool prior to initial fuel loading. The robotic device was operated remotely (as opposed to autonomously) with a simple forward/reverse control, and it cleaned 70-80% of the pool bottom. This paper reports that a simple cost-benefit analysis shows that the robotic device would be less expensive, on a per mission basis, than other cleaning alternatives, especially if it were used for other similar cleaning operations throughout the plant.

1671. PROTECTIVE CLOTHING LAUNDERING AND MONITORING AT NUCLEAR POWER PLANTS. HYLKO, J.M. (Virginia University, Charlottesville, VA); MILLER, M.L. (Roy F. Weston, Inc., Albuquerque, NM); BREHM, L.E.; PETERSON, S.K. (Northern States Power Co., Monticello Nuclear Generating Plant, Monticello, MN). Radiation Protection Management, Volume 5, Number 1, February 1988, pp. 59-64.

This paper reports that a small but significant number of skin contamination incidents at Northern States Power Company's Monticello Nuclear Generating Plant were believed to have been caused by residual contamination in laundered, clean protective clothing. Since very little information was available on this mode of skin contamination, a two-part study was undertaken to evaluate it more fully. The first part of this study consisted of a survey of protective clothing laundering and monitoring practices at 24 nuclear power plants. The second part of the study was a simple experiment to evaluate the effect of perspiration on the transfer of residual contamination from laundered clothing.

1672. MODESTY GARMENT USE AT NUCLEAR POWER PLANTS. OWEN, D.E. (Encore Technical Resources, Inc.); JOHNSTONE, G. (Interstate Nuclear Services Corp.). Radiation Protection Management, Volume 7, Number 1, February 1990, pp. 23-31.

This article presents the results of a telephone survey of modesty garment use at U.S. nuclear power plants. Modesty garments are launderable or disposable lightweight garments worn in radiological areas under cloth protective clothing (PCs). The types of modesty garments used, the benefits they provide, and other issues related to their use are discussed.

1673. HANDLING APPARATUS FOR TREATING WATER CONTAMINATED WITH RADIOACTIVITY IN NUCLEAR POWER PLANTS. Kerntechnischer Ausschuss (KTA), Koeln (Germany), Published by Heymanns, KTA-3603, 1991, in German, (9 p).

The rule contains specifications to be met in the process engineering and engineering design, construction layout and testing, to avoid uncontrolled discharge of radioactive material and to keep the radioactivity discharged to the minimum possible (sections 46(1) No. 1 and No. 2 Radiation Protection Act (StrlSchV) and to ensure compliance with the maximum limits on the discharge of radioactive substances in effluent.

1674. ESTABLISHING AN ALARA PROGRAM. BRITZ, W.L.; CLANCY, J. (Public Service Electric and Gas Co.); ST. LAURENT, B.H. (Bartlett Nuclear, Inc., Plymouth, MA). Radiation Protection Management, Volume 2, Number 4, July 1985, pp. 33-40.

In 1983, Public Service Electric And Gas Company set out to establish a separate, formal ALARA program for its nuclear power stations, but as the ALARA program was being implemented, two successive forced outages occurred at the company's Salem Generating Station. These outages put ALARA plans to the acid test and helped shape the company's ALARA program. This article describes the pre-job planning process for the outages, the ALARA techniques employed, and the final ALARA review criteria that were adopted.

1675. TURNING AROUND THE HEALTH PHYSICS EMERGENCY PREPAREDNESS PROGRAM AT A NUCLEAR POWER PLANT. SCHOFIELD, S. (San Onofre Nuclear Generating Station, San Clemente, CA). Radiation Protection Management, Volume 6, Number 5, October 1989, pp. 67-71.

The specific actions taken to improve the health physics aspects of an Emergency Preparedness Program at a commercial nuclear power station are described in this article. Obstacles to the

implementation of the improvements are discussed, as well as how these obstacles were overcome.

1676. ACE: ALARA CENTER'S INFORMATION SERVICE. KHAN, T.A.; BAUM, J.W.; DIONNE, B.J. (Brookhaven National Laboratory, Upton, NY). Radiation Protection Management, Volume 8, Number 4, August 1991, pp. 24-34.

This article describes the various types of information on dose-reduction and ALARA that are available at the Brookhaven National Laboratory's ALARA Center. Information on plant collective doses and dose rates, and research on dose-reduction, developments, processes and practices effective in reducing exposure and innovative health physics technology projects at nuclear power plants are presented in various data bases. The article discusses how this information may be most successfully used in both day-to-day and longer-term planning of optimization activities. Further, it describes the on-line service, called ACE, which provides access to this information through telephone linkup, using a personal computer and modem. Examples of typical sessions with ACE are given. The advanced facilities provided by ACE and the ways in which ACE may be more fully utilized are also discussed.

1677. RADIATION PROTECTION PLANNING AND BUDGETING AT NUCLEAR POWER STATIONS. BUNKER, A.S. (Techrite Co., Kentwood, MI). Radiation Protection Management, Volume 4, Number 3, May 1987, pp. 44-47.

This paper reports that the radiation protection staffs at nuclear power stations operate principally as service organizations - that is, they react to events and to the needs of other organizations (maintenance and operations) to provide radiation protection job coverage and support services. Thus, the workload of a radiation protection organization is determined to a large extent by factors outside of its control, yet it must still plan its work and budget its resources if it is to provide satisfactory service to its customers. The current approaches to planning and budgeting for radiation protection programs at nuclear power stations are reviewed here, along with the average annual operating budgets for these programs from 1984 to 1986.

1678. DUKE POWER COMPANY'S INDEPENDENT RADIATION WORKER PROGRAM. LEWIS, L. (Duke Power Co., Charlotte, NC); YONGUE, C.T. (Duke Power Co., Oconee Nuclear Station,

Seneca, SC). Radiation Protection Management, Volume 4, Number 3, May 1987, pp. 53-56.

In recent years, very large numbers of Duke Power Company's construction personnel have been transferred to the nuclear station Construction-Maintenance Department as the company's plant construction program has been completed. This paper reports that the large numbers of people involved and the transition from new plant construction to work in radiological environments has taxed the stations' exposure control programs and increased the need for continued training and other management controls to maintain exposures as low as reasonably achievable (ALARA). As a part of the response to this need, the Independent Radiation Worker (IRW) Program was initiated in late 1985 in an effort to reduce doses to radiation workers by directly involving them in what has been termed low exposure job coverage.

1679. DOSE REDUCTION FOR SNUBBER INSPECTION AND TESTING. MORRISON, G.M.; COTTON, S.R. (Entergy Operations, Inc.). Radiation Protection Management, Volume 8, Number 5, October 1991, pp. 30-38.

This paper reports that Health physics staff members at Grand Gulf Nuclear Station have implemented several dose reduction methods for snubber inspection, testing and changeout. These methods include construction maps to permit easy location of snubbers in the drywell, painting azimuth numbers on the inside drywell wall and biological shield wall to coincide with the maps, requiring pre-job briefings for quality inspectors and craft support personnel, using job history files for work planning, using experienced inspectors and craft personnel whenever possible, designating certain craft personnel solely for snubber work, and cutting out stuck snubber pins rather than attempting intact removal. The total dose for snubber-related tasks has been significantly reduced using these methods.

1680. PREOPERATIONAL ALARA WALK-DOWN AT THE SHEARON HARRIS NUCLEAR POWER PLANT. GREENE, D.W. (Carolina Power and Light Co., Harris Energy and Environmental Center, New Hill, NC). Radiation Protection Management, Volume 4, Number 3, May 1987, pp. 25-30.

Carolina Power and Light Company's Shearon Harris Nuclear Power Plant will enter commercial operation in 1987. During its design and construction, number of reviews were performed to ensure that personnel exposures in the completed plant would be ALARA.

The final review in this process consisted of a comprehensive walk-down of plant radiation controlled areas by health physics, operations, maintenance, and engineering personnel. The walk-down project was conducted over a 12-month period using detailed criteria and checklists. This article describes the project, presents examples of the criteria that were used, and list the problems that were identified for resolution as well as the good ALARA design features that were incorporated into the plant.

1681. PERFORMANCE FACTORS USED IN RADIATION PROTECTION PROGRAMS AT NUCLEAR POWER STATIONS. BUNKER, A.S. (Techrite Co., Kentwood, MI). Radiation Protection Management, Volume 3, Number 1, October 1985, pp. 71-75.

This paper reports that performance monitoring is necessary to one extent or another in every radiation protection program. It can be as simple as a supervisor touring the facility and mentally checking off key items or as complex as a computerized progress charging system with hundreds of inputs and inch-thick monthly reports to management. But, while the formality and complexity of performance monitoring systems vary from one radiation protection program to another, one thing does not - the value of using accurate and reliable performance factors.

1682. ESTIMATING LOW ENERGY PHOTON FRACTIONS AT NUCLEAR POWER STATIONS. SORBER, R.E.; TWIGGS, J.A.; MEI, D.N. (Duke Power Co., Charlotte, NC). Radiation Protection Management, Volume 5, Number 3, June 1988, pp. 55-59.

This paper reports that in order to verify that low energy photons (<70 keV) comprise only a small part of the total photon spectrum at its nuclear power stations, Duke Power Company developed a practical method for estimating the low energy photon component of a radiation field using an air ionization chamber and a tin attenuator. This method, the theory behind it, and the components of the measurement system are described here. The results of surveys using this method at Duke Power's three nuclear stations indicate that low energy photons account for no more than 8% of the total gamma exposure in normally occupied areas.

1683. ACCOUNTING FOR THE SELF-ABSORPTION OF BETAS IN COBALT-60 HOT PARTICLES. LANTZ, M.W.; STEWARD, J.B. (Arizona Nuclear Power Project, Palo Verde Nuclear

Generating Station, Phoenix, AZ). Radiation Protection Management, Volume 5, Number 1, February 1988, pp. 25-29.

This paper reports that the assumptions used dose calculations can be overly conservative for discrete hot particles (activated satellite chips and irradiated fuel fragments) due to self-absorption of betas within the particles. Using data from tests with a Co-60 hot particle, a model is developed to estimate the dose reduction factor afforded by self-absorption in a satellite chip with a known thickness. The model can be applied indirectly using ion chamber survey instrument readings (the thickness of the particle does not have to be measured). Tests with Co-60 particles found at the Palo Verde Nuclear Generating Station verify that self-absorption is significant -- in one case, a dose reduction factor of 7 was measured in a satellite chip with a visible thickness.

1684. SKIN DOSE ASSESSMENTS USING PORTABLE INSTRUMENTS AT THE MCGUIRE NUCLEAR STATION. BLANTON, J.D. (Duke Power Co., McGuire Nuclear Station). Radiation Protection Management, Volume 6, Number 6, December 1989, pp. 58-68.

There are several acceptable methods available to calculate skin doses resulting from hot particle contaminations, the majority of which are not practical for personnel to perform at the work location. This paper describes a method in which a portable instrument is used to perform a skin dose assessment without the delays associated with other methods. This method has been used for more than a year at the McGuire Nuclear Station, and has provided results which compare favorably with the VARSKIN computer code.

1685. AN OPERATIONAL-BASED ALARA DESIGN PROGRAM. WAGNER, W.A.; STOCKNOFF, M.S. (Stone, and Webster Engineering Corp., Cherry Hill, NJ); PIKE, D.L.; WARD, K.D. (Niagara Mohawk Power Corp., Lycoming, NY). Radiation Protection Management, Volume 2, Number 2, January 1985, pp. 77-79.

A frequent criticism of the nuclear power plant design and construction process is that operational considerations for maintaining occupational radiation exposures as low as reasonably achievable are not addressed until it is too late to incorporate desirable modifications. Lessons that have been learned in the construction and operation of another plants and problems foreseen by the utility's radiation protection

and engineering personnel often simply do not get the attention they deserve during the design and field engineering stages. Stone and Webster Engineering Corporation and Niagara Mohawk Power Corporation have sought to avoid just such problems by jointly implementing a comprehensive, multidisciplinary ALARA design program for the Nine Mile Point Unit 2 nuclear power plant. This paper reports that this ALARA design program is organized to: directly incorporate NMPC's operational experience and philosophy, efficiently review the design and identify improvements from an occupational exposure viewpoint, and expedite design modifications while minimizing cost and schedule impacts.

1686. A GRADED ALARA PROGRAM. SARVER, S.P.; FOLZ, C.E. (Virginia Electric and Power Co., Surry, VA). Radiation Protection Management, Volume 2, Number 2, January 1985, pp. 63-70.

This paper reports on the formal programs that have been implemented at nuclear power stations to maintain personnel occupational radiation exposures as low as reasonably achievable which all have one thing in common - forms. The Surry Power Station is no exception; however, the forms and the program itself have been broken down into discrete parts that can be applied in a graded approach to ALARA. Each of these parts, and each associated form, it therefore more useful to the individuals who make ALARA decisions than a single, one-size-fits-all type of checklist. Surry's graded approach to ALARA begins where almost all ALARA programs do, with the Radiation Work permit. Using job specific information (location, scope of work, crew sizes, and man-hrs) provided by a planner or supervisor on an RWP Request Form and job-site survey data, a health physics technician estimates the man-rem that will be needed for each major task in a job. The decision is then made as to what levels, or grades, of the ALARA program to apply to the job.

1687. A FILTERED SR/Y-90 BETA CALIBRATION SOURCE FOR NUCLEAR POWER PLANTS. WILLIAMS, D. (Indiana and Michigan Electric Co., D.C. Cook Nuclear Plant, Bridgman, MI). Radiation Protection Management, Volume 5, Number 4, August 1988, pp. 48-52.

This paper reports that, because portable radiation detection instruments and personnel dosimeters typically exhibit the characteristic of energy dependence in beta radiation fields, their accuracy can be improved by calibration to a beta source with a spectrum that is consistent with the beta spectra found

in the work place. The study reported here evaluated the feasibility of filtering a Sr/Y-90 calibration source to produce beta spectra similar to those observed in a nuclear power plant. A methodology was developed to compare the beta spectra of contamination samples to that of a Sr/Y-90 source filtered with various density thicknesses of Mylar. A statistical goodness of fit measure was used to select the Sr/Y-90 source filtration (900 mg/cm²) that most closely approximated the beta spectrum of the samples.

1688. ACCEPTABLE EXPERIENCE AND TRAINING FOR HP TECHNICIANS AT NUCLEAR POWER PLANTS. BARLEY, W.H.; HIATT, J.W. (Bartlett Nuclear, Inc., Plymouth, MA). Radiation Protection Management, Volume 5, Number 5, October 1988, pp. 70-74.

The guidelines proposed in this paper provide a means for evaluating the experience and training of health physics technicians versus industry standards. Particular attention is directed towards the amount of experience that can be credited for work at noncommercial nuclear facilities and for jobs at commercial power facilities that are related to but are not the same as health physics technicians. These guidelines are intended to be used to assist with the evaluation of an individual's background for consideration as a senior technician, but they are only one means of doing so. In addition, the individual's technical knowledge should be evaluated through either oral or written testing, past work performance should be evaluated through previous observation or reference checks, and on-the-job performance should be confirmed by supervision. All of the above should be a part of the qualification process for health physics technicians assigned to responsible positions.

1689. RESPIRATORY PROTECTION: LESSONS LEARNED AT THREE MILE ISLAND. GEE, E.F. (GPU Nuclear Corp., Three Mile Island Nuclear Generating Station, Middletown, PA). Radiation Protection Management, Volume 5, Number 6, December 1988, pp. 59-63.

At the time of the accident at Three Mile Island (TMI) Unit 2 in March 1979, the station was ill-prepared for the respiratory protection demands that arose. Although a recognized respiratory protection program that permitted the application of protection factors was in place, it lacked the depth and detail needed to handle the immediate accident and subsequent recovery. The respiratory protection program at TMI, like those at other nuclear power plants, was designed mainly for planned maintenance under

controlled situations. While it is fairly easy to make provisions for immediate responses to accidents such as fires and radiative spills, it is much more difficult to plan for the respiratory protection aspects of large-scale or long-term radiological accidents. This paper reports that the lessons learned at TMI illustrate that there are practical measures that can be taken to prepare for an extended radiological accident.

1690. PERSONNEL CONTAMINATION MONITOR SETUP AT THE MCGUIRE NUCLEAR STATION. BLANTON, J.D. (Duke Power Co., McGuire Nuclear Station, Cornelius, NC). Radiation Protection Management, Volume 5, Number 5, October 1988, pp. 34-40.

This paper reports that when Eberline PCM-1 personnel contamination monitors were first used in lieu of handprobe frisking at the McGuire Nuclear Station, long count times delayed personnel, and large numbers of alarms could not be confirmed by frisking. An evaluation determined that the initial PCM-1 alarm setpoint of 5000 dpm/100 cm² Co-60 was equivalent to only 48 to 70 cpm on a pancake GM probe-less than half the station release limit of 150 cpm. Changing the PCM-1 setpoint to 5000 dpm/100 cm² Cs-137 reduced count times by 70% and improved the correlation between alarms and frisking. The key to this change in PCM-1 setup was the decision to base alarm setpoints on frisker-equivalent count rates.

1691. INES: THE INTERNATIONAL NUCLEAR EVENT SCALE. Ceskoslovenska Komise pro Atomovou Energii, Prague (Czechoslovakia), Volume 5, INIS-mf-13189, 1991, in Czech, (36 p).

The publication contains the INES scale and deals with the impact on the surroundings of a nuclear facility, with criteria of the event impact on the power plant, and with criteria of in-depth protection. Also included are definitions, an example of a set of initiating events for a PWR type reactor, examples of assessment, and an Appendix.

1692. FROM ALAP TO ALARA AT THE SAN ONOFRE NUCLEAR GENERATING STATION. SCHOFIELD, S. (Southern California Edison, San Clemente, CA). Radiation Protection Management, Volume 6, Number 4, August 1989, pp. 40-48.

The San Onofre Nuclear Generating Station has operated for approximately the last 10 years with a formal ALARA Program. The program's evolution was dictated by station, regulatory and other agency

requirements and recommendations. The basis behind San Onofre's program makes for an interesting chronology and is likely representative of the U.S. nuclear power plant industry's ALARA development. This paper reports that ALARA programs have played an important role in the reduction of radiation worker exposures. Recently, personnel collective exposures have begun to level off. It appears that traditional job site ALARA controls must be complemented with source term reduction ALARA techniques if plant exposures are expected to continue to decline.

1693. COMPUTERIZATION OF OFF-SITE DOSE CALCULATIONS AT TWO NUCLEAR POWER PLANTS. LEI, W.; ROBERTSON, C.E.; MOORE, G.T.; RAWLS, B.E.; SIPP, J.R. (Carolina Power and Light Co., New Hill, NC). Radiation Protection Management, Volume 5, Number 1, February 1988, pp. 54-58.

The Brunswick Nuclear Project (BNP) consists of two boiling water reactors designed to generate a total net output of 1642 MWe. Unit 2 achieved commercial production in 1975, and Unit 1 began commercial operation in 1977. The Harris nuclear Project (HNP) is an 860 MWe pressurized water reactor that entered commercial operation in May of 1987. Both plant sites are operated by Carolina Power and Light Company (CP&L). During January 1984, BNP replaced its older effluent technical specifications (part of the plant's original license) with the newer generation of Radiological Effluent Technical Specifications (RETS) mandated by the U.S. Nuclear Regulatory Commission. The RETS for HNP were integrated directly into this initial technical specifications. The initial version of the off-site dose calculation manual (ODCM) for BNP was drafted by a vendor and then extensively rewritten by the plant staff. The manual for HNP was drafted by CP&L corporate staff. At the outset, it was realized how impractical it would be to attempt to manually perform all of the dose calculations and keep the necessary records. The alternative-computerization-required extensive in-plant and corporate efforts to identify the computing resources (hardware) needed, create the software for ODCM implementation, and test, verify, validate, and document the software. This paper discusses these efforts.

1694. A HOT PARTICLE TRAINING PROGRAM FOR HEALTH PHYSICS TECHNICIANS. RUSSELL, M.J.; LEWIS, M.M.; RIGBY, W.F.; WARNOCK, R.V. (Southern California Edison Co., San Onofre Nuclear Generating Station, San

Clemente, CA). Radiation Protection Management, Volume 5, Number 1, February 1988, pp. 35-40.

The measures that are needed to detect and control hot particles (irradiated fuel fragments and activated stellite particles) are quite different from the normal routine at nuclear power plants, and as a result, special training is needed. This article outlines the development of the San Onofre Nuclear Generating Station's Hot Particle Training Program for Health Physics Technicians, including the job and task analysis, the training objectives, the training materials, and the implementation and evaluation of the training program. In this paper the management, attitudinal, and technical goals of the training program are presented along with examples of training objectives and excerpts from the student handbook.

1695. ALARA VALVE/COMPONENT LOCATING PROGRAM. COON, K. (Arizona Public Service Co., Palo Verde Nuclear Generating Station, Phoenix, AZ). Radiation Protection Management, Volume 1, Number 4, July 1984, pp. 86-88.

This paper reports that an often overlooked aspect in dose reduction programs at nuclear power plants is the savings that can be realized in exposures from routine tasks. The exposures are small per task, but by virtue of the number of times the tasks are performed, they result in large overall exposures. At the Palo Verde Nuclear Generating Station, one such task that is being addressed by the Radiation Protection ALARA Group is the locating of specific valves and components in Radiation and High Radiation Areas. A three part program has been developed to assist plant personnel in minimizing their exposure from this routine task.

1696. RADIOLOGICAL IMPACT OF HIGH ACTIVITY WASTES DISPOSAL IN A GRANITIC ROCK. PRELIMINARY ANALYSIS. OLIVEIRA, A.A.; PALACIOS, E. (Comision Nacional de Energia Atomica, Buenos Aires (Argentina)). Proceedings of the 15th Scientific Meeting and 4th Latin American Meeting and 1st Session on Nuclear Power Plants in San Carlos de Bariloche, Argentina, 2-6 November 1987 CNEA, INIS-mf-12758, CONF-8711355--, 1987, in Spanish, pp. 546-551.

This work analyzes, by a simplified model, the radiological impact due to radioactive wastes release when engineering and geologic barriers individually fail. Doses are calculated resulting from the individuals of a hypothetical critical group and the

potential collective doses for each one of the considered situations.

1697. AN ENVIRONMENTAL TLD FOR NUCLEAR POWER PLANTS. MARCINOWSKI, F. (Michigan University, School of Public Health, Ann Arbor, MI). Radiation Protection Management, Volume 5, Number 5, October 1988, pp. 63-69.

In this paper a dosimeter is proposed for the purpose of detecting noble gas releases (principally Xe-133) from nuclear power plants. The dosimeter consists of two $\text{Li}_2\text{B}_4\text{O}_7\text{:Cu}$ (LiBO) elements and one $\text{CaSO}_4\text{:Tm}$ (CaSO) element. One LiBO element is enclosed only in plastic and is used to determine shallow dose, while the LiBO and CaSO pair, which are used to determine deep dose, are enclosed in a 1070 mg/cm² aluminum sphere and plastic. The dosimeter is designed to overcome the three major problems involved in monitoring for Xe-133: the effect of angular response, differences in irradiation geometry (overhead plume vs. submersion), and interference from naturally occurring high-energy gamma rays. Laboratory test data are presented to demonstrate the proposed dosimeter's response characteristics.

1698. DISCRETE RADIOACTIVE PARTICLES AT NUCLEAR POWER PLANTS: PREVENTION, MITIGATION AND CONTROL. DIONNE, B.J.; BAUM, J.W. (Brookhaven National Laboratory, Upton, NY). Radiation Protection Management, Volume 8, Number 5, October 1991, pp. 65-79.

This article provides radiological safety professionals at nuclear power plants with information on the prevention, mitigation, and control of discrete radioactive particles (DRP). In 1990, a health physicist from the ALARA Center at Brookhaven National Laboratory searched the literature, communicated with nine health physicists involved with DRP protection, and visited Southern California Edison Company's San Onofre Nuclear Station, and Arizona Public Service Company's Palo Verde Generating Station to collect data and information on DRPs. A root cause analysis then was performed on how discrete radioactive particles escape from the reactor and coolant system and either irradiate the skin of workers or get inadvertently released off-site. This article organizes the root causes and corrective actions into four categories: discrete radioactive particle production; work environment; worker; and off-site environment. A bibliography of related papers and reports is also included.

1699. CLEAN-UP SUB-SYSTEM FOR CHEMICAL DECONTAMINATION OF NUCLEAR REACTOR PRIMARY SYSTEMS. CORPORA, G.J.; SCHLONSKI, J.S.; BAUER, F.I.; MILLER, P.E. (Westinghouse Electric Corp., Pittsburgh, PA) Patent Number US 5089217 A, 18 February 1992.

This patent describes a chemical decontamination clean-up system for use on-line in a nuclear reactor primary system. It comprises: a back-flushable filter; means within the nuclear reactor primary system for pumping primary system fluids from the nuclear reactor primary system downstream to the back-flushable filter and thereafter through the decontamination system; a plurality of demineralizer banks arranged in parallel, each demineralizer bank comprising one or more demineralizers arranged in parallel wherein primary system fluids are demineralized; means for selectively directing the pumped primary system fluids from the back-flushable filter to a particular demineralizer bank; and means for returning primary system fluids from the demineralizer banks to the primary system.

1700. REMARKS ON SOME PROBLEMS OF RADIATION PROTECTION CONCERNING LOW ACTIVE MATERIALS IN NUCLEAR POWER PLANTS WITH PRESSURIZED WATER REACTORS. ESCHNER, G. (Kraftwerks und Anlagenbau ag Berlin (Germany)). 21st International Symposium on Radiation Protection Physics, 3-7 April 1989, Kernenergie (Berlin, Germany), Volume 34, Number 7/8, CONF-8904255--, December 1991, pp. 335-339.

Problems are discussed using as example a nuclide mixture accumulated on low active ion exchange resin and solution obtained during its regeneration. To decide whether such solutions can safely be given to a receiving body of water and whether or not low active ion exchangers should be shielded under specific conditions, calculations are made to describe the decay of the nuclide mixture. The decline of the dose rate is calculated radially from a rotational-symmetric ion exchanger loaded with the resin. To assess the radiotoxic danger the nuclide mixture, annual limited intake values are calculated and shown in diagrams. Furthermore, the potential of danger is defined and also shown in a diagram. The influence of daughter products is studied. The radiotoxicity of the nuclide mixture mainly depends on the iodine nuclides.

1701. ADDITIONAL RADIATION DOSE TO A SITE AS A RESULT OF THE CHERNOBYL

REACTOR ACCIDENT. Neue Zeitschrift fuer Verwaltungsrecht (Germany), Volume 10, Number 12, 16 December 1991, in German, pp. 1185-1186.

The court decision discussed in this article states that, assessing a site's radiation level for compliance with the dose limits given in section 45 Radiation Protection Ordinance (StrlSchVO), take calculations have to include and take into account radioactive effluents from normal operation of other plant in that area, but not radiation doses emanating from accidents such as the Chernobyl reactor blow-up. (BVerwG, decision of May 23, 1991 - 7 C 34/90). The judgment of the Federal Court of Administration is the result of an action started in October 1986 to prevent approval of the second partial license for the Brokdorf nuclear power plant (which was to be licensed for nuclear commissioning and test operation, specified normal operation and use of MOX fuel elements).

1702. ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED. (Nuclear Regulatory Commission, Washington, DC), NUREG-0940, Volume 11, Number 1, May 1992, (401 p).

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (January-March 1992) and includes copies of letters, Notices, and Orders sent by the Nuclear Regulatory Commission to licensees with respect to these enforcement actions. It is anticipated that the information in this publication will be widely disseminated to managers and employees engaged in activities licensed by the NRC, so that actions can be taken to improve safety by avoiding future violations similar to those described in this publication.

1703. ASSESSING PLANT SAFETY PERFORMANCE USING DIRECT AND INDIRECT INDICATORS. GROSSMAN, N.; MODARRES, M.; MOSLEH, A. (Maryland University, College Park, MD). Transactions of the 11th International Conference on Structural Mechanics in Reactor Technology, Edited by SHIBATA, H. (Tokyo University, Institute of Industrial Science (Japan)), 18-23 August 1991, Atomic Energy Society of Japan, CONF-91081-7--, 1991, pp. 51-56.

This paper presents conceptually, details of a framework developed for evaluation and integration of nuclear power plant safety performance indicators. Both the direct (mainly hardware related) and indirect (primarily organizational) indicators can be modeled and evaluated using this method. The procedure of

evaluating and assessing a performance indicator using the framework is described.

1704. AN APPROACH TO SELECTION AND TRAINING OF TECHNICAL SUPPORT PERSONNEL. KELLY, J.J.; BOSTROM, C.T.; KNAPP, P.J. (Southern California Edison Co., San Onofre Nuclear Generating Station, San Clemente, CA). Radiation Protection Management, Volume 1, Number 3, April 1984, pp. 33-46.

One approach to counteracting the shrinkage in numbers of personnel who are qualified and willing to hire into technical support positions at nuclear power stations is to grow your own. The development and implementation of a comprehensive grow your own training program for health physics technicians at the San Onofre Nuclear Generating Station is described. Job task analysis is used as the basis for identifying both training and qualification program content. The training objectives and method are formalized in Training and Administrative Guides, Program Descriptions, and Qualification Manuals that are used by students and instructors to assure uniform program content and quality. The hiring program stresses the recruiting of local personnel to improve long-term retention and utilizes a validated screening process for selecting those individuals who are most likely to successfully complete the training process and be competent performers on the job. In this paper the exam validation procedure is described in detail.

1705. COMPUTER AIDED TRAINING SYSTEM DEVELOPMENT. MIDKIFF, G.N. (Advanced Technology Engineering Systems, Inc., Savannah, GA). Radiation Protection Management, Volume 4, Number 1, January 1987, pp. 54-60.

The first three phases of Training System Development (TSD) -- job and task analysis, curriculum design, and training material development -- are time consuming and labor intensive. The use of personal computers with a combination of commercial and custom-designed software resulted in a significant reduction in the man-hours required to complete these phases for a Health Physics Technician Training Program at a nuclear power station. This paper reports that each step in the training program project involved the use of personal computers: job survey data were compiled with a statistical package, task analysis was performed with custom software designed to interface with a commercial database management program. Job Performance Measures (tests) were generated by a custom program from data in the task analysis database, and training materials

were drafted, edited, and produced using commercial word processing software.

1706. OPERATION OF FINNISH NUCLEAR POWER PLANTS. TOSSAVAINEN, K. (Finnish Centre for Radiation and Nuclear Safety (STUK), Helsinki (Finland)), STUK-B-YTO-90, December 1991. (30 p).

The Finnish nuclear power plant units Loviisa 1 and 2 as well as TVO 1 and II were in operation for almost the whole second quarter of 1991. The load factor average was 87.4%. In consequence of a fire, which broke out in the switchgear building, connections to both external grids were lost and TVO II relied on power supplied by four back-up diesels for 7.5 hrs. The event is classified as Level 2 on the International Nuclear Event Scale. The process of examining the non-leaking fuel bundles removed from the Loviisa nuclear reactors has continued. The examinations have revealed, so far, that the uppermost spacing lattices of the bundles exhibit deformations similar to those detected in the leaking fuel bundles removed from the reactors. This event is classified as Level 1 on the International Nuclear Event Scale. Other events in this quarter which are classified according to the International Nuclear Event Scale are Level Zero (Below Scale) on the Scale. The Finnish Centre for Radiation and Nuclear Safety has assessed the safety of the Loviisa and Olkiluoto nuclear power plants based on the new regulations issued on 14.2.1991 by the Council of State. The safety regulations are much more stringent than those in force when the Loviisa and Olkiluoto nuclear power plants were built. The assessment indicated that the TVO nuclear power plant meets these safety regulations. The Loviisa nuclear power plant meets the requirements with the exception of certain requirements related to the ensuring of safety functions and provision for accidents. At the Loviisa nuclear power plant there are several projects under consideration to enhance safety.

1707. CHEMICAL AND RADIOLOGICAL RISK FACTORS ASSOCIATED WITH WASTE FROM ENERGY PRODUCTION. CHRISTENSEN, T. (Medical Section, National Institute of Radiation Hygiene, Oesteraas (Norway)); FUGLESTVEDT, J. (Div. for Environmental Studies, Center for Development and Environment, University of Oslo (Norway)); BENESTAD, C. (Center for Industrial Research, Oslo (Norway)); EHDWALL, H. (National Institute of Radiation Protection, Stockholm (Sweden)). Science

of the Total Environment (Netherlands), Volume 114, Number 1, April 1992, pp. 87-97.

The toxic potential of waste from nuclear power plants and from power plants burning fossil fuels is estimated. The potential risks have been expressed as "risk potentials" or "person equivalents." These are purely theoretical units and represent only an attempt to quantify the potential impact of different sources and substances on human health. Existing concentration limits for effects on human health are used. The philosophy behind establishing limits for several carcinogenic chemicals is based on a linear dose-effect curve. That is, no lower concentration of no effect exists and one has to accept a certain small risk by accepting the concentration limit. This is in line with the establishment of limits for radiation. Waste products from coal combustion have the highest potential risk among the fossil fuel alternatives. The highest risk is caused by metals, and the fly ash represents the effluent stream giving the largest contribution to the potential risk. The waste from nuclear power production has a lower potential risk than coal if today's limit values are used. If one adjusts the limits for radiation dose and the concentration limit values so that a similar risk is accepted by the limits, nuclear waste seems to have a much higher potential risk than waste from fossil fuel. The possibility that such risk estimates may be used as arguments for safe storage of the different types of waste is discussed. In order to obtain the actual risk from the potential risk, the dispersion of the waste in the environment and its uptake and effects in man have to be taken into account.

1708. ACTIVATION STUDIES OF CONCRETE BINDING AGENT INGREDIENTS USED FOR NUCLEAR RADIATION SHIELDING. FRONTASEVA, M.V.; GUNDORINA, S.F.; NAZAROV, V.M. (Joint Institute for Nuclear Research, Dubna (USSR)); BABOSHIN, N.G.; ENGOVATOV, I.A.; LACDANSKIJ, P.A.; STEFANOV, N.I. (Moscow Civil Engineering Institute (USSR)). 22nd International Symposium on Radiation Protection Physics, 2-6 April 1990, Kernenergie (Berlin, Germany), Volume 34, Numbers 7/8, CONF-9004289--, December 1991, pp. 306-310.

The results of activation studies of concrete binding agent ingredients for shielding structures of nuclear installations under the aspect of their decommissioning are given. It is shown that for the long-lived induced radioactivity of construction mineral materials irradiated for 30 years and cooled for more than one

year such radionuclides as calcium, iron, cobalt, cesium and europium are responsible. Elemental content of the binding agent and raw-material components for their production is obtained by neutron activation analysis. The results show that the type of the binding agent influences the concrete shielding activity to a great extent. The concentration of the above mentioned elements should be taken into account even at the stage of nuclear power plants design. It would allow to make a prognosis on the volume and radioactivity of wastes as well as on the radioactivity effect felt by the staff engaged in decommissioning.

1709. FACTS AND FIGURES ON NUCLEAR ENERGY. Deutsches Atomforum e.V., Bonn (Germany), Published by Inforum Verlag, September 1991, in German, (77 p).

The pocket-size brochure presents information on energy demand and supplies, on the current progress and schedules in nuclear power plant construction, and the relevant reactor types, on activities for radiation protection and reactor safety, and the management of wastes from nuclear power plant. Many diagrams, flowsheets, and tables support the information.

1710. NRC INDIVIDUAL PLANT EXAMINATION PROGRAM OVERVIEW - SUMMARY AND STATUS. FLACK, J.H. (Nuclear Regulatory Commission, Washington, DC). Transactions of the 19th Water Reactor Safety Information Meeting. Compiled by WEISS, A.J. (Brookhaven National Laboratory, Upton, NY). 19th Nuclear Regulatory Commission (NRC) Water Reactor Safety Information Meeting, 28-30 October 1991, NUREG/CP-0118, CONF-911079--, October 1991, p. 14.9.

On August 8, 1985, the U.S. Nuclear Regulatory Commission (NRC) issued a policy statement on severe accidents which concluded that existing plants posed no undue risk to public health and safety. The policy statement, however, recognized the importance of past probabilistic risk assessment (PRA) activities in understanding severe accidents at nuclear power plants, and in identifying previously unrecognized severe accident vulnerabilities. Based on this observation, the NRC issued Generic Letter 88-20 (and several supplements) which required licensees of commercial nuclear power plants to perform probabilistic safety assessments of their facilities. The assessments are to consider both internal initiated (termed Individual Plant Examination (IPE)), and external initiated (termed Individual Plant Examination for

External Events (IPEEE)) events. Over the next few years, seventy-eight IPE/IPEEE submittals covering 114 nuclear units will be reviewed by the NRC staff. The review will focus on the process used by the licensee in fulfillment of the IPE and IPEEE objectives prescribed in Generic Letter 88-20. This paper provides an overview of the IPE review process and summarizes program status. In addition, a discussion of preliminary IPE/IPEEE findings and proposed licensee actions that result from their IPE effort are presented. These findings will be stored in a computerized data system to be used at a later date to gain generic insight into plant behavior for different classes of plants. Generic IPE/IPEEE insights should prove useful to the NRC staff in evaluating potential strengths and weaknesses of the regulations.

1711. INTEGRATED LEVEL 3 RISK ASSESSMENT FOR THE LASALLE UNIT 2 NUCLEAR POWER PLANT. PAYNE, A.C. JR.; BROWN, T.D.; MILLER, L.A. (Sandia National Laboratory, Albuquerque, NM). Transactions of the 19th Water Reactor Safety Information Meeting. Compiled by WEISS, A.J. (Brookhaven National Laboratory, Upton, NY). 19th Nuclear Regulatory Commission (NRC) Water Reactor Safety Information Meeting, 28-30 October 1991, NUREG/CP-0118, CONF-911079--, October 1991, pp. 16.5-16.6.

An integrated Level 3 probabilistic risk assessment (PRA) was performed on the LaSalle County Station nuclear power plant using state-of-the-art PRA analysis techniques. The objective of this study was to provide an estimate of the risk to the off-site population during full power operation of the plant and to include a characterization of the uncertainties in the calculated risk values. Uncertainties were included in the accident frequency analysis, accident progression analysis, and the source term analysis. Only weather uncertainties were included in the consequence analysis. In this paper selected results from the accident frequency, accident progression, source term, consequence, and integrated risk analyses are discussed and the methods used to perform a fully integrated Level 3 PRA are examined. LaSalle County Station is a two-unit nuclear power plant located 55 miles southwest of Chicago, Illinois. Each unit utilizes a Mark 2 containment to house a General Electric 3323 MWt BWR-5 reactor. This PRA, which was performed on Unit 2, included internal as well as external events. External events that were propagated through the risk analysis included earthquakes, fires, and floods. The internal event accident scenarios included transients, transient-induced LOCAs (inad-

vertently stuck open relief valves), anticipated transients without scram, and loss of coolant accidents.

1712. NUCLEAR R&D. General Accounting Office, Washington, DC, Resources, Community and Economic Development Div., GAO/RCED-91-207, September 1991, (24 p).

Within the next 20 years, the licenses for 42 of 113 nuclear power plants that the U.S. Nuclear Regulatory Commission (NRC) has licensed to operate will expire. NRC and industry have been developing information and analyses that would be needed as a basis for renewing the operating licenses of these plants, and NRC is developing guidance for industry on the information required for license renewal. At NRC's request, the National Research Council of the National Academy of Sciences examined the future role of NRC's regulatory research, including research on the aging of nuclear power reactors and the possibility of extending their operating licenses for 20 years beyond the normal 40-year license term. The Council issued a report in 1986 with many recommendations directed broadly toward revitalizing nuclear safety research; only four of these recommendations were directed at research related to license extension. This paper provides information on: the actions NRC has taken to implement the Council's recommendations concerning the need for NRC to conduct research on reactor aging to support its license renewal efforts, the research on reactor aging that the U.S. Department of Energy (DOE) and industry have initiated and completed as a result of the Council's recommendations that research be performed to prove that license conditions set by NRC can be met, and whether the results have been provided to NRC, and NRC's plan to refine the estimates of risks (or the probability of accidents) created by extending the life of the present generation of reactors.

1713. COBALT REDUCTION IN NUCLEAR PLANTS. OCKEN, H. (Electric Power Research Institute, Palo Alto, CA), EPRI Journal, Volume 16, Number 6, September 1991, pp. 48-49.

Because cobalt-60 is the largest contributor to general radiation fields at operating nuclear plants, a key aspect of EPRI's radiation control research has been to develop technologies that will reduce a plant's inventory of elemental cobalt. Recent efforts have made progress in eliminating the cobalt-base alloys used in components that require outstanding wear resistance, and in reducing cobalt impurity levels in construction alloys.

1714. DOSE LIMITS FOR RADIATION PROTECTION. Umwelt- und Planungsrecht (Germany), Volume 11, Numbers 11/12, 1 December 1991, in German, pp. 440-441.

Section 45 of the Radiation Protection Ordinance aims at keeping the radiation dose due to specified normal operation so low that radiation doses do not add to the natural radiation exposure. Therefore, the minimization order is applicable already far below the dose value defined as the limit upon which preventive radiation protection measures have to be taken to protect from health hazards. The minimization order is not restricted to the plant for which an operating license is applied for. Radioactive effluents from specified normal operation of other installations are to be taken into account as well, however, not radioactive effluents resulting from incidents or accidents. BVerwG, decision of May 23, 1991 - 7 C 34.90 - (OVG Lueneburg).

1715. INCREASING ALARA AWARENESS AT PEACH BOTTOM ATOMIC POWER STATION. THE UNIT 3 PIPE REPLACEMENT PROJECT. KURY, R.F.; ZACHARY, M.S.; GIANGIULIO, A. Radiation Protection Management, Volume 7, Number 6, Nov-Dec 1991, pp. 37-50.

This article describes several ALARA measures and awareness programs which were implemented during the recirculation piping replacement project at the Philadelphia Electric Company's Peach Bottom Power Station Unit 3. Due to the potential for accumulating high individual and collective doses, an aggressive, project-specific radiological protection program was instituted to maintain doses as low as reasonably achievable (ALARA). Additionally, it was very important to ensure that the project was managed in such a manner as to keep total project exposure below the NRC-recommended guideline of 2000 person-rem for pipe replacement activities.

1716. ALARA PROJECT DATA BASE. KNIAZEWCZ, B.G. (KLM Technologies, Inc., Walnut Creek, CA). Radiation Protection Management, Volume 3, Number 4, July 1986, pp. 72-73.

Brookhaven National Laboratory has been asked by the Office of Nuclear Regulatory Research at the U.S. Nuclear Regulatory Commission to establish and maintain a data base of information that will describe, on a current basis, research in the nuclear power field that is likely to result in lower radiation dose to workers. This paper reports that this information is being sought by the NRC to satisfy its congressional

mandate and to ensure that adequate efforts are being made by the industry and others. Also, this information will be particularly useful to those doing related research and those considering new projects for dose reduction at nuclear power plants.

1717. EXPERIENCES AND PROBLEMS OF SKIN IRRADIATION AT WORKPLACES IN GERMANY. HEINZELMANN, M. (Forschungszentrum Juelich GmbH (Germany)). Workshop on Skin Dosimetry: Radiological Protection Aspects of Skin Irradiation, 13-15 May 1991, Radiation Protection Dosimetry (United Kingdom), Volume 39, Numbers 1-3, CONF-9105118--, 1991, pp. 173-181.

A survey is given of weakly penetrating radiation in German nuclear power plants and during fuel element fabrication. The dose equivalent rate from weakly penetrating radiation may be up to 100 times as great as that from strongly penetrating radiation. The radiation fields in nuclear power plants differ greatly. K-emitters have to be taken into consideration. The radiation is frequently well shielded by radiation protection clothing. For this reason, skin doses from weakly penetrating radiation are generally small. During fuel element fabrication, the dose equivalent rate is mainly caused by high-energy β radiation. This radiation is only slightly attenuated by radiation protection clothing. Therefore the β skin doses are often larger than the dose equivalent from strongly penetrating radiation.

1718. RECOMMENDATION ON DOSIMETRY OF WEAKLY PENETRATING RADIATION IN NUCLEAR POWER PLANTS. HEINZELMANN, M. (Forschungszentrum Juelich GmbH (Germany)); HALLFARTH, G.; WELTE, U. (Kernkraftwerk Krümmel GmbH (KKK), HAMBURG (Germany)); REGULLA, D. (GSF Forschungszentrum, Neuherberg (Germany)); SCHNEPEL, G.H. (Bundesamt fuer Strahlenschutz, Bonn (Germany)). Workshop on Skin Dosimetry: Radiological Protection Aspects of Skin Irradiation, 13-15 May 1991, Radiation Protection Dosimetry (United Kingdom), Volume 39, Numbers 1-3, CONF-9105118--, 1991, pp. 183-186.

Recommendations on the dosimetry of weakly penetrating radiation have been drawn up for nuclear power plants. These recommendations for external irradiation are based on measurements in German nuclear power plants. The recommendations do not merely refer to dosimetry itself, they also deal comprehensively with radiation protection for weakly penetrating radiation. They take into account

radiation protection planning, measurement of the radionuclide spectrum, attenuation of radiation by protective clothing, dose rate and dose measurement, requirements for dosimeters, interpretation of results and documentation.

1719. EXPERIENCES AND PROBLEMS OF SKIN IRRADIATION DUE TO HOT PARTICLES AT WORKPLACES IN THE UNITED STATES. REECE, W.D. (Texas A&M University, Dept. of Nuclear Engineering, College Station, TX). Workshop on Skin Dosimetry: Radiological Protection Aspects of Skin Irradiation, 13-15 May 1991, Radiation Protection Dosimetry (United Kingdom), Volume 39, Numbers 1-3, CONF-9105118--, 1991, pp. 165-171.

The dosimetry and biological mechanisms of non-stochastic effects arising from hot particle exposures are complex yet interesting. Experiences in the workplace and the interplay between technical experts, advisory groups, personnel at nuclear utilities, and regulators in the United States are perhaps just as interesting. A brief history of hot particle exposures in the U.S. is presented with the corresponding guidance and actions of the U.S. Nuclear Regulatory Commission (NRC). Mechanisms of producing hot particles are explored and the strategies used by the nuclear industry to control hot particle exposure are examined.

1720. INTEGRATING RADIOLOGICAL AND MAINTENANCE INFORMATION SYSTEMS. PERRELLI, L. (Tera Information Engineering Corp., Berkeley, CA). Radiation Protection Management, Volume 3, Number 2, January 1986, pp. 41-48.

The maintenance planning and radiological information management functions at nuclear power stations are often supported by totally separate, stand-alone computer systems. Yet these functions overlap in several areas and share similar information needs. This article examines some of the problems inherent in the use of multiple systems, presents arguments for integrating the two functions under the umbrella of a large information management system, and discusses means of dealing with concerns about integration.

1721. NEUTRON MEASUREMENT TECHNIQUES AT THE R.E. GINNA NUCLEAR POWER PLANT. MIS, F.J. (Rochester Gas and Electric Corp., NY). Radiation Protection Management, Volume 8, Number 6, Nov-Dec 1991, pp. 57-65.

This paper reports that at the Ginna Nuclear Power Plant, a 470 MWe pressurized water reactor, entries into the reactor containment while at 100% power are frequently made for maintenance. An algorithm has been developed for the Panasonic UD-809 TLD to break out the thermal dose component of the neutron energy spectrum on one element, while the remainder of the neutron dose is developed from two other elements in the dosimeter. Breaking out the thermal neutron dose component is important because, while it represents only 12% of the total Deep Dose Equivalent, it composes 50% of the neutron flux. This high thermal flux would cause a Panasonic UD-802, or equivalent albedo dosimeter, to over-respond by 40 to 50 percent of the true neutron Deep Dose Equivalent.

1722. MULTIBADGING REDUCTIONS AT THE H.B. ROBINSON NUCLEAR PLANT. FARRELL, W.E.; CHRISTMAS, K.D.; SHELLEY, M.H. (Carolina Power and Light Co., H.B. Robinson Nuclear Project, Hartsville, SC). Radiation Protection Management, Volume 4, Number 5, September 1987, pp. 31-36.

Because of the uncertainties involved in high dose work at nuclear power stations, multibadging practices are often quite conservative--the main criterion is when in doubt, multibadge. An analysis of multibadging results from the 1986 refueling outage at the H.B. Robinson Nuclear Plant indicated that multibadging could be reduced if consistent criteria were used. This article reviews the guidance available on multi-badging for non-uniform radiation fields and presents a set of refined multibadging criteria, which incorporate minimum dose rate and integrated dose equivalent thresholds. When used to analyze the multibadging results from the 1986 outage, the criteria indicate that reductions in multibadging of 50% may be possible, without compromising the accuracy of whole-body dose equivalent measurements.

1723. DESIGN OF A SPENT RESIN MONITORING AND SAMPLING SYSTEM. MCLAIN, M.E.; LEE, P.J.T. (Texas A&M University, College Station, TX). Radiation Protection Management, Volume 4, Number 6, November 1987, pp. 59-63.

This paper reports that the sampling of radioactive spent resins presents two challenges: determining when adequate mixing of recirculating resin has been achieved and minimizing personnel radiation exposure while drawing high-activity samples. A computer-controlled, spent resin monitoring and sampling system has been designed to meet these challenges. The system monitors the level and/or energy of the

radiation emitted from the resin recirculation line, determines when the resin mixture is homogenous, and activates resin aliquot collection by an automatic sampling valve. The design has been tested with a laboratory mock-up and is ready for installation and full-scale testing in a power plant.

1724. TECHNIQUES FOR REDUCING RADIATION FIELDS IN NUCLEAR POWER PLANTS. WOOD, C.J. (Electric Power Research Institute, Palo Alto, CA). Radiation Protection Management, Volume 4, Number 5, September 1987, pp. 37-42.

This review article summarizes current radiation field control techniques for light-water reactors, including: cobalt replacement, preconditioning, chemistry control, and dilute chemical decontamination. Successful implementation of these techniques at power reactors requires the coordination of several disciplines with strong support from the radiation protection manager.

1725. EVALUATION OF AIRBORNE ALPHA-EMITTER HAZARDS AT A NUCLEAR POWER PLANT. GOLDIN, E.M. (Southern California Edison Co., San Onofre Nuclear Generation Station, San Clemente, CA). Radiation Protection Management, Volume 4, Number 6, November 1987, pp. 64-66.

This paper reports that during the first fuel cycle at San Onofre Nuclear Generating Station Units 2 and 3, it became apparent that some fuel cladding defects existed at Unit 3 but not at Unit 2. This raised the concern that during the first refueling at Unit 3, concentrations of transuranics might be high, presenting the possibility of airborne alpha-emitter hazards. In an attempt to address this problem, a review of data from recent reactor coolant samples was performed. The data used were obtained from off-site analyses of reactor coolant samples that had been performed for waste stream characterization. Only long-lived radioisotopes were included as indicators. Two sets of data were used--one set from samples taken near the end of the first Unit 2 refueling and after start-up following a short outage at Unit 3 and a second set from samples taken while both units were at power.

1726. A BOTTOMLESS WET VAC. HOMYK, W.A.; ENGLISH, C. (Consolidated Edison Co. of New York, Indian Point 2 Station, Buchanan, NY). Radiation Protection Management, Volume 4, Number 5, September 1987, pp. 60.

This paper reports that HEPA-filtered wet vacuum systems are used at many nuclear stations for decontamination and spill cleanup. One problem with a wet vacuum, however, is that the operator must periodically stop the unit and move it to a floor drain to empty the water that has been collected. Interrupting the job in this manner decreases productivity, increases personnel exposures, and increases the chances for the spread of contamination. Recognizing this, Indian Point Station personnel have modified some wet vacuum systems to allow the continuous direct pump-down of the units to a floor drain.

1727. CALVERT CLIFFS REFUELING POOL DECONTAMINATION: AN UPDATE. WATSON, B.A.; MONTANA, D.L. (Baltimore Gas & Electric Co., Calvert Cliffs Nuclear Power Plant, Lusby, MD). Radiation Protection Management, Volume 3, Number 4, July 1986, pp. 45-51.

Baltimore Gas and Electric Company described its first experience with the use of Sigma Engineering's WEPA mechanical cleaning equipment for a refueling pool decontamination in the April 1984 issue of Radiation Protection Management. Since the initial refueling pool decontamination at Unit 1 of the Calvert Cliffs Nuclear Power Plant in 1983, three additional refuelings and subsequent refueling pool decontaminations have occurred. Additional operating experiences with the WEPA process are discussed in this article, including a revised job sequence that has resulted in a 66% reduction in critical path time. A new model RM-3 WEPA floor scrubber is described, and an inexpensive, American-made replacement for the RM-2 hand-held scrubber is introduced.

1728. RADIATION PROTECTION ORGANIZATIONS AT NUCLEAR POWER STATIONS. BUNKER, A.S. (Techrite Co., Kentwood, MI). Radiation Protection Management, Volume 3, Number 5, October 1986, pp. 19-29.

In the spring and summer of 1986, Radiation Protection Management conducted a survey of radiation protection organizations at 34 nuclear power stations. This article presents the results of the survey and examines radiation protection organizations in five areas: program scope, staffing levels, organization structure, corporate support, and the use of committees. Examples of three types of organization structures are presented and discussed - line/staff, departmental, and empire. The departmental type of structure is the dominant form of organization, reflecting a growing trend towards the separation of radiation

protection operations and support functions as well as specialization in technician duties.

1729. RADON DAUGHTER CONSIDERATIONS IN A NUCLEAR POWER PLANT. VANDERMEY, T.J., Radiation Protection Management, Volume 4, Number 4, July 1987, pp. 52-58.

A boiling water reactor in the start-up phase experienced a significant number of personnel contamination monitor alarms caused by radon daughter plateout on hard hats, clothing, and shoes. Alarm frequencies were compared to environmental conditions and ventilation system operations to determine the effects of various factors on radon plateout. High normal ventilation, radon daughter concentrations in the plant were found to be similar to outdoor concentrations, and alarm frequencies were inversely related to relative humidity. When ventilation systems were shutdown, indoor radon levels and personnel contamination monitor alarm rates increased significantly. In this paper some suggestions for accounting for radon daughter contamination in monitoring and training programs are presented.

1730. NUCLEAR SAFETY AND HEALTH. ENGLAND-JOSEPH, J.A., General Accounting Office, Washington, DC, GAO/T-RCED-91-9, March 1991, (12 p).

The full extent of nonconforming parts usage in the federal government is unknown. However, large and small companies, both foreign and domestic, have sold nonconforming parts-including counterfeit and substandard items-to nuclear power plants, commercial and military aircraft, naval ships, weapons systems, and the space shuttle. Accidents resulting from the failure of nonconforming parts could be devastating, GAO testified. To eliminate this problem, GAO believes that an aggressive, government wide approach is needed, one that would ensure that federal agencies cooperate and share information about nonconforming products. This paper reports that while a centralized information system may not stop the proliferation of nonconforming products, it should help federal agencies make informed decisions about potential suppliers and products. GAO concludes that the Office of Management and Budget is in the best position to develop an effective, appropriate, and cost-beneficial plan to help resolve the problem of nonconforming parts.

1731. SEGREGATION OF DRY ACTIVE WASTES USING AN EBERLINE SORTING TABLE MONITOR. LAGARDE, G.L. (Florida

Power and Light Co., Turkey Point Power Plant, Miami, FL); HESCH, J.K. (Eberline Instrument Corp., Santa Fe, NM). Radiation Protection Management, Volume 4, Number 5, September 1987, pp. 55-59.

This paper reports that nuclear power plants have used a variety of monitoring systems to segregate material that must be handled as low-level radioactive waste from material that can be released for uncontrolled disposal. Most of these systems use gas-flow proportional detectors, scintillation detectors, or a combination of both types of detectors in a shielded counting chamber (bag monitor) or mounted above and below a belt-type conveyor system. At the Health Physics Society Meeting in July 1986, Eberline Instrument Corporation and National Nuclear Corporation introduced different versions of a new and simpler type of waste sorting system--a table with large-area plastic scintillation detectors mounted beneath a sorting/monitoring surface. The Eberline system, designated as the Model STM-200 Sorting Table Monitor, has been evaluated by Florida Power and Light Company at its Turkey Point Plant for a period of about six months.

1732. DRY ACTIVE WASTE MANAGEMENT AT NUCLEAR POWER STATIONS. BUNKER, A.S. (Techrite Co., Marietta, GA). Radiation Protection Management, Volume 2, Number 3, April 1985, pp. 23-32.

In February 1985, Radiation Protection Management conducted a survey of dry active waste (DAW) management practices at 21 nuclear power stations. This article presents the results of that survey in four main areas: material and area controls, waste segregation and sorting, waste processing, and radiological monitoring. The results of the survey indicated that DAW management practices were related to the volume of waste produced in the sense that high-volume waste generators tended to implement more measures to try to reduce the waste volumes shipped for disposal. The survey also revealed a mismatch between unrestricted release criteria and the survey methods used for monitoring and releasing trash for unrestricted disposal.

1733. POSSIBILITIES OF DOSE RATE REDUCTION IN PWR AND BWR PLANTS. MARCHL, T. (Siemens AG, Erlangen (Germany)) ATW, Atomwirtschaft-Atomtechnik (Germany), Volume 37, Number 2, February 1992, in German, pp. 90-93.

Over the past ten years, improvements in the construction and operation of nuclear power plants

increasingly have been devoted to reductions of personnel doses. Results of these efforts can be demonstrated to have been achieved worldwide approximately from the mid-1980s on. In 1985, the average personnel dose in German PWR and BWR was around 3 man-Sv per plant. By 1990, it had been reduced to 1.7 man-Sv per PWR plant, and to 2.3 man-Sv per BWR plant. This trend was strongly influenced by the new plants, with improved plant concepts, commissioned in the PWR line after 1985 and in the BWR line after 1984, which were due in particular to the use of cobalt substitute materials instead of stellites, and to improved plant shielding designs.

1734. INTERNATIONAL COMPARISON OF RADIATION EXPOSURES OF LWR PLANT PERSONNEL. GEWEHR, K. (Gesellschaft fuer Reaktorsicherheit MbH, Garching (Germany)). ATW, Atomwirtschaft, Atomtechnik (Germany), Volume 37, Number 2, February 1992, German, pp. 85-89.

Reducing the radiation exposure of nuclear power plant personnel is an international goal. An overview of the status attained in a number of countries is obtained from a comparison of individual and collective doses. The situation of nuclear power plants is different in different countries, which has consequences with respect to the radiation exposure of personnel. Clear differences appear in the employment of in-house and external personnel, with the resultant consequences regarding collective and individual doses. It can be seen that internationally, the efforts made to reduce radiation doses have been very effective and that a general harmonization of standardized collective doses at a level between 0.3 and 0.5 mSv/GWh has been achieved. The new German pressured water reactors, with their very low radiation exposures, show a potential for even further dose reduction.

1735. NUCLEAR POWER. A BOOK FROM THE ENERGY AND CLIMATE SERIES. Deutscher Bundestag, Bonn (Germany). Enquete-Kommission "Vorsorge zum Schutz der Erdatmosphäre," Volume 5. Published by Economica Verlag, Bonn (Germany) 1990, in German, (1759 p).

These reports describe the current state of knowledge in the following sectors: state-of-the-art and possible development of nuclear reactors, technical and economic potentials of the contribution of nuclear power in Germany to energy supply and to the reduction of climate-changing trace gas emissions (carbon dioxide, Kr-85), environmental, safety, disposal and acceptance

aspects of nuclear power use, and perspective of nuclear fusion. The band-width of opinions and derived conclusions documents central controversies in the debate about nuclear power.

1736. BNL ALARA CENTER EXPERIENCE WITH AN INFORMATION EXCHANGE SYSTEM ON DOSE CONTROL AT NUCLEAR POWER PLANTS. BAUM, J.W.; KHAN, T.A., (Brookhaven National Laboratory, Upton, NY.) Nuclear Energy Agency (NEA) Workshop on Work Management in Occupational Dose Control, 4-6 February 1992, BNL-47168, CONF-920287--1, 1992, (15 p).

The essential elements of an international information exchange system on dose control at nuclear power plants are summarized. Information has been collected from literature abstracting services, by attending technical meetings, by circulating data collection forms, and through personal contacts. Data are assembled in various databases and periodically disseminated to several hundred interested participants through a variety of publications and at technical meetings. Immediate on-line access to the data is available to participants with modems, commercially available communications software, and a password that is provided by the Brookhaven National Laboratory (BNL) ALARA Center to authorized users of the system. Since January 1992, rapid access also has been provided to persons with fax machines. Some information is available for "polling" the BNL system at any time, and other data can be installed for polling on request. Most information disseminated to date has been through publications. However, new protocols, which have been simplified by the ALARA Center staff, and the convenience of fax machines, are likely to make the information available earlier.

1737. NUCLEAR POWER HARNESSING IN THE LIGHT OF ENVIRONMENTAL CONSIDERATIONS AND WITH REGARD TO SAFETY, RADIOACTIVE WASTE MANAGEMENT, AND ACCEPTANCE ASPECTS. PROJECT A.4.3.A. A BOOK FROM THE ENERGY AND CLIMATE SERIES. KUEPPERS, C.; NOCKENBERG, B.; SAILER, M.; SCHMIDT, G. (Oeko-Institut E.V., Freiburg im Breisgau (Germany) Oeko-Institut E.V., Darmstadt (Germany)). Deutscher Bundestag, Bonn (Germany). Enquete-Kommission "Vorsorge zum Schutz der Erdatmosphäre," Volume 5. Published by Economica Verlag, 1990, in German (1759 p).

The environment, safety, and risk aspects of the harnessing of nuclear power, especially with regard to

nuclear reactors, supply and disposal facilities, are dealt with. The paper takes stock of the current state of knowledge on the investigated problem. Integral, comparative safety analyses of comprehensive nuclear power systems, proliferation aspects, and acceptance aspects are included.

1738. AN INITIAL EXAMINATION OF THE EFFECTS OF OVERTIME AND SHIFT SCHEDULING ON NUCLEAR POWER PLANT SAFETY PERFORMANCE. BAKER, K.; OLSON, J. (Battelle Human Affairs Research Centers, Seattle, WA); MONTGOMERY, J. (Pacific Northwest Laboratory, Richland, WA). Eighteenth Water Reactor Safety Information Meeting. Proceedings: Volume 1. Compiled by WEISS, A.J. (Brookhaven National Laboratory, Upton, NY). 18th Water Reactor Safety Information Meeting, 22-24 October 1990, NUREG/CP-0114, Volume 1, CONF-9010185, Volume 1, April 1991, pp. 57-71.

The research reported here uses currently available data to begin to document overtime and shift scheduling practices in the nuclear industry and assess their safety consequences. Although these data provide an opportunity to begin to investigate the safety consequences of overtime and the 12-hour operator shift schedule, the research conducted can only be seen as a preliminary investigation due to the fact that not all factors that potentially affect the plant safety performance measures can be controlled. With this caveat in mind, the research findings suggest that greater amounts of overtime worked by operators and other technical staff are associated poorer safety performance. In addition, the 12-hour operator shift schedule was found to be significantly related to a higher incidence of operator-error licensee event reports.

1739. IN-SERVICE INSPECTION OF NUCLEAR POWER PLANTS. SAFETY SERIES. International Atomic Energy Agency, Vienna (Austria), IAEA Safety Series, Number 50-P-2, 1991, (181 p).

This manual is intended to provide more comprehensive considerations on the management, organization, preparation, improvement and implementation of in-service inspection activities and the related surveillance. It also gives illustrative examples of good practices and recommendations from operating and other organizations that are consistent with the requirements and recommendations of the Code and Safety Guides. The manual is directed primarily towards plant management. This manual should be used in conjunction with the Code and the Safety Guides, in particular with IAEA Safety Series Nos.

50-C-O, 50-SG-O2, 50-SG-05, 50-SG-07, 50-SG-08 and 50-SG-D1, which contain recommendations of a general character about maintenance activities and radiation protection in an operating power plant, and with the "Manual on the Maintenance of Systems and Components Important to Safety." This manual is divided into four technical sections. The first introduces the purpose, structure and main requirements of the program. The second section describes constituents of the program, recommending its scope, scheduling, acceptance standards and documentation of results. The following section goes into details of the inspection program's contents, such as the selection of components, inspection locations, defect types, applicable techniques and procedures, and the evaluation of results. The last section specifies recommended methods and techniques for inspection, such as visual, ultrasonic, eddy current, magnetic particle and others. This main part of the Manual is complemented by a number of annexes which reproduce actual national examples of established procedures, ISI program parts, acceptance standards, personnel training programs, testing techniques and other aspects of in-service inspection, illustrating practical implementation of the recommendations of the Manual.

1740. GENERIC COST ESTIMATES. SCIACCA, F. (Science and Engineering Associates, Inc., Albuquerque, NM). NUREG/CR-4627, Revision 2, SEA-87-253-08-A:2-Rev.2, February 1992, (44 p).

The Nuclear Regulatory Commission has sponsored a number of generic cost estimating studies. These studies were prepared to aid analysts in preparing Regulatory Impact Analyses. These generic studies provide cost estimates that would have wide application to a large number of Regulatory Analyses being performed throughout the NRC and deal primarily with repair and modification activities that may be imposed on nuclear power plants as a result of regulatory actions. Abstracts of each of the generic cost estimating studies have been prepared and assembled in this catalog. These abstracts present the results of the more detailed studies in a compact, easily understood and readily usable format. Individual abstracts have been developed to treat the main-line topics of the generic studies. In addition, abstracts have been prepared covering important sub-topics or "stand-alones" which are of broad interest in RIA preparation. Revision 2 of this catalog incorporates a new methodology for estimating radiation-related impacts for nuclear plant physical modifications.

1741. PERSPECTIVES FROM THE COMMISSION OF THE EUROPEAN COMMUNITIES. THE SAFETY OF NUCLEAR POWER: STRATEGY FOR THE FUTURE. BRINKHORST, L.J. (Commission of the European Communities, Directorate General for Environment, Nuclear Safety and Civil Protection, Brussels (Belgium)). International Conference on the Safety of Nuclear Power: Strategy for the Future, International Atomic Energy Agency, Vienna (Austria), 2-6 September 1991, CONF-910933--, 1992, pp. 23-30.

In his opening speech, Mr. L.J. Brinkhorst presented on behalf of the Commission of the European Communities (CEC) views on the current status and future perspectives of nuclear safety. Having mentioned that the international safeguards regime have been functioning rather successfully, he stressed that there was not yet anything that resembles an international nuclear safety regime, that is, a system of international treaties, conventions and practices. In this respect the experience of the Community members in enhancing the level of excellence in nuclear safety and radiation protection could be the particular contribution of the CEC to the development of international nuclear safety. Then the speaker outlined the CEC efforts in upgrading the safety of nuclear power plants in Eastern Europe. Further he stressed an importance of issues of public information with the aim to improve public acceptance of nuclear energy. In conclusion Mr. L.F. Brinkhorst pointed out that it is time to start to integrate international nuclear safety into a coherent whole.

1742. THE OECD NUCLEAR ENERGY AGENCY - FOCUSING ON THE ESSENTIALS. THE SAFETY OF NUCLEAR POWER: STRATEGY FOR THE FUTURE. STADIE, K.B. (Nuclear Energy Agency, 75, Paris (France)). International Conference on the Safety of Nuclear Power: Strategy for the Future. International Atomic Energy Agency, Vienna (Austria), 2-6 September 1991, IAEA, CONF-910933--, 1992, pp. 31-40.

Having expressed his regret that countries did not, from the start, treat international cooperation in nuclear health and safety with the same urgency as the safeguarding of nuclear fuel, the speaker dwelled on the role of the OECD Nuclear Energy Agency (NEA) in international cooperation in the areas of nuclear safety, radiation protection, radioactive waste management, operational cooperation and international projects, and nuclear regulation. In conclusion the author stressed that given its membership, Statute and

working methods, the OECD NEA was in a unique position to quickly and efficiently provide the nuclear community with advanced reflections on a broad range of key safety issues; to explore and test novel mechanisms for international interaction; and to assist Member countries in integrating efforts in safety technology and safety assessment to the maximum extent feasible.

1743. PRESIDENT'S OPENING ADDRESS. THE SAFETY OF NUCLEAR POWER: STRATEGY FOR THE FUTURE. TOEPFER, K. (Bundesministerium fuer Umwelt, Naturschutz und Reaktorsicherheit, Bonn (Germany)). International Conference on the Safety of Nuclear Power: Strategy for the Future, International Atomic Energy Agency, Vienna (Austria), 2-6 September 1991, CONF-910933--, 1992, pp. 9-21.

In the opening address the President of the conference, Federal Minister K. Toepfer, outlined the conference objectives pointing out that in the process of tackling common problems and formulating concrete strategies and measures in the area of nuclear safety it is also necessary to redefine the bases and strategies for future development of the peaceful use of nuclear power. He further dwelled on the following issues: creation of an international regime for nuclear safety (a proposal was made to start with the work on an international convention on nuclear safety), solution of actual challenges in the field of nuclear safety and radiation protection, prospects for the future utilization of nuclear power.

1744. FOR THE BENEFIT OF HUMANITY. (International Atomic Energy Agency, Vienna (Austria)). Published by Kratky Film, 1988, a 21-minute sound and color film, available in both English and German.

This film, produced for the IAEA's 30th anniversary, gives a general overview of the IAEA's activities and all aspects of the peaceful uses of nuclear energy. It shows the applications of radiation and isotopes -- industry, medicine, agriculture and hydrology. It discusses the production of electricity by nuclear power plants and presents the IAEA's activities for nuclear safety, the IAEA safeguards system and the International Nuclear Information System (INIS).

1745. HANDLING APPARATUS FOR TREATING WATER CONTAMINATED WITH RADIOACTIVITY IN NUCLEAR POWER PLANTS, AS OF JULY 11, 1991. Kerntechnischer Ausschuss (KTA), Koeln (Germany), Bundesanzeiger (Germany),

Volume 44, Number 7a, 11 January 1992, in German, pp. 18-24.

The rule contains specifications to be met in the process engineering and engineering design, construction layout and testing, to avoid uncontrolled discharge to radioactive material and to keep the radioactivity discharged to the minimum possible (sections 46 (1) No. 1 and No. 2 Radiation Protection Act (StrlSchV) and to ensure compliance with the maximum limits on the discharge of radioactive substances in effluent.

1746. AN EASY-TO-UNDERSTAND REVIEW OF THE PEACEFUL USES OF NUCLEAR ENERGY. ELECTRICITY GENERATION BY POWER PLANT. RELIABILITY AND SAFETY. THE ATOMIC PROCESSES. RECYCLING IN THE NUCLEAR FUEL CYCLE. Informationskreis Kernenergie, Bonn (Germany), Published by Inforum Verlag, 1991, in German, (42 p).

The brochure presents a brief but nevertheless complete survey of nuclear energy and its peaceful use for electricity generation, explains the design of nuclear power plant, the atomic processes, radiation protection, reactor safety engineering, and the fuel cycle.

1747. RESEARCH AND DEVELOPMENT ON DECOMMISSIONING OF NUCLEAR FACILITIES IN JAPAN. AKIYAMA, M. (Tokyo Univ (Japan)). Transactions of the 11th International Conference on Structural Mechanics in Reactor Technology. Edited by SHIBATA, H. (Tokyo University, Institute of Industrial Science (Japan)). Atomic Energy Society of Japan, Tokyo (Japan), 18-23 August 1991, CONF-910817--, 1991, pp. 45-56.

The Japan Power Demonstration Reactor (JPDR) first began nuclear power generation in Japan in 1963. Thereafter, and since the Tokai unit 1 (GCR) started commercial operation in 1986 and the Tsuruga unit 1 (BWR), and Mihama unit 1 (PWR) started commercial operation in 1970, the development, construction, and operation of nuclear power plants in Japan has continued considering to ensure the safety. Japanese commercial reactors are mainly light water reactors (BWR, PWR). Thirty nine nuclear power plants are currently being operated and nuclear power generation accounts for about 25% of the total yearly power generation in Japan. However it is forecast that plans of decommissioning commercial nuclear power plants will be drawn up soon after the mid-1990s. Therefore, after termination of operation it is necessary to

take measures to decommission nuclear power plants in order to promote the understanding of nuclear power generation in the public including the local communities. It is important to consider ensuring the safety and to establish the technology for decommissioning of nuclear power plants because of radioactive components and the massive structures, etc. and also it is important to establish the systems and regulations for the funding and the disposal of waste. Such preparation is being steadily carried out in Japan. This paper addresses status of the preparation activities for future decommissioning in Japan focussing on the policies and technological developments currently being promoted.

1748. THE SAFETY OF GERMAN NUCLEAR POWER PLANTS -- SECOND EDITION. Bundesministerium fuer Umwelt, Naturschutz und Reaktorsicherheit, Bonn (Germany), Available from FIZ Karlsruhe, September 1991, in German, (42 p).

There are in all 22 operational power reactors in the Federal Republic of Germany. Together they have a capacity of approximately 24,000 MWe. Questions concerning safety, waste management, monitoring and control and international safety partnership are dealt with. There is a comparison with the Tschernobyl concept.

1749. STATIONARY SYSTEM FOR MONITORING LOCAL DOSE RATES WITHIN NUCLEAR POWER PLANTS, AS OF JULY 11, 1991. Kerntechnischer Ausschuss (KTA), Koeln (Germany), Bundesanzeiger (Germany), Volume 44, Number 7a, 11 January 1992, in German, pp. 4-10.

The rule contains requirements to be met by the system. It determines the measurement problem and stipulates requirements concerning measuring points, measuring layout, breakdown and danger notification system, recording and registration of measuring values as well as documentation, special construction of equipment for measurements during and after incidents, maintenance and repair, as well as inspections. The rule applies to the stationary system for monitoring local dose rates in controlled areas.

1750. INDUSTRY CONSIDERATIONS OF CONCENTRATION AVERAGING. KACZMARSKY, M.M. (Ebasco Services Incorporated, New York, NY); TUIITE, P.T. (WMG Inc., Montrose, NY). Twelfth Annual U.S. DOE Low-level Waste Management Conference, Summary and Proceedings. EG&G Idaho, Inc., Idaho Falls, ID, 28-29 August 1990, CONF-9008119--, 1990, pp. 243-252.

Disposal of low level radioactive waste (LLW) which meets the requirements of 10 CFR Part 61 for Class A, B, or C waste is currently permitted at commercial disposal facilities. Disposal packages containing metallic irradiated core components are typically classified Class B or C due to the high concentration of certain radionuclides. Should an individual component contain radionuclides that exceed Class C limitations, averaging of the component is permitted with the volume of the components contained within the same disposal package. Concentration averaging is utilized extensively to successfully dispose of irradiated hardware waste from nuclear power plants. Disposal of irradiated hardware waste at commercial low level radioactive waste disposal facilities accounts for the largest volume of Class C waste, as defined by 10 CFR Part 61. Irradiated hardware waste shipments also contain the largest concentration of radioactivity when compared to all other low level waste shipments. A typical disposal liner may contain 10,000 to 20,000 curies and have a surface dose of a few thousand R/HR. The spent fuel pools at nuclear power plants were not typically designed to store large quantities of irradiated hardware waste. The impact on storage space for spent fuel therefore necessitates the removal and disposal of irradiated hardware waste. Different types of irradiated hardware waste are generated at commercial light water nuclear power plants and certain considerations are taken into account for the classification of these components to comply with 10 CFR Part 61. In addition, the concentration averaging alternative permitted by the disposal facility has a significant impact on the disposal of irradiated hardware waste components.

1751. MAINTENANCE OPERATIONS AND RADIATION PROTECTION IN FRENCH NUCLEAR POWER PLANTS. GODIN, R. (Électricité de France (EDF), Paris (France)); STRICKER, L. (Électricité de France (EDF), Saint-Denis (France)). Colloque Radioprotection et Maintenance des Centrales Nucleaires a l'Horizon 1992, 24-26 May 1988, Radioprotection (Bulletin de la Societe Francaise de Radioprotection) (France), Volume 25, Number 1, CONF-8805411--, 1990, in French, pp. 9-18.

In the field of radiation protection, the Service de la Production thermique (the Thermal and fossil generation division, SPT) at EDF aims at being among the best and at keeping for EDF agents an exposure potential allowing them to act in an emergency case. Doses to EDF or subcontractors' workers must therefore be optimized during maintenance operations.

These aims can be reached through special techniques and by a know-how of the actors at several levels: (1) upstream, operating the plant with a clean primary circuit (which limits dose rates) and solubilizing corrosion products during reactor cooling and primary circuit drainage; (2) before the intervention, thorough preparation of the work and operation of protection means suited to each site; (3) during the intervention, the actors' know-how requires suitable training and qualification of the operation leader, awareness of all the actors and, sometimes training on mockup; (4) finally, for special operations on account of their repetitiveness or because they take place in hostile places, special tools are developed. The techniques thus developed, the systematic use of feedback have made it possible to reach a most acceptable result. Radiation protection proceeds from the maintenance operation quality and thus shares in improving the operation safety of the french nuclear power plants.

1752. DEVELOPMENT OF AN ULTRASONIC CLEANING METHOD FOR FUEL ASSEMBLIES. HEKI, H.; KOMURA, S.; KATO, H.; SAKAI, H. (Toshiba Corp., Kawasaki City (Japan)); HATTORI, T. (Tokyo Electric Power Co., Kashiwazaki-shi (Japan)). Annual Meeting of the American Nuclear Society (ANS), 2-6 June 1991, Transactions of the American Nuclear Society, Volume 63, CONF-910603--, 1991, pp. 392-393.

Almost all radiation buildup in light water reactors is the result of the deposition of activated corrosion and wear products in out-of-core areas. After operation, a significant quantity of corrosion and wear products is deposited on the fuel rods as crud. At refueling shutdowns, these activation products are available for removal. If they can be quickly and easily removed, buildup of radioactivity on out-of-core surfaces and individual exposure dose can be greatly reduced. After studying various physical cleaning methods (e.g., water jet and ultrasonic), the ultrasonic cleaning method was selected as the most effective for fuel assembly cleaning. The ultrasonic cleaning method is especially able to efficiently clean the fuel without removing the channel box. The removed crud in the channel box would be swept out to the filtration unit. Parameter survey tests were carried out to evaluate the optimum conditions for ultrasonic cleaning using a mock-up of a short section of fuel assembly with the channel box. The ultrasonic device used was a 600-W ultrasonic transducer operating at 26-kHz ultrasonic frequency.

1753. METHOD FOR ENHANCING PERSONNEL SAFETY IN OPERATING A NUCLEAR REACTOR PLANT, AND MEANS THEREFOR. SKARPELOS, J.M.; LIN, C.C., Patent Number US 5028384 A, General Electric Co., San Jose, CA, July 1991.

This patent describes a method of enhancing personnel safety in the operation of a water cooled, steam producing, boiling water nuclear fission reactor wherein a portion of circulating coolant water is vaporized to steam by heat produced from fissioning nuclear fuel within a reactor pressure vessel and the produced steam with any entrained liquid water and volatile components from the coolant water is passed through a steam separator and steam dryer within the reactor pressure vessel before discharge to a power generating means. It comprises the steps of adding hydrogen to the circulating coolant water to repress corrosion within the circulating system and inhibiting the conveyance of volatile nitrogen compounds comprising ammonia with the steam from the reactor pressure vessel by catalytic oxidation with titanium dioxide of any volatile nitrogen compounds comprising ammonia carried with the steam to non-volatile nitrogen compounds comprising nitrates and nitrites.

1754. IN-PILE PWR LOOP COOLANT CHEMISTRY STUDIES IN SUPPORT OF DOSE REDUCTION. KOHSE, G.E.; SANCHEZ, R.G.; DRISCOLL, M.J.; AMES, M.; HARLING, O.K. (Massachusetts Institute of Technology, Cambridge, MA). Proceedings of the 1991 JAIF International Conference on Water Chemistry in Nuclear Power Plants. Japan Atomic Industrial Forum, Inc., Tokyo (Japan), 22-25 April 1991, CONF-9104345--, 1991, pp. 291-295.

In a project supported by the Electric Power Research Institute and the Empire State Electric Energy Research Corporation, an in-pile loop capable of close simulation of PWR primary coolant system conditions has been constructed and operated at MIT to evaluate chemistry control strategies which can reduce radioactive corrosion product deposition, and hence radiation doses to maintenance personnel. Seven one-month-long runs have been completed, in which LiOH and H₃BO₃ additions were varied to evaluate the effect of pH. Runs at pH 300°C = 6.5, 7.0, 7.2 and 7.5 were carried out. End-of-run corrosion product and radionuclide deposits were assayed as a function of position. The results are in accord with a solubility-controlled mechanism for transition metal transport, showing a substantially higher activity

deposition at low pH (6.5), and a weak minimum between pH 7.0 and 7.5.

1755. OUTAGE PERFORMANCE OF THE KASHIWAZAKI-KARIWA NUCLEAR POWER STATION. IDESAWA, MASATO (Tokyo Electric Power Co., Niigata (Japan)); YOKOTA, MASASHI (Tokyo Electric Power Co., Washington, DC). Annual Meeting of the American Nuclear Society (ANS), 2-6 June 1991, Transactions of the American Nuclear Society, Volume 63, CONF-910603--, 1991, p. 340.

The generating capacity of Japanese nuclear power plants amounts to 31,480 MW and is equivalent to ~30% of Japan's net capacity. The Tokyo Electric Power Company (TEPCO) provides service for 22 million customers, including those in metropolitan Tokyo. TEPCO operates 13 boiling water reactor (BWR) plants whose total capacity amounts to 12,396 MW; they supplied 70 billion kWh in 1990. This paper covers the maintenance management of TEPCO's nuclear power plants and provides detailed examples of the Kashiwazaki-Kariwa nuclear power station (KKNPS). Another four plants will start commercial operation by 1997 at KKNPS. Of these, the last two are advanced BWR plants, which were cooperatively developed by TEPCO and foreign BWR vendors. To meet the increasing electricity demand of recent years, TEPCO is planning to enhance preventive maintenance, plant monitoring, and application of automatic systems based on past operating experiences to ensure more safe and reliable nuclear power plants.

1756. REPORT ON RADIOACTIVE DISCHARGES AND ENVIRONMENTAL MONITORING AT NUCLEAR POWER STATIONS DURING 1990. HURST, M.J.; THOMAS, D.W., (Nuclear Electric plc, Bedminster Down (United Kingdom)). INIS-GB-413, HS/NSOB/HP-R--003., October 1991, (99 p).

This report presents the details for 1990 of radioactive discharges and environmental monitoring at Nuclear Electric sites. In addition to the main section which summarizes the discharges and monitoring at the Company's nuclear sites as a whole, appendices are presented covering the data in detail for individual sites. In each case the radiological impact on the general public has been estimated. Discharges generally were not substantially different from those of recent years. Several points relating to doses to specific groups relevant to specific sites are mentioned. All radioactive effluent discharges from

power stations were within authorization limits. Radiation doses to members of the public resulting from these discharges, and from direct radiation from the Stations, were in all cases less than the limit of 1 mSv per year recommended by ICRP 60.

1757. OSART GUIDELINES. 1992 EDITION. International Atomic Energy Agency, Vienna (Austria), IAEA-TECDOC-635, January 1992, (158 p).

The IAEA Operational Safety Review Team (OSART) Guidelines provide overall guidance for the experts to ensure the consistency and comprehensiveness of the operational safety review. Specific guidelines are provided as a guide for the systematic review in the following areas important to operational safety: management, organization and administration, training and qualification, operations, maintenance, technical support, radiation protection, chemistry, emergency planning and preparedness. Additional guidance and reference material has been prepared by the IAEA to complement the expertise of the OSART members.

1758. THE USE OF COST-BENEFIT ANALYSIS IN ALARA DECISIONS. RADIOACTIVE DISCHARGES FROM NUCLEAR POWER PLANTS. WESTERN, D.J. (Nuclear Electric plc., Bristol (United Kingdom)), Journal of Radiological Protection (United Kingdom), Volume 11, Number 3, September 1991, pp. 173-178.

In the UK, the discharge of radioactivity to the environment by nuclear power stations is closely regulated. In addition to the setting of numerical limits on discharges, the UK approach requires that "best practicable means" be adopted to reduce discharges. This is somewhat analogous to the ALARP or ALARA approach in other areas of radiological protection. Separate regulations govern areas such as radiation doses and the generation and accumulation of radioactive waste on the site. This paper highlights the issues facing a UK operator when preparing evidence to justify to the various regulatory bodies a particular level of discharge to the environment. The paper explains that the Regulations demand optimization, rather than minimization, of discharges and the shows how it should be possible to justify a waste management strategy to the Regulators by using agreed values in a cost-benefit analysis. The paper stresses the importance to the nuclear industry of securing a consistent, clear and objective approach by the Regulators and argues that the regulation of the industry under legislation relating to the minimization of harm to people and environment should not be-

come confused with issues relating either to national and international political pressures or to public relations strategies.

1759. OCCUPATIONAL EXPOSURES AT ÉLECTRICITÉ DE FRANCE NUCLEAR POWER PLANTS IN 1988. Radioprotection (Bulletin de la Société Française de Radioprotection) (France), Volume 25, Number 1, 1990, in French, pp. 65-70.

In 1988, gross electricity generation from nuclear origin (70 percent of the total French production) amounted to 257 TWh; cumulated dose was 23.2 Sv for EDF personnel (on the whole 88.12 Sv). For the EDF personnel (14,675 workers), average dose was about 1.58 mSv/year. For the whole nuclear power plants, the exposure did not exceed the maximum permissible dose equivalent (50 mSv per year or 30 mSv per quarter). No significant results were shown by occupational monitoring. Since 1983, no exceeding of admissible limits have been established (external or internal exposure).

1760. RECENT DEVELOPMENTS IN RADIO-PROTECTION TECHNOLOGY IN USA. WOOD, C.J. (Electric Power Research Institute, Nuclear Power Div., Palo Alto, CA). Radioprotection (Bulletin de la Société Française de Radioprotection) (France), Volume 25, Number 3, 1990, in French, pp. 217-224.

The implementation of technology to control radiation fields at U.S. nuclear power plants is occurring at a rapid pace and has helped utilities achieve significant reductions in occupational exposures in recent years. Recent studies to replace cobalt, develop BWR zinc injection and PWR elevated pH chemistry programs, demonstrate preconditioning methods for replacement of components, and qualify chemical decontamination have provided utilities with cost-effective methods for minimizing exposures.

1761. PROVISIONS AGAINST DAMAGE UNDER ATOMIC ENERGY LAW AND "REMAINING RISK." STEINBERG, R., ROLLER G. Published in Provisions against Damage under Atomic Energy Law in the Context of Licensing, Safeguarding of Industry and Governmental Supervision. Two Expert Opinions. STEINBERG, R. and SCHNEIDER, H.P., Published by Nomos Verlagsges., 1991, in German, pp. 9-114.

The main issue of the study is the question as to the necessary provisions against damage prescribed by Paragraph 7 Section 2 No. 3 of the Atomic Energy

Act as distinguished from the remaining risk that must be accepted. After a consideration of the scope to deny a license the concretization of the necessary provisions against damage by the Radiation Protection Ordinance and the events and measures comprised by the requirement to make provisions are examined. Further the implications for the protection of third parties of the term "provisions against damage," which has found a uniform definition through the jurisdiction of the Federal Administrative Court following the Wyhl judgment of 12.12.1985 are dealt with. Finally, the legal relevance of the "remaining risk" is discussed and legal means of negotiating it with due respect to social adequacy are sought.

1762. DECONTAMINATION DURING DECOMMISSIONING. KOCH, C. VON; GRUNER, R. (Siemens AG Unternehmensbereich KWU, Offenbach am Main (Germany)). *Bereich Energieerzeugung, Kerntechnik* (Germany), Volume 56, Number 6, December 1991, pp. 372-375.

The strategy selected for decommissioning of a nuclear facility depends to a great extent on the radiological conditions prior to dismantling. Therefore, to facilitate decommissioning, the activity inventories should be reduced by decontamination. The history of decontamination is closely related to the maintenance, repair and backfitting of nuclear power plants. As a consequence of the wide variety of existing forms of contamination, special decontamination processes have been established which are likewise suitable for decommissioning. Nevertheless, new and improved processes are being developed. As an example of a more recent development in chemical decontamination, the process CORD, which was successfully applied to the primary loop of the reactor BR 3, is described in some detail.

1763. EXPERIENCE ACCUMULATED IN EXPERT CONSULTANCY ABOUT DECOMMISSIONING NUCLEAR POWER PLANTS. RAUM, D.; SCHMID, F. (Technischer Ueberwachungs-Verein BAYERN E.V., Muenchen (Germany)). *ATW, Atomwirtschaft, Atomtechnik* (Germany), Volume 36, Number 12, December 1991, in German, pp. 551-555.

Consultancy by the Technical Inspectorate depends on a commission by the public authorities. The expert consultant is called in to examine whether the provisions required in accordance with the state-of-the-art have been made against damage arising from the decommissioning of the plant. The rules of procedure under the Atomic Energy Act define the type and

scope of the documentation to be filed in the licensing procedure. In the evaluation of these documents, no difference is made between decommissioning and erecting a plant. The disassembly manual contains all data important with respect to safety.

1764. FISCAL YEAR 1993 BUDGET ESTIMATES. Nuclear Regulatory Commission, Div. of Budget and Analysis, Washington, DC, NUREG-1100, Volume 8, January 1992, (180 p).

The U.S. Congress has determined that the safe use of nuclear materials for peaceful purposes is a legitimate and important national goal. It has entrusted the NRC with the primary Federal responsibility for achieving that goal. The NRC's mission, therefore, is to ensure adequate protection for the public health and safety, the common defense and security, and the environment in the use of nuclear materials in the United States. The NRC's scope of responsibility includes regulation of commercial nuclear power plants; research, test, and training reactors; fuel cycle facilities; medical, academic, and industrial uses of nuclear materials; and the transport, storage, and disposal of nuclear materials and wastes. The NRC carries out its mission by setting standards and requirements that licensees must meet to design, construct, and operate safe facilities, in the form of rules, license conditions, and regulatory guidance; inspecting facilities and taking enforcement action, as necessary, to ensure that such standards are followed; and conducting research to support, confirm, or refine judgments used in regulatory decisions. The technologies involved in the use of nuclear energy are relatively new and complex. Regulatory decisions often require conservatism to account for technical uncertainty. Conservatism should be modified appropriately as increased understanding of physical phenomena and interactions is achieved. Further, essential functions must be maintained through appropriate combinations of high component and system reliability, redundancy, and diversity to provide multiple barriers to the release of radiation (defense-in-depth).

1765. ACTIVE WELL COINCIDENCE COUNTER MEASUREMENTS OF ENRICHED URANIUM FUEL ASSEMBLIES IN SCANNING AND STATIONARY MODES. KRICK, M.S.; COWDER, L.R. (Los Alamos National Laboratory, NM); MALTSEV, V.; CHERNIKOV, A.; MOKEENKO, P.; D'YADKOV, K.; IVANOV, V. (Kurchatov (I.V.) Nuclear Power Plant, Sverdlovsk (USSR)); LAGATTU, A.; LOPATIN, Y.; CZOCK, K.; RUNDQUIST, D.; PEDRAZA, L. (International

Atomic Energy Agency, Vienna (Austria)). Los Alamos National Laboratory, NM, LA-12224-MS, ISPO--334, December 1991, (38 p).

The Active Well Coincidence Counter (AWCC) was evaluated at the Belyarskaya Nuclear Power Plant for the verification of BN600 enriched-uranium fuel assemblies. A custom AWCC insert, new coincidence electronics, and custom software were used. Measurements were performed in stationary and scanning modes to verify the amount of $^{235}\text{U}/\text{cm}$ and the total ^{235}U . A calibration curve for this type of fuel was established. A stationary measurement determines the amount of $^{235}\text{U}/\text{cm}$ to 2.6% (1 σ) in 1000 s; a single scan at 0.23 cm/s determines the length of the enriched region to 1.4% (1 σ) and determines the total ^{235}U to 4.2% (1 σ).

1766. INTERFACE BETWEEN RADIATION PROTECTION AND NUCLEAR SAFETY. BENGTTSSON, G. (Swedish Radiation Protection Institute, Stockholm (Sweden)); HOEGBERG, L. (Swedish Nuclear Power Inspectorate, Stockholm (Sweden)). (Nuclear Energy Agency, Paris (France)). 2nd NEA Seminar, 12-13 September 1990, Organisation for Economic Co-operation and Development, CONF-9009466--, 1991, pp. 209-221.

Interface issues concern the character and management of overlaps between radiation protection and nuclear safety in nuclear power plants. Typical examples include the selection of inspection and maintenance volumes in order to balance occupational radiation doses versus the safety status of the plant, and the intentional release to the environment in the course of an accident in order to secure better plant control. The paper discusses whether it is desirable and possible to employ a consistent management of interface issues with trade-offs between nuclear safety and radiation protection. Illustrative examples are quoted from a major Nordic research program on risk analysis and safety rationale. These concern for instance in-service inspections, modifications of plant systems and instructions after the plant has been taken into operation, and studies on the limitations of probabilistic safety assessment. They indicate that in general there are no simple rules for such trade-offs.

1767. MITIGATION OF THE IMPACT OF REDUCED RADIATION EXPOSURE LIMITS ON NUCLEAR PLANT OPERATIONS. WOOD, C.J. (Electric Power Research Institute, Palo Alto, CA). Proceedings of the 1991 JAIF International Conference on Water Chemistry in Nuclear Power Plants.

Japan Atomic Industrial Forum, Inc., Tokyo (Japan), 22-25 April 1991, CONF-910447--, 1991, pp. 4-8.

This paper assesses the impact of lower exposure limits on operations and maintenance activities in the context of the decreasing trend in exposures at U.S. nuclear power plants. The number of workers with annual doses exceeded 2 rem (20 mSv) decreased by 40% in 1989 compared to 1988, but some key experienced workers, particularly skilled labor contractors, would be impacted by possible future limits on lifetime doses. Studies at a U.S. utility with both PWR and BWR plants of the potential impact of reduced exposure limits will be described. Mitigating actions implemented by utilities are discussed, particularly advanced radiation field control technology. The status of replacement of cobalt-containing hard-facing alloys, BWR zinc injection, PWR primary system elevated pH, and preconditioning of replacement components will be reviewed.

1768. SURFACE PRETREATMENT OF PRIMARY SYSTEM COMPONENTS TO REDUCE RADIATION BUILDUP. ASAY, R.H. (Radiological and Chemical Technology, Inc., Santa Clara, CA). Proceedings of the 1991 JAIF International Conference on Water Chemistry in Nuclear Power Plants. Japan Atomic Industrial Forum, Inc., Tokyo (Japan), 22-25 April 1991, CONF-910447--, 1991, pp. 102-106.

Radiation level buildup in light water reactors systems has become of increasing concern to operating utilities. Occupational exposure reduction through engineering methods is one of the main ways to keep exposures as low as reasonably achievable (ALARA). For both the boiling and pressurized water reactor primary systems, pretreatment of the alloy steel surfaces has proven to be very cost-effective for significantly reducing the out-of-core radiation level buildup. The two main surface pretreatment processes which are readily amenable for application are; surface finishing, and applying a protective film to retard corrosion and deposition once the surface is exposed to primary coolant. Both full-scale application and long term testing has proven these two methods are complimentary in terms of maximum benefit obtained.

1769. DESIGN OF THE CONDENSATE CLEAN-UP SYSTEM FOR THE NEW ABB ATOM REACTOR GENERATION - BWR 90. BERG, A.; FEJES, P.; HEDIN, G. (ABB Atom AB, Vaesteraas (Sweden)). Proceedings of the 1991 JAIF International Conference on Water Chemistry in Nuclear Power

Plants. Japan Atomic Industrial Forum, Inc., Tokyo (Japan), 22-25 April 1991, CONF-910447--, 1991, pp. 79-84.

The design of the CCU system for the new ABB Atom reactor generation - BWR 90 - is based on the feed-back of extensive operating experience from earlier delivered systems. The worldwide development of the CCU process has also been taken into consideration. For BWR 90, a single filter/demineralizer operating in the temperature range 50-65°C and with at least 40 ppb dissolved oxygen in the condensate at the inlet of the CCU system has been selected. Special attention will be paid to vessel hydrodynamics and backwashing technique. Automatic control and proper handling equipment will furthermore be included in the system.

1770. MITIGATION OF RADIATION BUILDUP IN THE BWR BY FEEDWATER ZINC ADDITION. MARBLE, W.J.; COWAN, R.L. (General Electric Co., San Jose, CA). Proceedings of the 1991 JAIF International Conference on Water Chemistry in Nuclear Power Plants. Japan Atomic Industrial Forum, Inc., Tokyo (Japan), 22-25 April 1991, CONF-910447--, 1991, pp. 55-65.

The benefit of adding zinc to the BWR primary system as a method of reducing primary system radiation buildup was first hypothesized in 1983. Following corroborating laboratory testing, the first zinc addition system was installed at the Hope Creek plant in 1986. Currently, nine BWRs have implemented zinc addition and an additional four units are in the process of being installed. This paper provides the current status of this technology. It reviews the radiation buildup results following zinc addition and re-examines the Zn-65 outage release concern identified- at the first refueling outage for Hope Creek. The status of the search for a zinc supply depleted in Zn-64 is discussed and the potential availability schedule is provided. Recent experience on the impact of hydrogen water chemistry on BWR radiation buildup, both with and without zinc addition, is reviewed.

1771. WATER CHEMISTRY FOR DOSE REDUCTION IN THE FUKUSHIMA DAIICHI NUCLEAR POWER STATION. MIYAGAWA, T.; TAKAHASHI, T.; SAKAI, T.; TAKIZAWA, Y. (Tokyo Electric Power Co., Inc., Okuma, Fukushima (Japan)). Fukushima Daiichi Nuclear Power Station). Proceedings of the 1991 JAIF International Conference on Water Chemistry in Nuclear Power Plants.

Japan Atomic Industrial Forum, Inc., Tokyo (Japan), 22-25 April 1991, CONF-910447--, 1991, pp. 44-48.

From the commencement of Japanese Boiling Water Reactor (BWR) plants in 1970, radioactive corrosion products accumulating in the primary system were observed to increase, and occupational exposure during annual inspections rose. Occupational exposure in BWRs has shown continuous decrease since 1982 owing to the enforcement of activity control measures.

1772. ACTUAL RECORDS AND PROSPECT OF LONG-TERM DOSE REDUCTION BY IRON CONTROL IN FEEDWATER AT NEWLY CONSTRUCTED BWR PLANTS. SAITO, S. (Tohoku Electric Power Co. Ltd., Sendai (Japan)); OTOHA, K.; HASHIMOTO, H.; MIZUKAMI, K.; MIYAZAKI, H.. Proceedings of the 1991 JAIF International Conference on Water Chemistry in Nuclear Power Plants. Japan Atomic Industrial Forum, Inc., Tokyo (Japan), 22-25 April 1991, CONF-910447--, 1991, pp. 39-43.

The newly constructed BWR plants, of which condensate purification systems were modified to improve crud removal, have been using the so-called iron control in feedwater, making significant dose reduction. At present, the number of plants that have been practicing iron control in feedwater from the first cycle operation is nine, and the actual record of operation is about 30 reactor-years. The electric utilities will continue to improve this method and apply it to new plants to maintain long-term dose reduction.

1773. OCCUPATIONAL DOSE REDUCTION AT DEPARTMENT OF ENERGY CONTRACTOR FACILITIES: BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA. DIONNE, B.J.; LANE, S.G.; BAUM, J.W. (Brookhaven National Laboratory, Upton, NY), DOE/EH-0234T, BNL--43228, Volumes 1-3, November 1991, (127 p).

Promoting the exchange of information related to implementation of the As Low as Reasonably Achievable (ALARA) philosophy is a continuing objective for the Department of Energy (DOE). This report, prepared by the Brookhaven National Laboratory (BNL) ALARA Center for the DOE Office of Health, contains the third in a series of bibliographies on dose reduction at DOE facilities. This report also contains abstracts from the two previous volumes. The BNL ALARA Center was originally established in 1983

under the sponsorship of the U.S. Nuclear Regulatory Commission to monitor dose-reduction research and ALARA activities at nuclear power plants. This effort was expanded in 1988 by the DOE's Office of Environment, Safety and Health to include DOE nuclear facilities. This bibliography contains abstracts relating to various aspects of ALARA program implementation and dose-reduction activities, with a specific focus on DOE facilities. Abstracts included in this bibliography were selected from proceedings of technical meetings, journals, research reports, searches of the DOE Energy Data Base, and reprints of published articles provided by the authors. Facility types and activities covered in the scope of this report include: radioactive waste, uranium enrichment, fuel fabrication, storage, and reprocessing, facility decommissioning, hot laboratories, tritium production, research, test and production reactors, weapons fabrication and testing, and accelerators. Material on improved shielding design, decontamination, containments, robotics, job planning, improved operational techniques, and other topics are also included.

1774. A NUCLEAR SAFETY VIEW ON THE PROPOSED ICRP APPROACH TO PROBABILISTIC EXPOSURES AND THE POSSIBILITIES AND LIMITATIONS OF USING PRA IN THIS FIELD. TANGUY, P.Y. (Électricité de France (EDF), Paris (France)). The Interface in Nuclear Safety and Public Health. Nuclear Energy Agency, Paris (France), Proceedings of the 2nd NEA Seminar, 12-13 September 1990, Organisation for Economic Co-operation and Development, CONF-9009466--, 1991, pp. 153-162.

The extension of the ICRP system of radiation protection to encompass potential exposures in the new recommendations published by that body in early 1990 represents a significant innovation. The author reviews and provides a personal commentary on the reservations which the International Consultative Committee for Nuclear Safety of the IAEA has voiced with regard to these recommendations. The paper also reviews the potential contribution that Probabilistic Safety Analyses can make in nuclear power plant safety management, and discusses their limitations and role according to level of assessment.

1775. IMPACT OF REDUCED RADIATION EXPOSURE LIMITS ON NUCLEAR PLANT OPERATIONS. FAY, C.W. (Wisconsin Electric Power Co., Milwaukee, WI); WOOD, C.J. (Electric Power Research Inst., Palo Alto, CA). The Interface in Nuclear Safety and Public Health. Nuclear Energy

Agency, Paris (France), Proceedings of the 2nd NEA Seminar, 12-13 September 1990, Organisation for Economic Co-operation and Development, CONF-9009466--, 1991, pp. 175-185.

This paper assesses the impact of lower exposure limits on operations and maintenance activities in the context of the decreasing trend in exposures at U.S. nuclear power plants. The number of workers with annual doses exceeding 2 Rem (20 mSv) decreased by 40% in 1989 compared to 1988, but some key experienced workers, particularly skilled labor contractors, would be impacted by possible future limits on lifetime doses. Mitigating actions implemented by utilities are discussed, including dose leveling, enhanced training programs and radiation field control technology.

1776. A REGULATORY VIEW ON THE APPLICABILITY OF THE NEW ICRP RECOMMENDATIONS TO NUCLEAR SAFETY ASPECTS. GONZALEZ G.E. (Consejo de Seguridad Nuclear, Madrid (Spain)); NAEGELIN, R. (Swiss Federal Nuclear Safety Inspectorate, Wuerenlingen (CH)); VUORINEN, A.P.U. (Finnish Centre for Radiation and Nuclear Safety (STUK), Helsinki (Finland)). The Interface in Nuclear Safety and Public Health. Nuclear Energy Agency, Paris (France), Proceedings of the 2nd NEA Seminar, 12-13 September 1990, Organisation for Economic Co-operation and Development, CONF-9009466--, 1991, pp. 163-174.

The new ICRP recommendations will cause several changes in the radiological practice. This paper discusses these changes from the point of regulatory view of nuclear safety. To avoid adverse short-term effects, the new risk estimates should be adopted in radiation protection standards with great care. The ultimate objective of nuclear safety is to protect people environment and property against radiological hazards. Improvements in principles and practices developed by the ICRP are important in reaching the primary goal. A severe nuclear accident must be prevented in advance. Every scientific and technical means have to used; optimization is not the solution of the problem.

1777. EUROPEAN SAFETY APPROACHES. RYDER, E.A. (Health and Safety Executive, Nuclear Installation Inspectorate, London (United Kingdom)); LAVERIE, M. (Ministere de l'Industrie et de la Recherche, Paris (France)). Service Central de Surete des Installations Nucleaires; GAST, K. (Bundesministerium fuer Umwelt, Naturschutz und Reaktorsicher), The Interface in Nuclear Safety and Public Health.

Nuclear Energy Agency, Paris (France), Proceedings of the 2nd NEA Seminar, 12-13 September 1990, Organisation for Economic Co-operation and Development, CONF-9009466--, 1991, pp. 89-98.

The paper reviews the European development and approaches towards safety. It concludes that comparable levels of safety are achieved and that there are similarities in approaches, although the individual country's expression of probabilistic and deterministic safety criteria and standards may vary. Current work being undertaken by European countries under the auspices of OECD-NEA, and the CEC in the fields of radiation protection and nuclear safety are reviewed.

1778. THE PRINCIPAL NUCLEAR SAFETY POLICIES IN JAPAN. UCHIDA, H. (Nuclear Safety Commission, Tokyo (Japan)). The Interface in Nuclear Safety and Public Health. Nuclear Energy Agency, Paris (France), Proceedings of the 2nd NEA Seminar, 12-13 September 1990, Organisation for Economic Co-operation and Development, CONF-9009466--, 1991, pp. 99-115.

The basic policy for nuclear safety in Japan is to give first priority to the prevention of occurrence and propagation of abnormal events rather than to the mitigation of the consequences of incidents and accidents. Preventive maintenance is an important countermeasure for aging degradation. The seismic design policies and other specific design concepts relevant to safety are discussed. It is recognized that PSA is important and indispensable for comparing safety problems. However, the applicable limitations associated with PSA should be addressed. The interface issues in Health Risk and Nuclear Safety should be discussed by concerned experts.

1779. THE UNITED STATES SAFETY GOALS. REMICK, F.J. (Nuclear Regulatory Commission, Washington, DC). The Interface in Nuclear Safety and Public Health. Nuclear Energy Agency, Paris (France), Proceedings of the 2nd NEA Seminar, 12-13 September 1990, Organisation for Economic Co-operation and Development, CONF-9009466--, 1991, pp. 79-88.

Following the accident at Three Mile Island, the U.S. Nuclear Regulatory Commission imposed many new requirements. The NRC's credibility with the general public was at an all time low, while regulatory instability was at an all time high. In partial response to those concerns, the NRC issued in 1986 a policy statement on safety goals. It became obvious that implementation of the safety goals was not an easy

task. This paper describes recent Commission directives to its staff on implementation of the Safety Goals Policy Statement.

1780. THE PROBLEMS OF ACHIEVING NUCLEAR SAFETY OBJECTIVES. KOUTS, H.J.C. (Defense Nuclear Facilities Safety Board, Washington, DC). The Interface in Nuclear Safety and Public Health. Nuclear Energy Agency, Paris (France), Proceedings of the 2nd NEA Seminar, 12-13 September 1990, Organisation for Economic Co-operation and Development, CONF-9009466--, 1991, pp. 47-54.

The problems inherent in setting safety goals and in assessing their achievement are discussed. A difficulty underlying all such actions is the need for probabilistic safety assessments, which are not as widespread as they need to be for use of safety goals. The difficulties associated with use of PSA are discussed. Other difficulties in achieving goals stem from the age of some plants, so that they do not have all modern recognized hardware features, and the resistance of some managements to change. The problems of assessing the meeting of safety goals are discussed.

1781. TRANSLATIONAL REPROCESSING OF SPENT FUEL ELEMENTS IN THE LIGHT OF EUROPEAN COMMUNITY LAW. SCHEUING, D.H., a book from the Schriftenreihe Europaisches Recht, Politik und Wirtschaft Series, Volume 148, Published by Nomos Verlagsges, 1991, in German, (93 p).

Objections are being raised against the current reprocessing of fuel elements from German nuclear power plants in France and Great Britain on the ground that, measured by German protection requirements, it cannot be regarded as "innocuous utilization" of radioactive waste material; this brings a momentous intervention of the German authorities against the operators of German nuclear power plants into consideration. Yet would not such a "national solo attempt" conflict with European Community law? This question is illuminated in its different aspects. First the issue is examined from the point of view of radiation protection law under the Euratom Treaty and of the aim of the European Community to establish the single market. Subsequent focal points are an inquiry into compatibility with the freedom of merchandise traffic and commercial services as provided by European Community law. The outcome is that European Community law does not oppose the German authorities intervening. Rather, such self-discipline practiced by member states of the benefit of the European environment is admissible so long as the

other European Community members states do not establish equally stringent standards on their own accord of European Community law itself does not provide protection on a high level.

1782. FOUNDATION FOR THE ADEQUACY OF THE LICENSING BASES. Nuclear Regulatory Commission, Washington, DC, Office of Nuclear Reactor Regulation, NUREG-1412, December 1991, (109 p).

The objective of this report is to describe the regulatory processes that assures that any plant-specific licensing bases will provide reasonable assurance that the operation of nuclear power plants will not be inimical to the public health and safety to the end of the renewal period. It is on the adequacy of this process that the Commission has determined that a formal renewal licensing review against the full range of current safety requirements would not add significantly to safety and is not needed to assure that continued operation throughout the renewal term is not inimical to the public health and safety or common defense and security. This document illustrates in general terms how the regulatory process has evolved in major safety issue areas. It also provides examples illustrating why it is unnecessary to re-review an operating plant's basis, except for age-related degradation unique to license renewal, at the time of license renewal. The report is a supplement to the Statement of Considerations for the Nuclear Regulatory Commission's rule (10 CFR Part 54) that established the criteria and standards governing nuclear power plant license renewal.

1783. THE POSSIBILITIES AND LIMITS OF INCREASING POWER FROM A TECHNICAL POINT OF VIEW. BURNS, D. (Kernkraftwerk Leibstadt ag, Leibstadt (Switzerland)). SVA Information Meeting: "How Old do Power Plants Grow," 16 September 1991, Schweizerische Vereinigung fuer Atomenergie (SVA), CONF-9109324--, 1991, in German, pp. R6-1 - R6-6.

Experience shows that increasing the power of existing nuclear plants allows increased production of electrical power in reasonable conditions. An increase in the thermal power of up to 15% is possible in principle by exploiting the reserves without reducing the safety, availability or service life of the plant. The systematic examination of all relevant aspects of the plant and its operational management is a precondition, however. An improvement in efficiency may also be attractive although this rather limits the

additional production but is nevertheless extremely attractive from an economic point of view.

1784. NUCLEAR SAFETY REVIEW. Published in the IAEA Yearbook 1991. International Atomic Energy Agency, Vienna (Austria), IAEA, 1991, pp. D1-D97.

A survey of IAEA activities in the field of nuclear safety with particular emphasis on the international co-operation in safety assessment of research reactors, early WWER type reactors is given. Consequences of the Chernobyl accident for human health and environmental contamination have been assessed in a co-operative IAEA venture, comprehending the commission of the European communities. Basic safety standards have been reviewed and harmonized, operational safety upgraded, service life of nuclear power plants with regard to their safety has been studied as well as human reliability and accident prevention. Radioactive waste management, impacts of natural radioactivity (radon), transport of radioactive material, emergency planning and preparedness were the other important topics of IAEA activities in 1991.

1785. THE NORDIC PROGRAM FOR NUCLEAR SAFETY 1990-1993. Aarsrapport for 1990. Nordisk Kontaktorgan for Atomenergispørgsmaal, Roskilde (Denmark), NKS-91-1, February 1991, in Danish, Swedish, Norwegian, and English (56 p).

The status of ongoing projects under The Nordic Program for Nuclear Safety (NKS) 1990-1993, and the economy of the program is presented. A review of projects, projects managers and coordinators, and a list of members of NKS and associated members is included.

1786. CURRENT STATUS OF PERSONNEL EXPOSURE AT NUCLEAR POWER PLANTS AND OTHER MEDICAL, INDUSTRIAL AND EDUCATIONAL FACILITIES IN JAPAN. SASAKI, FU. (Japan Power Engineering and Inspection Corp., Tokyo (Japan)). Thermal and Nuclear Power (Japan), Volume 42, Number 7, July 1991, in Japanese, pp. 848-855.

The state of radiation exposure of the workers engaging in radiation works in Japanese nuclear power stations, the factors of the radiation exposure of the workers engaging in radiation works, the countermeasures for reducing exposure in nuclear power stations, the state of radiation exposure of doctors, the workers engaging in radiation works, researchers and others in

medical, industrial, research and educational and other facilities in Japan, the factors of their radiation exposure and the countermeasures for reducing the exposure, and the comparison of the exposure in nuclear power stations with that in medical, industrial, research and educational facilities are reported.

1787. ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED. Nuclear Regulatory Commission, Office of Enforcement, Washington, DC, NUREG-0940, Volume 10, Number 3, November 1991, (215 p).

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (July-September 1991) and includes copies of letters, Notices, and Orders sent by the U.S. Nuclear Regulatory Commission to licensees with respect to these enforcement actions. It is anticipated that the information in this publication will be widely disseminated to managers and employees engaged in activities licensed by the NRC, so that actions can be taken to improve safety by avoiding future violations similar to those described in this publication.

1788. COLLECTIVE RADIATION DOSE IN WEST EUROPEAN POWER REACTORS AND SOME OBSERVATIONS ON DISTRIBUTION OF DOSES BETWEEN INDIVIDUAL WORKERS. SCHNUER, K. (Directorate General, Environment, Nuclear Safety and Civil Protection, Commission of the European Communities (CEC), Luxembourg (Luxembourg)); BROOKES, I.R. (Nuclear Electric PLC, Barnwood (United Kingdom)). Workshop on Statistics of Human Exposure to Ionizing Radiation, 2-4 April 1990, Radiation Protection Dosimetry (United Kingdom), Volume 36, Numbers 2-4, CONF-900438--, 1991, pp. 187-189.

Collection, analysis and discussion of radiation dose statistics is one way in which the European Commission promotes the ALARA (As Low As Reasonably Achievable) principle in nuclear power plants of member countries. Some of the analyses undertaken are presented and the following major conclusions drawn: (1) Collective dose and the number of people receiving dose are log-normally distributed through the reactor population; (2) Collective dose in European water reactors is slowly increasing with age; (3) The number of people receiving measurable dose at each site varies from 300 to over 2000 with a geometric mean value of about 1000; and (4) Average individual dose, collective dose and individual dose distributions are closely related and the relationship appears to be independent of the reactor type. If this

relationship is not recognized when setting criteria for individual, average individual and collective dose, incompatibility can arise and cause difficulties when criteria are applied in operation and design.

1789. A SURVEY OF DOSES TO WORKER GROUPS IN THE NUCLEAR INDUSTRY. KHAN, T.A.; BAUM, J.W. (Brookhaven National Laboratory, Upton, NY), 1991 Radiation Exposure Management Seminar, 20-23 October 1991, BNL-NUREG-46742, CONF-9110253--1, 1991, (13 p).

The U.S. National Council on Radiation Protection and Measurements (NCRP) has suggested "...as guidance for radiation programs that cumulative exposure not exceed the age of the individual in years x 10 mSv (years x 1 rem)." The International Commission on Radiological Protection (ICRP) has recommended a dose limit of 10 rem averaged over 5 years. With these developments in mind, the U.S. Nuclear Regulatory Commission (NRC) requested the ALARA Center of the Brookhaven National Laboratory to undertake two parallel studies. One study, which is still ongoing, is to examine the impact of the newly recommended dose limits on the nuclear industry as a whole; the other study was intended to assist in this larger project by looking more closely at the nuclear power industry. Preliminary data had indicated that the critical industry, as far as the impact of new regulatory limits were concerned, would be the nuclear power industry, because, it was conjectured, there existed a core of highly skilled workers in some groups which routinely get higher than average exposures. The objectives of the second study were to get a better understanding of the situation vis á vis the nuclear power industry, by identifying the high-dose worker groups, quantifying the annual and lifetime doses to these groups to see the extent of the problem if there was one, and finally to determine if there were any dose-reduction techniques which were particularly suited to reducing doses to these groups. In this presentation, the authors describe some of the things learned during our work on the two projects. For more detailed information on the project on dose-reduction techniques for high-dose worker groups in the nuclear power industry, see NUREG/CR-5139. An industry/advisory committee has been set up and is in the process of evaluating the data from the larger project on the impact of new dose limits and will shortly produce its report.

1790. THIRTY YEARS OF INTERNATIONAL NUCLEAR STANDARDS IN ISO AND IEC.

BECKER, K. (Din German Nuclear Standard Committee, Berlin (Germany)), 7-10 November 1988, Nuclear Engineering and Design (Netherlands), Volume 127, Number 2, CONF-881106--, June 1991, pp. 151-157.

Of over 4700 nuclear and radiation protection standards, laws, regulations, recommendations, etc., which have been issued by 52 countries and several governmental international organizations such as IAEA, or non-government organizations such as ICRP, ICRU, ISO and IEC, more than 90% are national, and more than half of those of a non-regulatory nature. Both number and importance of international standards are, however, increasing. The two most important non-governmental international standards organizations are the International Organization for Standardization (ISO), which founded a Technical Committee (TC) 85 "Nuclear Energy" with several subcommittees in 1957, and the closely associated International Electrotechnical Commission (IEC) with its TC 45 "Nuclear Instrumentation" formed in 1960. There are over 20 member countries actively participating in the work in each of them, with additional observer countries and liaison to the relevant other international organizations. A brief review is given on some of the experience which has been gained by ISO/TC 85 and IEC/TC 45, and its work program and accomplishments, considering in particular recent developments. It covers nuclear safety and instrumentation, radiation protection, and the nuclear fuel cycle, as well as interface issues including definitions and terminology, interaction of the activities with the work of other national and international bodies, etc. Some problem areas are also briefly discussed.

1791. EMISSION MONITORING OF NUCLEAR INSTALLATIONS. EDELHAEUSER, H.; GANS, I.; RUEHLE, H.; WINKELMANN, I., Published in the Proceedings of Together for Radiation Protection, Compiled by NARROG, J., Fachverband fuer Strahlenschutz e.V., Karlsruhe (Germany), 19-20 October 1990, FS-90-53-T, CONF-9010443--, 1990, in German, pp. 56-77.

The paper deals with emission monitoring based on the rules of the Nuclear Engineering Committee, during normal operation of nuclear power plants, on the one hand, in particular waste water and exhaust air monitoring, control measuring by independent measuring points, and on the other hand, in the event of incidents and accidents. Furthermore, it deals with nuclear reactor remote monitoring as and instrument of surveillance according to section 19 of the Atomic

Energy Act. Remote monitoring of reactors has to be further developed in order to efficiently fulfill the new tasks under the information agreements between the European Communities and the IAEA.

1792. IMMISSION MONITORING OF NUCLEAR INSTALLATIONS. WINTER, M. (Kernforschungszentrum Karlsruhe GmbH, Hauptabteilung Sicherheit (Germany)). Published in the Proceedings of Together for Radiation Protection, Compiled by NARROG, J., Fachverband fuer Strahlenschutz e.V., Karlsruhe (Germany), 19-20 October 1990, FS-90-53-T, CONF-9010443--, 1990, in German, pp. 78-94.

The paper describes first of all the background of environmental radioactivity monitoring in the Federal Republic of Germany, and then explains the tasks and importance as well as implementation of environmental monitoring: setting up of monitoring program; regulations and dose limits under the Radiation Protection Ordinance; structure and scope of monitoring programs. Environmental monitoring of nuclear installations can, by and large, rightfully be considered a resolved task. The restricting formulation of environmental monitoring. It can be considered optimal the moment its defined protective aim, namely proof of observing the limit values specified in the Radiation Protection Ordinance, is achieved at minimum expense. The legislator and the competent authorities within the Federal Government and in the States remain called upon to see to it that more importance is attached in future to the flood of measuring results from environmental monitoring when assessing real or hypothetical impacts by nuclear installations on the environment.

1793. PROSPECTS OF RADIATION PROTECTION IN THE EUROPEAN COMMUNITIES. ERISKAT, H. (Commission of the European Communities, Div of Radiation Protection, Luxembourg (Luxembourg)). Published in the Proceedings of Together for Radiation Protection, Compiled by NARROG, J., Fachverband fuer Strahlenschutz e.V., Karlsruhe (Germany), 19-20 October 1990, FS-90-53-T, CONF-9010443--, 1990, in German, pp. 305-317.

The main task of the Communities in the field of radiation protection consists of establishing uniform safety standards, and see to it that they are applied, with the member states being responsible not only for regulating radiation protection on their territories, but also promoting international harmonization of the respective laws. In anticipation of the new ICRP

recommendations announced some time ago, the Commission has initiated preliminary work towards a partial review the aim of streamlining the updating radiation protection without questioning the main principles of the existing Euratom basic standard guideline. However, it seeks to come to an agreement in this regard with other international organizations.

1794. NUCLEAR ENERGY LEGISLATION IN THE FEDERAL REPUBLIC OF GERMANY IN 1989. NOBBE, U. Published in the *Yearbook of the Nuclear Industry 1990*. Edited by MUELLER, W.D.; HOSSNER, R. Published by Verlagsgruppe Handelsblatt, 1990, in German, pp. A63-A76.

Among the most important activities of the Federal government and Parliament, are some important amendments to the Nuclear Power Act (Atomgesetz) and the quarterly reports on radioactivity in the environment and exposure to radiation, and on incidents in nuclear power plants. At legislative and regulatory level, the new Radiation Protection Regulations came into force, aimed at harmonizing with the basic European Community standards, and also the act to set up a Federal Office of Radiation Protection. Applications for modification in a number of nuclear power plants were submitted or granted within the administrative procedures. The Reactor Safety Commission passed its recommendation on the pilot reprocessing plant at Gorleben in 1989; the Radiation Protection Commission passed its recommendation on the radiological assessment of plutonium. The Federal Administrative Court and Higher Administrative Courts issued important rulings with fundamental significance for the control by administrative courts over nuclear licensing procedures under law. For instance, the enforcement of standards action against the Nuclear Power Act (Atomgesetz), the controversy between federal and state governments over the authority of orders from the Federal Department of the Environment; further rulings such as the one on armed security guards, the Ahaus ZWL and the operating license for Wuergassen, on the extent and scope of retrofitting and the use of MOX fuel elements.

1795. PECULIARITIES OF THE ATOMIC ENERGY ACT AND THE RADIATION PROTECTION LAWS IN THE NEW FEDERAL STATES OF GERMANY. KRUEGER, F.W.; ARNDT, H. (Bundesamt fuer Strahlenschutz, Berlin (Germany)). Published in the *Proceedings of Together for Radiation Protection*. Compiled by NARROG, J. Fachverband fuer Strahlenschutz e.V.,

Karlsruhe (Germany), 19-20 October 1990, FS-90-53-T, CONF-9010443--, 1990, in German, pp. 36-41.

The new section 57a of the Atomic Energy Act provides that licenses and permits for nuclear power plants in the new Federal States become ineffective by June 30, 1995; for the transport of radioactive materials by June 30, 1992. Moreover, it provides that all other licenses and permits become ineffective by June 30, 2000, unless a shorter expiration term is fixed in any of them. Licenses and permits continuing in force are considered to have been granted according to the Atomic Energy Act. Gradually, those licenses and permits, in particular their supplementary regulations, have to be brought into agreement with the now valid atomic energy and radiation protection supervisory procedures, and with the technical and organizational radiation protection requirements. Not later than by the dates specified above, complete agreement must have been achieved with the new legal regulations.

1796. OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA. KHAN, T.A.; VULIN, D.S.; LANE, S.G.; BAUM, J.W. (Brookhaven National Laboratory, Upton, NY). Nuclear Regulatory Commission, Div. of Regulatory Applications, Washington, DC, NUREG/CR-3469, Volume 6, BNL-NUREG--51708, Volume 6, October 1991, (70 p).

In the continuing effort to collect and disseminate information on radiation dose reduction at nuclear power plants, the ALARA Center at Brookhaven National Laboratory publishes a series of bibliographies of selected readings in radiation protection and ALARA. This is the sixth report in that series. The abstracts in this bibliography were selected from proceedings of technical meetings and conferences, journals, research reports, and searches of information databases of the U.S. Department of Energy. The subject material of these abstracts relates to radiation protection and dose reduction, and ranges from the use of robotics, to operational health physics, to water chemistry. Also included is material on the design, planning, and management of nuclear power stations, as well as on decommissioning and safe storage efforts. This report contains 266 abstracts, along with subject and author indices. The author index is exclusively for this volume. The subject index contains headings for this volume in bold face, as

well as reference to previous volumes. All information in this and previous volumes of the series is also available through our on-line information system called ACE (ALARA Center Exchange). ACE is accessible through fax machines or personal computers interfaced with modems. The bibliography database and other databases are kept current with new abstracts, information on research projects, and recent news of international events related to ALARA at nuclear power plants. Access to the system is provided freely to the ALARA community. For password certification, manuals, and other information about our system, please contact the ALARA CENTER, Building 703M, Brookhaven National Laboratory, Upton, NY 11973, or call (516) 282-3228.

1797. SAFETY IS THE KEY TO A NUCLEAR FUTURE. ROGERS, K.C., Public Utilities Fortnightly, Volume 127, Number 7, 1 April 1991, pp. 54-56.

Perhaps the most important element of retaining commercial nuclear power as a viable option for meeting the energy needs of our nation is leadership in the pursuit of excellence in nuclear safety. Excellence is achieved by the high-quality performance of people - people who develop the advanced technologies involved, people who design the plants, people who build the plants, people who operate the plants, people who maintain the plants, and people from the author's own organization, the Nuclear Regulatory Commission (NRC), who regulate these activities. The commission and those in the industry have many common objectives, and there are some specific steps that need to be taken by both the NRC and the industry to achieve them. This article examines those steps.

1798. GDR EXPERIENCE FROM THE USE OF A NUCLEAR POWER STATION WITH WWER-440 REACTORS FOR DISTRICT HEATING PURPOSES. BEDNAREK, B. (Zachodni Okreg Energetyczny-Poznan (Poland)). Energetyka (Poland), Volume 41, Number 3, 1987, in Polish, pp. 86-88.

GDR experiences from the use of Bruno Leuschner nuclear power plant for Greifswald heating are described. The stress is given on safety analysis.

1799. SECURE AND EFFECTIVE VALVE STEM SEALING IN PWR POWER GENERATING PLANTS. REYNOLDS, J. (Latty International, Chesterfield, VA). Proceedings of the Third EPRI

Power Plant Valves Symposium. Edited by EVANS, S.O. (EPRI Nondestructive Evaluation Center, Charlotte, NC). Electric Power Research Inst., Palo Alto, CA, EPRI Nondestructive Evaluation Center, Charlotte, NC, 20-23 August 1990, EPRI-NP/GS-7339, CONF-9008113--, June 1991, pp. 4B.33-4B.41.

The PWR power generating plant combines severe operating conditions with the highest safety requirements, making it one of the most demanding environments for seals. An analysis of the conditions inherent in its operation reveals: an aggressive and radioactive fluid at high temperature and pressure; frequent thermal shocks; and hazards for maintenance personnel in the containment area unless the reactor is shut down. The achievement of today's quality and safety standards owes much to the experience, research and testing carried out by the Électricité de France during its graduation from its first nuclear unit to become the world's most important manager of PWR plants with over 45 now under its control. The number of valves involved in the French nuclear program is in excess of 1,300,000. Knowing what the affect of a leak can be, especially if it necessitates a shutdown of the power station, the need to insure the quality of valve sealing can be appreciated. At the beginning of their nuclear building program, the EdF was finding that valves, representing only 2 percent of the investment in a PWR plant, caused 20% of the unwanted outages and cost 60% of the total of plant maintenance. In this report, the author endeavors to show how this problem was solved by team work and concerted action by the EdF, the valve constructors and seal manufacturer, not forgetting the importance of informing and training the maintenance and repair teams within the power stations themselves.

1800. OPERATION OF FINNISH NUCLEAR POWER PLANTS. Edited by TOSSAVAINEN, K. Finnish Centre for Radiation and Nuclear Safety (STUK), Helsinki (Finland). Dept. of Nuclear Safety, STUK-B-YTO-88, August 1991, (20 p).

In the Quarterly Reports on the operation of the Finnish nuclear power plants such events and observations are described relating to nuclear and radiation safety which the Finnish Centre for Radiation and Nuclear Safety considers safety significant. Also other events of general interest are reported. The report also includes a summary of the radiation safety of the plants' workers and the environment, as well as tabulated data on the production and load factors of the plants. The Finnish nuclear power plant units Loviisa 1 and 2 as well as TVO I and II were in

commercial operation during the whole first quarter of 1991. The load factor average was 99.1 %. Failures have been detected in the uppermost spacing lattices of nuclear fuel bundles removed from the Loviisa nuclear reactors. Further investigations into the significance of the failures have been initiated. In this quarter, renewed cooling systems for the instrumentation area were introduced at Loviisa 1. The modifications made in the systems serve to ensure reliable cooling of the area even during the hottest summer months when the possibility exists that the temperature of the automation equipment could rise too high causing malfunctions which could endanger plant safety. Occupational radiation doses and external releases of radioactivity were below prescribed limits in this quarter. Only small amounts of radioactive substances originating in nuclear power plants were detected in samples taken in the vicinity of nuclear power plants.

1801. SAFETY ASSESSMENT OF SFR - A SWEDISH REPOSITORY FOR LOW AND INTERMEDIATE LEVEL RADIOACTIVE WASTE. CARLSSON, J.; PAPP, T.; FORSSTROM, H. (Swedish Nuclear Fuel Co. Stockholm, (Sweden)). Proceedings of the Symposium on Safety Assessment of Radioactive Waste Repositories. Nuclear Energy Agency, Paris (France), 9-13 October 1989, Organisation for Economic Co-operation and Development, CONF-891008--, 1990, in French; pp. 157-166.

In 1988 a license for operation of the final disposal facility for reactor waste, SFR, was given by the Swedish authorities. The SFR has been designed to make possible a simple and controllable as well as a safe disposal of the low-and intermediate level radioactive material arising from the operation and maintenance of the Nuclear power plants in Sweden. The safety of the repository has been assessed based on scenarios and models selected in a reasonably pessimistic way. The calculations show that the release can be expected to be well below the radiation protection limits. A further safety evaluation will be required before a permission for sealing is given.

1802. HANDLING APPARATUS FOR TREATING WATER CONTAMINATED WITH RADIOACTIVITY IN NUCLEAR POWER PLANTS. Kerntechnischer Ausschuss (KTA), Koeln (Germany), Published by Heymanns, KTA--3603 (draft), 1990, in German, (15 p).

The rule contains specifications to be met in the process engineering and engineering design, construction layout and testing, to avoid uncontrolled

discharge of radioactive material and to keep the radioactivity discharged to the minimum possible (sections 46(1) No. 1) and No. 2 Radiation Protection Act (StrlSchV) and to ensure compliance with the maximum limits on the discharge of radioactive substances in effluent.

1803. DECOMMISSIONING A NUCLEAR REACTOR. WATER BOILER REACTOR PROJECT. MONTOYA, G.M., Los Alamos National Laboratory, NM, Environmental Remediation '91 Conference, 8-11 September 1991, LA-UR-91-2874, CONF-910981--18, 1991, (24 p).

The process of decommissioning a facility such as a nuclear reactor or reprocessing plant presents many waste management options and concerns. Waste minimization is a primary consideration, along with protecting a personnel and the environment. Waste management is complicated in that both radioactive and chemical hazardous wastes must be dealt with. This paper presents the general decommissioning approach of a recent project at Los Alamos. Included are the following technical objectives: site characterization work that provided a thorough physical, chemical, and radiological assessment of the contamination at the site; demonstration of the safe and cost-effective dismantlement of a highly contaminated and activated nuclear-fueled reactor; and techniques used in minimizing radioactive and hazardous waste.

1804. GENERIC ENVIRONMENTAL IMPACT STATEMENT FOR LICENSE RENEWAL OF NUCLEAR PLANTS. Nuclear Regulatory Commission, Washington, DC, Div. of Safety Issue Resolution, NUREG-1437, Volume 2, August 1991, (597 p).

The U.S. Nuclear Regulatory Commission (NRC) anticipates that it will receive applications for renewal of the operating licenses of a significant portion of existing nuclear power plants. This Generic Environmental Impact Statement (GEIS) examines the possible environmental impacts that could occur as a result of renewing licenses of individual nuclear power plants under the proposed 10 CFR Part 54. The GEIS, to the extent possible, establishes the bounds and significance of these potential impacts. The analyses in the GEIS encompass all operating light-water power reactors. For each type of environmental impact the GEIS attempts to establish generic findings covering as many plants as possible. While plant and site-specific information is used in developing the generic findings, the NRC does not intend for the GEIS to be a compilation of individual plant environmental impact statements. This GEIS has three

principal objectives: (1) to provide an understanding of the types and severity of environmental impacts that may occur as a result of license renewal of nuclear power plants under 10 CFR Part 54; (2) to identify and assess those impacts that are expected to be generic to license renewal, and (3) to support a rulemaking (10 CFR Part 51) to define the number and scope of issues that need to be addressed by the applicants in plant-by-plant license renewal proceedings. To accomplish these objectives, the GEIS makes maximum use of environmental and safety documentation from original licensing proceedings and information from state and federal regulatory agencies, the nuclear utility industry, the open literature, and professional contacts. This document, Volume 2, provides the appendices to this report.

1805. WHITE PAPER ON NUCLEAR SAFETY IN 1990. Nuclear Safety Commission, Tokyo (Japan), Published by the Ministry of Finance, Printing Bureau, Tokyo (Japan), 5 March 1991, in Japanese, (434 p).

The ensuring of safety is the most important prerequisite for the development and utilization of atomic energy. The Nuclear Safety Commission has promoted the various important measures which become the basis for ensuring safety, such as the execution of strict safety examination, the perfection of the guidelines used for safety examination, the holding of public hearings to consider the opinion of local people, the investigation of failures and troubles, the reflection of the teachings obtained as the results to safety measures and so on. As the results, the safety of Japanese nuclear facilities has maintained high level and reliability internationally. In the white paper on nuclear safety in this year, the activities of the Nuclear Safety Commission and the present status of the measures related to the ensuring of safety in general nuclear facilities are reported. The way of thinking on the safety of atomic energy, which becomes the base of the measures for ensuring safety, is explained. The deterministic safety evaluation currently carried out in Japan and the study on probabilistic safety evaluation are described. The safety measures for radioactive wastes, the protection from the effect of radiation and so on are reported.

1806. EXPERIENCES IN LIMITING RADIATION EXPOSURE TO THE EMBRYO/FETUS IN NUCLEAR POWER PLANTS. KELLY, J.J. (New York Power Authority, White Plains, NY). Implementation of Current NCRP and ICRP Guidance and Revised 10 CFR Part 20: Proceedings. Edited by

JORGENSEN, D.B.; SEAGONDOLLAR, L.W.; WATSON, J.E. JR. 24th Midyear Topical Meeting of the Health Physics Society, 20-25 January 1991, CONF-910137--, 1991, pp. 218-231.

This paper presents the results of a survey of operating nuclear reactors and nuclear fuel fabrication facilities in the U.S. The survey obtained information on the number of women radiation workers in those plants over the last ten years and the number of workers potentially exposed to radiation while pregnant. Information on plant exposure limits for pregnant workers practiced at these plants and whether these limits comply with NCRP guidance and proposed NRC regulatory limits will also be presented. The discussion will include the effects of unions, labor arbitration, and legal actions on these policies. The unique problems of fuel manufacturers in addressing the proposed NRC regulations for embryo/fetus exposure will also be presented.

1807. OCCUPATIONAL RISK AND LIFETIME EXPOSURE. LAPP, R.E. (Lapp, Inc., Alexandria, VA). Implementation of Current NCRP and ICRP Guidance and Revised 10 CFR Part 20: Proceedings. Edited by JORGENSEN, D.B.; SEAGONDOLLAR, L.W.; WATSON, J.E. JR. 24th Midyear Topical Meeting of the Health Physics Society, 20-25 January 1991, CONF-910137--, 1991, pp. 204-212.

Any lowering of annual radiation limits for occupational exposure should be based on industry experience with lifetime doses and not on a worst case career exposure of 47 years. Two decades of experience show a lifetime accumulation of less than 1.5 rem for workers with measurable exposure. This is 5% of the normal lifetime exposure of Americans to natural and medical radiation. Any epidemiology of the U.S. nuclear power workforce's two-decade-long exposure would have to focus on excess leukemia. Application of the Hiroshima and Nagasaki cancer mortality shows that too few leukemias would be expressed to permit a feasible epidemiology. Ionizing radiation appears to be a mild carcinogen as compared to physical and chemical agents presented in the occupational environment. A realistic factor in determining any change in occupational exposure limits for ionizing radiation should take into account the past performance of the licensee and potential health effects applicable to the workplace. Specifically, the lifetime exposure data for workers at nuclear power plants and naval shipyards should be considered. The nuclear industry and the U.S. Navy have detailed data on the annual exposure of workers

with a combined collective exposure approaching 1 million worker-rem. The lifetime dose for naval personnel and shipyard workers averages 1.1 rem J 1990. Shipyard workers have an annual dose of 0.28 rem per work-year and a mean exposure time of 4.4 years. The data apply to workers with measurable dose.

1808. ALARA AT NUCLEAR POWER PLANTS. BAUM, J.W. (Brookhaven National Laboratory, Upton, NY). Implementation of Current NCRP and ICRP Guidance and Revised 10 CFR Part 20: Proceedings. Edited by JORGENSEN, D.B.; SEAGONDOLLAR, L.W.; WATSON, J.E. JR., 24th Midyear Topical Meeting of the Health Physics Society, 20-25 January 1991, CONF-910137--, 1991, pp. 165-181.

Implementation of the ALARA principle at nuclear power plants presents a continuing challenge for health physicists at utility corporate and plant levels, for plant designers, and for regulatory agencies. The relatively large collective doses at some plants are being addressed through a variety of dose reduction techniques. Initiatives by the ICRP, NCRP, NRC, INPO, EPRI, and BNL ALARA Center have all contributed to a heightened interest and emphasis on dose reduction. The NCRP has formed Scientific Committee 46-9 which is developing a report on ALARA at Nuclear Power Plants. It is planned that this report will include material on historical aspects, management, valuation of dose reduction (\$/person-Sv), quantitative and qualitative aspects of optimization, design, operational considerations, and training. The status of this work is summarized in this report.

1809. EXPOSURE MANAGEMENT - INPO PERSPECTIVE. JACOBS, R.H. (Institute of Nuclear Power Operations, Atlanta, GA). Implementation of Current NCRP and ICRP Guidance and Revised 10 CFR Part 20: Proceedings. Edited by JORGENSEN, D.B.; SEAGONDOLLAR, L.W.; WATSON, J.E. JR., 24th Midyear Topical Meeting of the Health Physics Society, 20-25 January 1991, CONF-910137--, 1991, p. 163.

The slide presentation provides an overview of INPO's mission and programs. The focus of the presentation is on the nuclear power industry occupational radiation exposure trends, current activities, and the future outlook for controlling radiation exposure. Discussion will include progress the industry has made over the last 10 years, unplanned radiation exposures and observations of the root cause of these

incidents, precursors to high exposure incidents, and INPO's continuing role in radiation exposure management.

1810. RECOMMENDATIONS, REQUIREMENTS, AND RADIOACTIVE PARTICLES. BELL, J.M. (Nuclear Regulatory Commission, Washington, DC). Implementation of Current NCRP and ICRP Guidance and Revised 10 CFR Part 20: Proceedings. Edited by JORGENSEN, D.B.; SEAGONDOLLAR, L.W.; WATSON, J.E. JR., 24th Midyear Topical Meeting of the Health Physics Society, 20-25 January 1991, CONF-910137--, 1991, pp. 140-152.

Hot particles consisting of activated metal debris and fuel fragments have received increased attention in the last five years. This increased attention resulted from the increased use of more sensitive whole body friskers at nuclear power plants, the relatively high local skin doses associated with the particles, and skin dose limits that were established before hot particles, and skin dose limits that were established before hot particles became a problem and before radiobiological effects data for the particles became available. The skin dose distribution and biological effects associated with hot particles differ from those associated with more uniform skin contamination and differences exist in the scientific community as to which effects should be protected against by a limit on exposures from particles. The NRC staff recognized the need for provisions in the Federal regulations specific to hot particle exposures and requested guidance from the National Council on Radiation Protection and Measurements (NCRP). NCRP Report No. 106 was provided to the NRC early in 1990. The International Commission on Radiological Protection (ICRP) is also developing recommendations for limits on exposures from hot particles. The NRC is supporting research on hot particle effects and will likely develop a rule for hot particle exposures.

1811. PRACTICAL METHODS FOR EXPOSURE CONTROL/MANAGEMENT AT NUCLEAR FACILITIES. TWIGGS, J.A. (Duke Power Co., Charlotte, NC). Implementation of Current NCRP and ICRP Guidance and Revised 10 CFR Part 20: Proceedings. Edited by JORGENSEN, D.B.; SEAGONDOLLAR, L.W.; WATSON, J.E., JR. 24th Midyear Topical Meeting of the Health Physics Society, 20-25 January 1991, CONF-910137--, 1991, p. 118.

Exposure management/reduction is very important to Duke Power Company. Practical exposure con-

trol/reduction techniques applied to their reactor vessel head disassembly outage activity have consistently reduced personnel exposure for this task. The following exposure control methods have worked for use and will be the industry's direction for the 1990s. A summary of these methods includes: (a) move the responsibility of exposure management from the Radiation Protection group to the Maintenance group; (b) reduce area source term by removal of source; (c) improve working environments in radiation areas by minimizing protective clothing usage; and (d) maximize the use of electronic instruments to allow remote monitoring.

1812. NUCLEAR POWER INDUSTRY REVIEW OF 10 CFR 20 IMPLEMENTATION DOCUMENTS. SCHMITT, J.F.; MAISLER, J.J., Implementation of Current NCRP and ICRP Guidance and Revised 10 CFR Part 20: Proceedings. Edited by JORGENSEN, D.B.; SEAGONDOLLAR, L.W.; WATSON, J.E. JR., 24th Midyear Topical Meeting of the Health Physics Society, 20-25 January 1991, CONF-910137--, 1991, pp. 34-37.

The revision of 10 CFR 20, Standard for Protection Against Radiation will result in changes to radiation protection programs at commercial nuclear power facilities. The record of the nuclear power industry for radiation protection of public and worker health and safety is very good and continues to improve. Because of its commitment, the industry is concerned that implementation of the changes required by the new regulation be accomplished in a correct, effective and efficient manner which continues the improvements in radiation protection. Development and review of implementation documents (e.g., Regulatory Guides) associated with the revised Part 20 is an important part of this effort. This paper discusses activities performed and plans developed by NUMARC, the nuclear power industry's organization responsible for generic regulatory matters, regarding implementation documents associated with the revised 10 CFR 20.

1813. MEASUREMENT OF HTO PERMEATION RATES THROUGH VARIOUS CANDIDATE DISPOSABLE PLASTIC-SUIT MATERIALS. REPORT NO. 90-232-K. KRASZNAI, J.P.; AUCOIN, B.D., (Ontario Hydro, Research Division, Toronto, Ontario (Canada)). MIC-91-03163, 1990, (29 p).

This study investigated the protection factor for tritiated water exposure afforded by various disposable plastic suit materials. The suits currently in use

in Ontario Hydro generating stations are made of polyvinyl chloride (PVC). The study researched alternative materials to PVC which would not only meet or exceed the tritium protection capability of PVC but be compatible with incineration as a disposal option. Results of simple permeation tests using tritiated water to identify candidate materials are given here.

1814. AECB STAFF REVIEW OF BRUCE NGS A OPERATION FOR THE YEAR 1989. REPORT NO. INFO-0367. ANNUAL PUBLICATION. Atomic Energy Control Board, Ottawa, Ontario (Canada), MIC-91-03349, 1990, (30 p).

This report presents AECB project staff's review of major licensing issues and operational performance of the station during 1989, using Ontario Hydro reports and official correspondence and observations of AECB staff at the site. The report is limited to aspects of the station performance that AECB staff considers to have safety significance. The review covers compliance, maintenance, testing, radiation protection, effluents and environmental monitoring, reactor process, control and fuel handling systems, performance of special safety and secondary systems, shutdown systems, emergency cooling injection, negative pressure containment, significant events, quality assurance, chemistry control, chemistry, training, emergency drills, maintenance, management and security. Significant licensing matters include primary heat transport pump trips, containment modifications, pressure tube integrity, and standards of operation and maintenance. A list of significant events is also given.

1815. THE ALARA CENTER AND ITS INFORMATION SERVICE--ACE. KHAN, T.A.; BAUM, J.W., (Brookhaven National Laboratory, Upton, NY) Annual Meeting of the Boiling Water Reactors (BWR) Owner's Group, 1-2 August 1991, BNL-NUREG-46542, CONF-9108157--1, August 1991, (34 p).

In compliance with its congressional mandate to oversee the radiation safety of workers at nuclear power plants, the NRC asked Brookhaven National Laboratory (BNL) to create a center to help monitor efforts that were likely to reduce occupational radiation exposure. The NRC project required the ALARA Center to evaluate dose-reduction research and the ALARA-related programs, and to note any areas where additional effort may be fruitful. The Center also was directed to inform the NRC of promising research and developments related to

ALARA that were being carried out internationally, and to examine areas where international collaboration may be valuable. This document discusses the objectives of the ALARA Center.

1816. MANAGING NUCLEAR MAINTENANCE. Electrical World, Volume 204, Number 2, February 1990, pp. 43-45 and p. 48.

For utilities operating nuclear powerplants, the rules of the game continue to change. Conflicting regulatory pressures and a tougher competitive environment will make management's job more complicated and difficult in the 1990s. Dealing with these pressures successfully requires greater attention to maintenance effectiveness. Utilities can help shape their future environment by developing a well-planned strategy to guide their actions. Parts of the strategy that are discussed include developing a sound maintenance philosophy, selecting a service company, radiation exposure, and managing spare parts. This article also addresses the Swedish experience in maintenance, German philosophy regarding maintenance and the current maintenance practices of the Électricité de France.

1817. TREATMENT AND DISPOSAL OF RADIOACTIVE WASTES AND COUNTERMEASURES. NOMURA, KIYOSHI (Tokyo University (Japan)). Anzen Kogaku (Journal of Japan Society for Safety Engineering) (Japan), Volume 29, Number 6, 1990, in Japanese, pp. 459-467.

The treatment and disposal of radioactive wastes are important subjects, together with the development of dismantling techniques accompanying the decommissioning measures for nuclear power plants and the development of reprocessing techniques for nuclear fuel cycle. About 25 years have elapsed since the beginning of commercial nuclear power generation in 1966, and the time has come when the solution of the problems of waste treatment and disposal must be tackled on full scale. The features and the amount of generation of radioactive wastes, the way of thinking on the treatment and disposal, and the present status of the treatment and disposal are outlined. For securing the stable supply of energy and solving the environmental problem of the earth such as acid rain and warming, nuclear power generation accomplishes important roles. The objective of waste treatment is based on the way of thinking of "as low as reasonably achievable" (ALARA). The radioactive wastes are classified into alpha waste and beta-gamma waste. The present status of RI wastes, the techniques of treating radioactive wastes, the nuclide separation,

extinction treatment, and the disposal in strata of high level radioactive wastes, and the disposal of low level wastes, are reported.

1818. RADIATION PROTECTION AND REACTOR SAFETY. Bundesministerium fuer Umwelt, Naturschutz und Reaktorsicherheit, Bonn (Germany), INIS-mf-14009, March 1990, in German, (152 p).

The Chernobyl reactor accident caused bewilderment, fear, and anxiety among the population. How safe are reactors? Which precautions to protect lives and health have been taken? These questions are posed particularly in the areas of radiation protection, reactor safety, supply and waste management of nuclear power plants and other nuclear installations. For all these areas the present report contains an analysis of facts; it informs about political measures during the 11th legislative period of the German Bundestag, and shows prospects of future developments.

1819. DEVELOPMENT OF AN ADVANCED RADIATION WORKER LABORATORY. HOFFMAN, M.A.; KLETT, L.D. (Toledo Edison Co., OH). Proceedings of the Ninth Symposium on the Training of Nuclear Facility Personnel. Oak Ridge National Laboratory, TN, 14-18 April 1991, CONF-9104135--, April 1991, pp. V.A. 4.1-V.A. 4-15.

Laboratory training is a proven method of providing and evaluating hands on skills and knowledge. In an attempt to maximize the effectiveness of radiation worker training of craft personnel, a program consisting of classroom training coupled with a novel laboratory mock-up was generated utilizing the ingenuity of technical skills trainers while minimizing cost. Program development and the results of trainee feedback are presented.

1820. INNOVATIVE TECHNIQUES IN MAINTENANCE TRAINING. SOILEAU, J.B.; BLACKWELL C.; RACKOS, N.; ELMER, L.B., Proceedings of the Ninth Symposium on the Training of Nuclear Facility Personnel. Oak Ridge National Laboratory, TN, 14-18 April 1991, CONF-9104135--, April 1991, pp. V.A. 2.1-V.A. 2.9.

Performance based training has its beginning with a Job and Task Analysis (JTA), and culminates with the presentation of the developed training material. For optimum training effectiveness, a post training feedback mechanism is utilized to update and/or upgrade material content. A Job Task Analysis uses subject matter experts and supporting documentation to define

the skills and knowledge necessary to perform functions from total position responsibilities to the simplest tasks, depending on desired results. Once skills and knowledge are defined, decisions on needed curriculum can be made including objectives, exams, and training settings (classroom, lab, or job site). This focused curriculum and the determination of best training settings are innovations of the performance based training system. Past training experience has illustrated that a baseline level of classroom training acts as a catalyst for optimum hands-on training. The trainee exits the training phase able to do the job proficiently. This provides the training customer with quality training with minimum investment. Maintenance Training at Comanche Peak Steam Electric Station (CPSES) integrates the knowledge and skill needs by providing the individual with focused classroom presentations followed by sufficient laboratory time to practice actual task activities. Labs are configured to represent plant equipment and work environment to the maximum extent possible and duplicate in-plant work orders and procedures are used. This allows actions to be analyzed and corrective feedback given as necessary. As an added benefit, any mistakes made during training has minimal, if any, impact on plant performance.

1821. SATISFYING THE DIVERSE DEVELOPMENT NEEDS OF AN ENGINEERING ORGANIZATION. ZARKESH, L.P. (Toledo Edison Co., OH). Proceedings of the Ninth Symposium on the Training of Nuclear Facility Personnel. Oak Ridge National Laboratory, TN, 14-18 April 1991, CONF-9104135--, April 1991, pp. II.A.2.1-II.A.2.20.

The Engineering Department at Davis-Besse Nuclear Power Station has established an aggressive philosophy for professionally developing their staff. This philosophy has evolved over the last four years into a program with specialized administrative tools which not only satisfies the intent of industry training guidelines, but also accentuates the development of the individual. This program consists of three parts: (1) The Development Program - A program constructed to actively integrate system and applied science courses, management and interpersonal skill courses, design basis courses (e.g., pipe break analysis, support design, etc.) special process courses (e.g., human factors, ALARA, etc.) and external seminars sponsored by industry experts; (2) The Individual Development Plan (IDP) - A documented course of action, developed annually, in which the employee and the line supervisor jointly contribute to the identification of career goals and strategic professional objectives;

and (3) The Training Database - A PC database developed to retain and manage course information (e.g., requests, attendance priorities, schedule, history, etc.). The paper describes these three facets of the training program.

1822. SUPERVISORY SELECTION AND DEVELOPMENT YIELDS PAYBACK. RUBINO, E.A. (Philadelphia Electric Co., PA). Proceedings of the Ninth Symposium on the Training of Nuclear Facility Personnel. Oak Ridge National Laboratory, TN, 14-18 April 1991, CONF-9104135--, April 1991, pp. VI.A.4.1-VI.A.4.6.

The selection and development of first-line supervisors is a critical factor in a strategic effort to bring about organizational change, to improve performance and productivity and to enhance professionalism in the work force. Candidates for first-line supervisory positions at Philadelphia Electric Company are no longer promoted purely on the basis of seniority. A Supervisory Selection Assessment Center permits candidates to demonstrate leadership potential. Successful candidates then attend an eight-week Supervisory Development Academy, away from the work place. On successful completion of the Academy, the supervisor assembles his or her work team and, with training and leadership, shapes a performance-based work team ready to plan, execute, and assess its own work. Since this radically different program was initiated in September 1989, results have been highly acclaimed by management. Contractor support has been sharply reduced, budgets have been met and reduced more work has been accomplished and fewer overtime hours have been needed. Workers attend training with no loss of team productivity. Each SDA graduate and his team sets goals to improve quality, increase productivity, increase efficiency, maintain ALARA, improve safety and effect cultural change while improving job satisfaction and morale.

1823. TEST PROGRAM FOR SHIELDING IN NUCLEAR POWER PLANTS. DIN Deutsches Institut fuer Normung e.V., Berlin (Germany). Normenausschuss Kerntechnik, Published by Beuth, DIN--25453, July 1991, in German, (9 p).

The standard applies to shielding installed for personnel protection in nuclear power plants, either permanently or for routine use during start-up and normal operation. It intends to set up a test program for shielding in order to assure their being up to the standard values set in planning. In the test program, data will be acquired in such manner that the local

dose rates measured can be compared with the standard values set in planning and conclusions can be drawn concerning deviations of the local dose rate from these standard values.

1824. FUNCTIONS OF AUTOMATED PRODUCTION CONTROL SYSTEM AT TEMELIN NUCLEAR POWER PLANT. MITOSINKA, J.; KOREC, J. (Vyskumny Ustav Jadrovych Elektram, Trnava (Czechoslovakia)). Collection of Papers Presented at Conference on Application of Computers in Nuclear Power Plants, 15-18 October 1990, INIS-mf-12967, CONF-9010419--, 1990, in Slovak, pp. 82-101.

The automated control system for the Temelin nuclear power plant is being built up based on the principal requirements as follows: possibility of modular expansion, compatibility of the equipment, possibility of operation in the local as well as public network, and possibility of employing computer structures with different configurations. The functions of the control system are described in detail; these concern the technological process control level, the whole-plant production control level, the radiological safety field, control of the nuclear power plant chemistry, control and evaluation of the power plant production, and management of the basic equipment.

1825. CLASSIFICATION OF ROOMS IN THE CONTROLLED AREA OF NUCLEAR POWER PLANTS ACCORDING TO LOCAL DOSE RATES. DIN Deutsches Institute fuer Normung e.V., Berlin (Germany). Normenausschuss Kerntechnik, Published by Beuth, DIN--25440, July 1991, in German, (4 p).

The purpose of the standard is to put into a representative and clearly arranged form the local dose rates taken as a basis for the planning of the rooms of the controlled area in nuclear power plants. This is supposed to facilitate communication between the authorities and institutions taking part in the planning, licensing and building of nuclear power plants.

1826. STUDY OF ACTIVATION OF BINDING MATERIALS FOR RADIATION SHIELDING CONCRETES OF REACTOR FACILITIES. GUNDORINA, S.F. (Joint Institute for Nuclear Research, Lab of Neutron Physics, Dubna (USSR)); BABOSHIN, N.G.; ENGOVATOV, I.A. (Moskovskij Inzhenerno-Stroitel'NYJ Inst., Moscow (USSR)). Joint Institute for Nuclear Research, Laboratory of Neutron Physics, Dubna (USSR). JINR-18-90-182, 1990, in Russian, (18 p).

The results of activation studies of concrete binding agent ingredients for shielding structures reactor facilities in the aspect of their decommissioning are given. It is shown that for the long-lived induced radioactivity of structural mineral materials irradiated for 30 years and cooled by more than one year such radionuclides as calcium, iron, cobalt, cesium and europium are responsible. Elemental content of the binding agent and raw material components for their production is obtained with neutron activation analysis. The results show that the type of the binding agent influences to a great extent the concrete shielding activity. The concentration of the above mentioned elements should be taken into account even at the stage of nuclear power plants design. It would allow one to make prognosis on the volume and radioactivity of wastes as on the radioactivity effect suffered by the stuff engaged in the decommissioning.

1827. SHIELDING DESIGN FOR SPENT-FUEL SHIPPING IN HIGH-CAPACITY TRUCK CASKS. BOSHoven, J.K.; SU, SHIAWDER (General Atomics, San Diego, CA). Winter Meeting of the American Nuclear Society (ANS) and Nuclear Power and Technology Exhibit, 26-30 November 1989, Transactions of the American Nuclear Society, Volume 60, CONF-891103--, November 1989, pp. 525-526.

General Atomics (GA), under contract to the U.S. Department of Energy (DOE), is developing a high-capacity legal-weight truck transportation system for shipment of pressurized water reactor (PWR) and boiling water reactor (BWR) spent fuels from commercial nuclear power plants to a repository or monitored retrievable storage facility. The system includes two cask versions optimized for PWR and BWR fuels, respectively. The PWR version carries four fuel assemblies designated the GA-4 cask, and the BWR version accommodates nine fuel assemblies designated the GA-9 cask. This paper presents the shielding aspects of the GA-4 and GA-9 casks.

1828. CONTROL ROOM HABITABILITY DURING SEVERE ACCIDENTS. SIU, R.P. (United Engineers and Constructors Inc., Philadelphia, PA). Winter Meeting of the American Nuclear Society (ANS) and Nuclear Power and Technology Exhibit, 26-30 November 1989, Transactions of the American Nuclear Society, Volume 60, CONF-891103--, November 1989, pp. 541-542.

The requirements for protection of control room personnel against radiation hazards are specified in

10CFR50, Appendix A, GDC 19. The conventional approach involves a mechanistic evaluation of the radiation doses to control room personnel during design-basis accidents. In this study, an assessment of control room habitability during severe accidents is conducted. The potential levels of radiation hazards to control room personnel are evaluated in terms of both magnitude and probability of occurrence. The expected values for the probabilities of exceeding GDC-19 limits and the cumulative probability distributions of control room doses are determined. In this study, a pressurized water reactor with a large dry containment has been selected for analysis. The types of control rooms evaluated in this study include designs with: (a) filtered local intakes only, (b) filtered recirculation only, (c) filtered local intakes and recirculation, and (d) filtered dual remote intakes and recirculation. From the observations, it is concluded that, except for control room D, all other control room designs may require improvements in order to provide adequate radiation protection during severe accidents, particularly in terms of reducing whole-body gamma doses and skin doses. Potential design improvements include reduction of intake flows for concepts relying on pressurization, reduction in overall leakages, and control room pressurization through the use of bottled air supply.

1829. MEASUREMENT OF HOT-PARTICLE DOSE PROFILES USING EXOELECTRON DOSIMETERS. MOELLER, M.P.; MERWIN, S.E. (Battelle-Pacific Northwest Laboratories, Richland, WA). Winter Meeting of the American Nuclear Society (ANS) and Nuclear Power and Technology Exhibit, 26-30 November 1989, Transactions of the American Nuclear Society, Volume 60, CONF-891103--, November 1989, pp. 520-521.

Since 1985, staff at Battelle's Pacific Northwest Laboratories (PNL) have analyzed hot-particle contaminants and assisted in the evaluation of hot-particle exposure events at nuclear power plants. More recently, staff have performed an assessment of the thresholds for nonstochastic skin effects caused by hot particles of different sizes and beta energies. The purpose of this paper is to describe methods used for measuring hot-particle dose rates and methods for characterizing dose profiles from particles. The use of exoelectron dosimeters to quantify depth and lateral dose profiles represents a significant improvement in the characterization of radioactive sources, especially for hot particles. Researchers would like to develop a definitive model to relate the dose profiles from hot particles to the induction of nonstochastic skin effects.

This objective is now made possible through development of an accurate measurement technique to confirm theoretical calculations. Exoelectron dosimeters can be used to perform such measurements as well as dose rate measurements to 1 cm² of skin for regulatory purposes.

1830. EVALUATIONS FOR THE USE OF TEMPORARY LEAD SHIELDING ON PIPING. OLSON, D.E. (Sargent & Lundy, Chicago, IL). Winter Meeting of the American Nuclear Society (ANS) and Nuclear Power and Technology Exhibit, 26-30 November 1989, Transactions of the American Nuclear Society, Volume 60, CONF-891103--, November 1989, pp. 540-541.

The use of temporary lead shielding on piping has proven to be an effective way of reducing person-rem exposure. Engineering evaluations addressing the effect of the shielding's weight load and dynamic response load on the piping and support hardware must precede the application of shielding. Developing effective procedures and software for evaluating the effect of shielding will result in increased and effective shielding use and will help ensure compliance with U.S. Nuclear Regulatory Commission (NRC) requirements and the Institute of Nuclear Power Operations (INPO) recommendations for technical evaluations and tracing of shielding use.

1831. AECB STAFF ANNUAL REPORT ON POINT LEPREAU G.S. FOR THE YEAR 1989. REPORT NO. INFO-0365. ANNUAL PUBLICATION. Atomic Energy Control Board, Ottawa, Ontario (Canada), MIC-91-03350, 1990, (16 p).

Review by AECB staff of the operation of Point Lepreau Nuclear Generating Station during 1989. The review is based on information contained in the documents submitted by New Brunswick Power under the terms of the Operating License and on information gathered by AECB staff during routine site monitoring, inspections and audits. The review covers compliance, radiation protection, effluents and environmental monitoring, process systems, performance of special safety systems, significant events, quality assurance, maintenance, chemistry control, management, training, emergency exercises and drills, security and measures of performance. Significant licensing matters and activities include license renewal, containment leak rate testing, follow-up to the Chernobyl accident, and steam feedwater failures in the turbine building.

1832. CONTROL OF HOT PARTICLES CHALLENGES GPU NUCLEAR AND THE NUCLEAR POWER INDUSTRY. LODDE, G.M. (GPU Nuclear Corporation, Middletown, PA). Winter Meeting of the American Nuclear Society (ANS) and Nuclear Power and Technology Exhibit, 26-30 November 1989, Transactions of the American Nuclear Society, Volume 60, CONF-891103--, November 1989, pp. 517-518.

During the past four years, a number of nuclear generating stations, including GPU Nuclear, have reported occurrences of hot particles that are small and, in most cases, microscopic particles of radioactive contamination with relatively high specific activity. A survey conducted by Electric Power Research Institute (EPRI) indicated that approximately 70% of the nuclear power plants indicated that they have some problem with hot particles. Both pressurized water reactors (PWRs) and boiling water reactors (BWRs) have found hot-particle contamination in the plant environment and, occasionally, on radiation workers. A hot particle on the skin produces a very steep dose gradient; the dose drops off very rapidly as the distance from the particle increases. The local absorbed dose produced from a hot particle on the skin may exceed the administrative limit established by the utility and on occasion exceeds the regulatory limit, which results in an overexposure reportable to the U.S. Nuclear Regulatory Commission (NRC). In an effort to contain these particles nuclear plant operators and management must be committed to the control of hot particles. The nuclear plant operators and the Three Mile Island Unit 2 (TMI-2) cleanup team are implementing new procedures, training programs, and more rigid contamination control techniques in order to control hot particles in the work environment. This paper briefly discusses the nuclear industries' experience, GPU Nuclear's experience, hot-particle detection, dosimetry, control measures, and costs associated with GPU Nuclear's hot-particle-control program.

1833. ENFORCEMENT ACTIONS: SIGNIFICANT ACTIONS RESOLVED. Nuclear Regulatory Commission, Office of Enforcement, Washington, DC, NUREG-0940, Volume 10, Number 2, July 1991, (387 p).

This compilation summarizes significant enforcement actions that have been resolved during one quarterly period (April-June 1991) and includes copies of letters, Notices, and Orders sent by the U.S. Nuclear Regulatory Commission to licensees with respect to

these enforcement actions. It is anticipated that the information in this publication will be widely disseminated to managers and employees engaged in activities licensed by the NRC, so that actions can be taken to improve safety by avoiding future violations similar to those described in this publication.

1834. NRC REGULATORY AGENDA. Nuclear Regulatory Commission, Div. of Freedom of Information and Publications Service, Washington, DC, NUREG-0936, Volume 10, Number 2, August 1991, (132 p).

The NRC Regulatory Agenda is a compilation of all rules on which the NRC has recently completed action or has proposed, or is considering action and all petitions for rulemaking which have been received by the commission and are pending disposition by the Commission. The Regulatory Agenda is updated and issued each quarter.

1835. ULTRASONICS FOR FLUID-LEVEL MONITORING IN NUCLEAR POWER PLANT PIPES. JOSSINET, G. (Électricité de France, Saint-Denis (France)); DAVID, J. (Électricité de France, Noisy le Grand (France)); STEELE, C. (Duke Power Company, Cornelius, NC); LEON, J. (Merlin Gerin Provence, Smyrna, GA). Winter Meeting of the American Nuclear Society (ANS) and Nuclear Power and Technology Exhibit, 26-30 November 1989, Transactions of the American Nuclear Society, Volume 60, CONF-891103--, November 1989, pp. 563-565.

Ultrasonic techniques are of great interest for non-invasive physical measurements of fluids in hostile environments. Électricité de France (EdF) has developed several products in this area to monitor sensitive parameters in nuclear plants. As an example, we present a system to monitor the water level of the primary coolant circuit during a refueling outage. Pressurized water reactor refueling requires a plant outage during which the steam generator tubes must be drained. Primary circuit pipes are drained for this purpose. Nevertheless, it is necessary to maintain a minimum level of fluid in the system to guarantee proper operation of the residual heat removal (RHR) system. Damage to the RHR pumps because of drainage of the suction port must be avoided. Fine and accurate monitoring of this operation is difficult using conventional differential pressure level transmitters. Due to several incidents in French nuclear plants regarding RHR operations, the EdF maintenance department (SPT/UTO), in 1988, asked the EdF research and development department (DER) to study

and develop a new method of monitoring this critical parameter. The method and results are described.

1836. BELOW REGULATORY CONCERN: CRCPD (CONFERENCE OF RADIATION CONTROL PROGRAM DIRECTORS, INC.) MCBURNEY, R.E. (Texas Bureau Rad Control, Austin, TX). Winter Meeting of the American Nuclear Society (ANS) and Nuclear Power and Technology Exhibit, 26-30 November 1989, Transactions of the American Nuclear Society, Volume 60, CONF-891103--, November 1989, pp. 508-509.

The Conference of Radiation Control Program Directors, Inc. (CRCPD) represents, collectively, agencies that regulate most of the radioactive material in the United States, excluding nuclear weapons production and nuclear power plants. As such, this organization is constantly aware of the need for effective regulation of not only radioactive materials but of all sources of radiation. Likewise, exemption of products and practices whose future regulation would yield little gain in risk reduction is also needed from a regulatory standpoint. Definition of that cut-off is an essential part of setting regulatory standards. The CRCPD has found the development of national standards in this area to be very important to our organization. If a national policy on exemptions from regulatory control is to be effective, state and local radiation control programs will play an important implementation role. The NRC (Nuclear Regulatory Commission) intends that rules following the development of a BRC (below regulatory concern) policy for exemptions will be a matter of compatibility for agreement states. Additionally, all radiation control programs could be potentially impacted by the application of the policy to naturally occurring and accelerator-produced materials. Therefore, the CRCPD has developed a consensus position on BRC and intends to provide coordinated input during the development of the policy and subsequent rulemaking.

1837. APPLICATIONS FOR BURNING CONTAMINATED OIL AT OFF-SITE FOSSIL STATIONS. TERRELL, G.F.; BRYANT, J.W. (Duke Power Company, Charlotte, NC). Winter Meeting of the American Nuclear Society (ANS) and Nuclear Power and Technology Exhibit, 26-30 November 1989, Transactions of the American Nuclear Society, Volume 60, CONF-891103--, November 1989, pp. 510-511.

Duke Power Company is actively pursuing a course to allow the burning of slightly contaminated waste oil from its nuclear power plants at two of its fossil

stations. Duke Power Company operates three nuclear power plants in North and South Carolina. At each of these sites, waste oils containing very low levels of radioactive material are generated. This oil currently falls into two basic categories: hazardous petroleum-based and nonhazardous synthetic oil. This oil is unacceptable in any form at the Barnwell low-level-waste (LLW) facility and must be solidified to be buried at either the Beatty, Nevada, or Richland, Washington, facilities. Since 1984, Duke Power had handled its contaminated waste oil in three ways: storage on-site, solidification and burial at a LLW facility, and incineration in an on-site auxiliary boiler at Oconee nuclear station. For a number of reasons, these options do not appear to be acceptable as long-term solutions to the ever-increasing problem. Prior to submitting an application to the involved states to burn this oil, an agreement had to be reached among the various departments within Duke Power on the acceptability of this course of action. By making some extremely conservative calculations and using some specific data for the fossil unit, we then calculated some worst-case activities that could be expected from burning slightly contaminated oil at the specified fossil unit. These worst-case activities were then used to determine the annual dose rate estimates for permit applications.

1838. GETTING EXPOSURES DOWN AT U.S. PLANTS. OCCUPATIONAL RADIATION EXPOSURE AT NUCLEAR PLANTS. WOOD, C.J. (Electric Power Research Inst., Palo Alto, CA). Nuclear Engineering International (United Kingdom), Volume 36, Number 442, May 1991, pp. 16-18.

For some years, U.S. utilities have been committed to maintaining radiation exposures to workers at nuclear plants as low as reasonable achievable but have recorded much higher exposures than other countries with large LWR programs such as France, Germany and Japan. As a result U.S. utilities have been aggressively reducing individual and cumulative exposures and average occupational exposures since the early 1980s. In 1989 there was a drop from 345 to 290 man-rem per plant at PWRs and from 510 to 440 man-rem per plant at BWRs. Preliminary data for 1990 suggest exposures similar to 1989.

1839. RECENT DECREASING TREND OF PERSONNEL EXPOSURE IN NUCLEAR POWER PLANTS IN JAPAN AND ANALYSIS OF MAJOR FACTORS CONTRIBUTING TO ITS REDUCTION. SASAKI, FUMIAKI (Japan Power Engineering and Inspection Corp., Tokyo (Japan)).

Karyoku Genshiryoku Hatsuden (Thermal and Nuclear Power) (Japan), Volume 42, Number 2, February 1991, in Japanese, pp. 157-162.

The radiation exposure of workers at nuclear power plants in Japan has been decreasing steadily in both total and per capita average although the total number of such plants is increasing. The total and the average exposure greatly decreased from 131.27 person-sievert and 3.8 millisievert in 1978, which were largest in history, down to 87.39 person-sievert and 1.6 millisievert, respectively, in 1989. The present article first outlines statistics of radiation exposure of workers at major facilities throughout Japan. Then a breakdown by type of nuclear reactor used in the facilities (boiling water, pressurized water) is described and discussed. A separate section is devoted to statistics for power plants with boiling water reactors. As a typical example, various improvements made to decrease the radiation exposure at the Fukushima No. 1 Nuclear Power Plant, where six reactors have been installed, are described. These improvements include the introduction of high-performance shielding material for the containment vessels, efficient cleaning and decontaminating equipment and filters to prevent iron rust from entering the reactors.

1840. A PRIORITIZATION OF GENERIC SAFETY ISSUES. EMRIT, R.; RIGGS, R.; MILSTEAD, W.; PITTMAN, J., Nuclear Regulatory Commission, Div. of Safety Issue Resolution, Washington, DC, NUREG-0933, July 1991, (1506 p).

This report presents the priority rankings for generic safety issues and related to nuclear power plants. The purpose of these rankings is to assist in the timely and efficient allocation of NRC resources for the resolution of those safety issues that have a significant potential for reducing risk. The report focuses on the prioritization of generic safety issues. Issues primarily concerned with the licensing process or environmental protection and not directly related to safety have been excluded from prioritization. The prioritized issues include: TMI Action Plan items under development; previously proposed issues covered by Task Action Plans, except issues designated as Un-resolved Safety Issues (USIs) which had already been assigned high priority; and newly-proposed issues. Future supplements to this report will include the prioritization of additional issues. The safety priority rankings are High, Medium, Low, and Drop and have been assigned on the basis of risk significance estimates, the ratio of risk to

costs and other impacts estimated to result if resolutions of the safety issues were implemented, and the consideration of uncertainties and other quantitative or qualitative factors. To the extent practical, estimates are quantitative.

1841. AN APPROACH TOWARD ESTIMATING THE SAFETY SIGNIFICANCE OF NORMAL AND ABNORMAL OPERATING PROCEDURES IN NUCLEAR POWER PLANTS. GRANT, T.F. (Battelle Human Affairs Research Centers, Seattle, WA); HARRIS, M.S. (Battelle Pacific Northwest Laboratory, Richland, WA). Seventeenth Water Reactor Safety Information Meeting. Proceedings: Volume 1. Compiled by WEISS, A.J. (Brookhaven National Laboratory, Upton, NY). Nuclear Regulatory Commission, Office of Nuclear Regulatory Research Washington, DC, 23-25 October 1989, NUREG/CP-0105, Volume 1, CONF-8910222, Volume 1, March 1990, pp. 325-347.

This paper describes the methodology and results of a value-impact assessment that was prepared for the U.S. Nuclear Regulatory Commission. The purpose of the assessment was to provide guidance to the NRC in determining whether to implement regulatory action that would specify requirements for the preparation of acceptable normal and abnormal operating procedures (NOPs and AOPs) by the Commission's licensee nuclear power plants. Current procedure inadequacies and a candidate NRC regulatory action that could be instituted to correct them are described. The results of this value impact assessment are expressed in terms of ten cost/benefit attributes that could be affected by such NRC regulatory action. Five of these attributes required the calculation of change in public risk that could be expected to result from the action that, in this case, required determining the safety significance of NOPs and AOPs. In order to estimate this safety significance, a multi-step methodology was created that relies on an existing probabilistic risk assessment to provide a quantitative framework for modeling the role of operating procedures.

1842. THE SELECTION OF FIELD COMPONENT RELIABILITY DATA FOR USE IN NUCLEAR SAFETY STUDIES. COXSON, B.A. (Nuclear Electric PLC, Knutsford (United Kingdom)); TABAIE, M. (Electrowatt, Birchwood (United Kingdom)). Safety and Reliability in the 90s: Will Past Experience or Prediction Meet our Needs. Edited by WALTER, M.H. (Electrowatt Consulting Engineers and Scientists, Warrington (United Kingdom) Safety

and Reliability Society, Manchester (United Kingdom)); COX, R.F. (AEA Technology, Safety and Reliability, Warrington (United Kingdom) Safety and Reliability Society, Manchester (United Kingdom)). SARSS '90: Safety and Reliability Society Symposium, 19-20 September 1990, Published by Elsevier Applied Science, CONF-9009400--, 1990, pp. 238-250.

The paper reviews the user requirements for field component failure data in nuclear safety studies, and the capability of various data sources to satisfy these requirements. Aspects such as estimating the population of items exposed to failure, incompleteness, and under-reporting problems are discussed. The paper takes as an example the selection of component reliability data for use in the Pre-Operational Safety Report (POSR) for Sizewell "B" Power Station, where field data has in many cases been derived from equipment other than that to be procured and operated on site. The paper concludes that the main quality sought in the available data sources for such studies is the ability to examine failure narratives in component reliability data systems for equipment performing comparable duties to the intended plant application. The main benefit brought about in the last decade is the interactive access to data systems which are adequately structured with regard to the equipment covered, and also provide a text-searching capability of quality-controlled event narratives.

1843. INTERIM SPENT-FUEL STORAGE OPTIONS AT COMMERCIAL NUCLEAR POWER PLANTS. THAKKAR, A.R.; HYLKO, J.M. (Fluor Daniel, Inc., Chicago, IL). International Conference on Energy in the 90's, Edited by HOBBS, B.F., 11-13 March 1991, American Society of Civil Engineers, CONF-910306--, 1991, pp. 134-139.

Although spent fuel can be stored safely in water-filled pools at reactor sites, some utilities may not possess sufficient space for life-of-plant storage capability. In-pool storage capability may be increased by reracking assemblies, rod consolidation, double tiering spent-fuel racks, and by shipping spent fuel to other utility-owned facilities. Long-term on-site storage capability for spent fuel may be provided by installing (dry-type) metal casks, storage and transportation casks, concrete casks, horizontal concrete modules, modular concrete vaults, or by constructing additional (pool-type) storage installations. Experience to date has provided valuable information regarding dry-type or pool-type installations, cask handling and staffing requirements,

security features, decommissioning activities, and radiological issues.

1844. REACTOR SHIELDING WALL. KUME, T.; FURUKAWA, H., (Hitachi Ltd., Tokyo (Japan)), Japanese Patent Number JP 3-29891 A, Hitachi Ltd., Tokyo (Japan), 7 February 1991, in Japanese, (3 p).

The present invention concerns shielding walls suitable to the reduction of radiation doses in a reactor container dry well upon periodical inspection of a nuclear power plant. The reactor shielding walls are extended upwardly near the dry well. Then, a door is disposed to a penetrating port of a large diameter such as for a main steam pipeline. Then, when an operator enters the reactor container dry well upon periodical inspection, the door is closed. The radiation dose is higher in the upper portion in the lower portion of the reactor container dry well than in the prior art structure. However, the radiation dose in the upper portion of the dry well can be reduced by the present invention. Accordingly, the radiation exposure dose to the operator can be reduced to attain a safety effect.

1845. SHOWING COMPLIANCE WITH PROBABILISTIC SAFETY CRITERIA AND OBJECTIVES. VERSTEEG, M.F. (Nuclear Safety Department, Voorburg (Netherlands)); PLUG, C. (Ministry of Housing, Physical Planning and the Environment, Leidschendam (Netherlands)). CSNI Workshop on PSA Applications and Limitations. Proceedings. Edited by MOLINA, T., U.S. Nuclear Regulatory Commission, Div. of Systems Research, Washington, DC, Sandia National Labs. Committee on the Safety of Nuclear Installations (CSNI) on Probabilistic Safety Assessment (PSA) Applications and Limitations, 4-6 September 1990, NUREG/CP-0115, SAND-90-2797; CONF-9009346--, February 1991, pp. 231-247.

In the Netherlands probabilistic safety criteria and goals (PSCs) on the public health level have been developed. As far as these PSCs are applicable to the operation of nuclear power plants, this article describes some pitfalls of using probabilistic safety assessments (PSAs) in showing compliance with these risk criteria and goals. Therefore this article gives some guidance in dealing with uncertainties and cutting off the frequencies in the extreme low probability domain where the results of the PSA are very uncertain, and showing compliance with safety goals and criteria becomes ambiguous and disputable. It is not an aim of this article to suggest the use and the development of safety criteria and goals for the

various aspects of risk. Although this article more or less is restricted to the nuclear world, some conclusions are also valid for the use of non-nuclear PSAs, thus applicable to risk management in general.

1846. RADIATION PROTECTION TRAINING: PHILOSOPHY, QUALIFICATIONS AND TRAINING IN CANADA. RHEAUME, R., Commission of the European Communities, Luxembourg, Seminar on Radiation Protection Training and Information for Workers, 28-30 November 1988, EUR-12177, CONF-8811372--, 1989, pp. 81-112.

The Canadian nuclear power industry record concerning radiation protection is commendable, with low doses and no over-exposures, and this despite the much higher risk in this sector compared with the other sectors studied. Various technical factors have contributed towards this achievement, of course, but the philosophy adopted and the high level of qualification of the radiation protection personnel have played a major role. A great deal of work has been put into training nuclear plant operatives up to the desired level, since many were not particularly well-educated. The solution has been to improve the teaching aspect of the training, and to clearly identify the workers' needs by analyzing the tasks which they are required to perform. Consequently, the creation of a systematic training program for users of radiation sources in universities and hospitals would help to improve radiation protection, especially as the users in this sector are generally better educated than the maintenance workers in power stations. In the mining sector, training does not have such a significant role to play in dose reduction, but it will nevertheless have an indirect impact: once miners know more about radiation and its health implications, their demands, on issues such as ventilation, for example, will become more specific and have more substance.

1847. ESTABLISHMENT OF RADIATION PROTECTION BOUNDARIES FOR NUCLEAR POWER PLANTS. NICOLL, R.M. (Tennessee Valley Authority, Chattanooga, TN). Health Physics, Volume 60, Number 5, May 1991, pp. 725-731.

Activities involving the possession and use of radioactive material require the application of controls to ensure the health and safety of the worker and general public. One of the first steps in adequately controlling any licensed activity is the establishment of zones and boundaries that will exist for purposes of restricting or regulating personnel radiological exposure. Title 10 of the Code of Federal Regulations, Part 20 (10 CFR 20) defines the "restricted area" (RA) and

prescribes the administrative and radiological protection controls pertinent to the RA. For nuclear power facilities, the determination of the physical relation of the RA boundary with other physical or administrative boundaries, such as the site boundary, the radiologically controlled area (RCA), and the exclusion area boundary, must be based on an evaluation that considers all applicable federal requirements and limits for each zone or boundary contained within or interfacing with the RA. This paper presents a discussion of the factors that should be considered and an evaluation methodology that can be utilized based on the generation of two-dimensional isodoses.

1848. A SYSTEM TO AID LOCKOUT PROCEDURES. MEAUZOONE, L. (Électricité de France, Service de la Production Thermique, Villeurbanne (France)); MADER, J. (Sodeteg-T.A.I., BUC (France)). Balancing Automation and Human Action in Nuclear Power Plants. Proceedings Series. International Atomic Energy Agency, Vienna (Austria), Nuclear Energy Agency, Paris (France), 9-13 July 1990, CONF-900710--, 1991, in French, pp. 279-294.

At a nuclear power plant maintenance work involving mechanical and electrical circuit equipment requires that the working area be isolated and is subject to strict procedures designed to ensure the safety of personnel and the integrity of equipment. KAC (computer to aid lockout procedures) is a computer system for a nuclear facility (N4 series) which can be used to automate certain aspects of procedure monitoring and isolation design without replacing the operator, who continues to have overriding control. The system thus offers support for operation and decision making. The operator retains responsibility for and full control over operation. The KAC system helps in the detection of anomalies and, when automatic verification is likely to be exhaustive, does not allow violation of the safety rules. The purpose of KAC is thus to provide maximum assistance at the cost of a minimum of constraints in applying the protection and safety rules.

1849. OPERATORS' VIEW OF KEY ISSUES CONFRONTING NUCLEAR POWER PLANT DECOMMISSIONING. ESSMANN, J. (Preussenelektra, Hannover (Germany)). Decommissioning of Nuclear Installations. Edited by PFLUGRAD, K.; BISCI, R.; HUBER, B.; SKUPINSKI, E. (Commission of the European Communities, Brussels (Belgium)). Directorate General for Science, Research and Development) 89th International Conference on Decommissioning of

Nuclear Installations, 24-27 October 1989, Published by Elsevier Applied Science, CONF-891077--; EUR--12690., 1990, pp. 20-27.

Within some UNIPED member countries the first generation of nuclear power plants are reaching, or have reached, the end of their working lives after more than 25 years of successful operating time. In the past in the UNIPED countries many studies are available, which demonstrate the technical and financial feasibility of decommissioning. The more the decommissioning becomes concrete the more it is time to strike the balance and to look deeper on probable key issues, which could have an influence on realization of decommissioning. From the operators' viewpoint all relevant factors will be discussed and the key issues confronting the present planning of decommissioning of nuclear power plant decommissioning will be identified. It will be shown that all the possible options at present can not be used because the necessary requirements have not been met so far.

1850. REPORT ON THE YEAR 1989. Gemeenschappelijke Kernenergiecentrale Nederland NV, Arnhem (Netherlands), GKN-1989, 1990, in Dutch, (29 p).

This report gives a survey of the operation of the nuclear power plant "Dodewaard" during 1989. Also the state of affairs with regard to COVRA, international developments, information, radiation control, radioactive waste and exploitation during 1989 is summarized. Finally some future perspectives are outlined.

1851. 1989 MAINTENANCE OPERATIONS OF THE PAKS NUCLEAR POWER PLANT UNITS, HUNGARY. LUKACS, P.; ZSOLDOS, F. (Pskdi Atomeroemue Vallalat, Paks (Hungary)). PAV Koezlemenye (Paksi Atomeroemue Vallalat) (Hungary), Volume 1, 1990, in Hungarian, pp. 65-69.

The 1989 reactor maintenance activities of the Paks Nuclear Power Plant units are overviewed. The following maintenance activities are presented briefly: maintenance design, mechanical, electrical and control system maintenance, materials testing, radiation protection inspection and quality assurance. Maintenance and repair of damages of the main circulation pump is reported. The neutron flux inspection system was reconstructed.

1852. HANDLING OF WWER-440 TYPE NUCLEAR REACTOR INTERNAL STRUCTURES

USING A VIDEO SYSTEM. LAFFERTON, I. (Paksi Atomeroemue Vallalat, Paks (Hungary)). PAV Koezlemenye (Paksi Atomeroemue Vallalat) (Hungary), Volume 1, 1990, in Hungarian, pp. 62-64.

The hauling, handling and motioning of internal reactor structure should be automated as far as possible in order to avoid hazards by intensive nuclear radiation. A TV-camera, remote-control, automatic crane and hauling control system applied at the Paks Nuclear Power Plant, Hungary, for the lifting and handling of burnt fuel elements during reloading are described. The crane operator can work in radiation protected room by the video system. This means that the radiation burden of personnel participating in reloading can be reduced to the quarter of value obtained without the video-camera remote-control system.

1853. 89TH INTERNATIONAL CONFERENCE ON DECOMMISSIONING OF NUCLEAR INSTALLATIONS. PFLUGRAD, K.; BISC, R.; HUBER, B.; SKUPINSKI, E. (Commission of the European Communities, Brussels (Belgium)). Directorate General for Science, Research and Development, 24-27 October 1989, Published by Elsevier Applied Science, CONF-891077--; EUR-12690, 1990, (858 p).

The European Community (EC) is conducting research on the decommissioning of nuclear installations with the prime objective of developing effective techniques and procedures to ensure the protection of man and his environment against the potential hazards of nuclear installations that have been finally shut down. This book contains the results of the 1984-1988 five-year program of research on the decommissioning of nuclear power plants which were presented at the International Conference on the Decommissioning of Nuclear Installations, held at the Palais des Congres, Brussels, Belgium, 24-27 October 1989. The program extended to all types of installations of the nuclear fuel cycle and produced numerous scientific and technical achievements, some of which have already been tested on an industrial scale. Great progress has in particular been achieved in the reduction of metal waste arising from decommissioning, due to advances in areas such as aggressive decontamination procedures and techniques for melting and recycling low-level radioactive waste metal. Decommissioning of Nuclear Installations will be of specific interest to those in the nuclear, reliability, safety, mechanical and structural engineering industries. All 82 papers are indexed separately. The discussions at

the end of each session are recorded but are not indexed separately.

1854. INSPECTION ROBOT FOR TANK WALLS IN NUCLEAR POWER PLANTS. SATO, K.; FUKAGAWA, Y.; TOMINAGA, I., (JGC Corporation, Yokohama (Japan)). Proceedings of the International Topical Meeting on Remote Systems and Robotics in Hostile Environments, 29 March - 2 April 1987, Published by the American Nuclear Society, CONF-870301--, 1987, pp. 177-181.

To improve the safety and reliability of nuclear power plants, it is vital to ensure that equipment and facilities are thoroughly maintained to allow full time operation under the design operating conditions. Inspections prevent unscheduled shutdowns, and result in improved operating efficiency and economy in power generation. A robot which travels on cylindrical shell walls and bottom dishes has been developed to inspect large carbon or stainless steel storage tanks containing radioactive liquids, and to protect maintenance personnel from radioactive exposure. The robot named IROW is equipped with a wall traversing mechanism consisting of a pair of vacuum disks with wheels, and an inspection mechanism consisting of devices for tank walk surface condition observation and shell wall thickness measurement. IROW tested to inspect tanks whose surface dose rate ranged from 1.0 to 10 rem/hr. Compared to inspection by personnel, IROW reduced the working period to 1/3; the total labor man-days to 1/4; and the accumulative dose rate to 1/8.

1855. METHOD TO CONTROL N-16 RADIATION LEVELS IN STEAM PHASE BWRS. COWAN, R.L. II; DUITNA, D.; LAW, R.J.; LIN, C.C.; RUIZ, C.P.; SIMPSON, J.L., Patent Number U.S. 4992232 A, General Electric Co., New York, NY, 12 February 1991.

This patent describes improved method of utilizing hydrogen water chemistry in a boiling water reactor nuclear power plant, the plant including a reactor which boils water to produce a liquid phase and a steam phase, a turbine for extracting power from the steam phase, a condenser for condensing steam from which power has been extracted and a feed water system returning the condensed steam as boiler feed water to the reactor, wherein the process of injecting hydrogen into the boiler feed water is utilized to reduce stress corrosion, the improvement comprises the step reducing the transfer of radioactive gaseous nitrogen compounds from the liquid phase to the

steam by inhibiting the formation of the nitrogen compounds in the liquid phase.

1856. SURVEYOR MOBILE SURVEILLANCE SYSTEM FOR HAZARDOUS ENVIRONMENTS. SILVERMAN, E.B.; SIMMONS, R.K. (Ard Corp., Columbia, MD); KNIAZEWYCZ, B.G.; DARVISH, A.R. (KLM Engineering, Inc., Walnut Creek, CA); IRVING, T.L. (Niagara Mohawk Power Corp., Syracuse, NY)). Proceedings of the International Topical Meeting on Remote Systems and Robotics in Hostile Environments, 29 March - 2 April 1987, Published by American Nuclear Society, CONF-870301--, 1987, pp. 65-70.

A successful program was recently conducted to test and evaluate a commercially ready, wireless, remotely operated surveillance system for use in nuclear power plants. This evaluation of the Surveyor mobile surveillance system took place at Niagara Mohawk Power Corporation's Nine Mile Point Nuclear Power Station. The remotely operated vehicle measures radiation, temperature and relative humidity and provides optical inspection capability. The vehicle is readily maneuvered in 36-inch wide passageways and labyrinth entries and can climb stairs, negotiating 180-degree turns on stair landings. The system consists of a supervisory control station and a rugged, remotely-operated, battery-powered vehicle. The surveyor system is specifically designed to decrease personnel radiation exposure by supplementing the functions of an auxiliary operator or wealth physics technician to perform periodic component inspections inside particular areas within a nuclear power plant.

1857. 5TH INTERNATIONAL CONFERENCE ON WATER CHEMISTRY OF NUCLEAR REACTOR SYSTEMS. British Nuclear Energy Society, London (United Kingdom), Royal Society of Chemistry, London (United Kingdom) European Nuclear Society, Petit-Lancy (Switzerland), Institution of Chemical Engineers, London (United Kingdom), 23-27 October 1989, CONF-891026--, 1989.

The water chemistry aspects of nuclear reactors are of critical importance. This book is a state-of-the-art review of the best international experience. It embodies the expertise presented at the fifth triennial international conference on the water chemistry of nuclear reactor systems in October 1989. The book is published in two volumes. Topics covered range widely and are grouped into eight sections. These include PWR experience (13 papers), radiation control measures (8 papers), BWR operational experience (10 papers), radiolysis in BWR coolants (11 papers),

decontamination (10 papers), secondary-side chemistry (8 papers), water chemistry purification (8 papers), and fission product chemistry (4 papers). There are also 45 poster papers on aspects of water chemistry. All the papers are indexed separately. Discussion on the papers is included in volume 2 but is not indexed.

1858. ALTERNATIVES TO CURRENT PROCEDURES USED TO ESTIMATE CONCENTRATIONS IN BUILDING WAKES. RAMSDELL, J.V. JR. (Pacific Northwest Laboratory, Richland, WA)). Proceedings of the 21st DOE/NRC Nuclear Air Cleaning Conference. Volume 2, Sessions 9-16. Edited by FIRST, M.W. (Harvard University, Harvard Air Cleaning Lab, Boston, MA), Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, 13-16 August 1990, NUREG/CP-0116, Volume 2, CONF-900809, Volume 2, February 1991, pp. 714-729.

This paper discusses a new model for estimating concentrations in building wakes and offers alternative guidance for evaluating the consequences of releases from nuclear facilities. The new model accounts for about 57% of the variability in concentrations observed in wakes. Models suggested in current guidance account for less than 10% of the variability. The alternative guidance demonstrates application of the new model. It covers ground-level releases, roof-top vent and short-stack releases, and elevated releases. It also includes a procedure for accounting for uncertainty in building-wake diffusion estimates.

1859. EPRI RESEARCH PUBLICATIONS, PRODUCTS, AND EXPERTISE IN MAINTENANCE. Electric Power Research Institute, Palo Alto, CA, EPRI-NP-7014-Rev.1, May 1991, (140 p).

The intent of this revision to the original NP-7014D cross-referencing index is to identify EPRI expertise and the associated EPRI products with respect to the Maintenance Issues. Whether prepared by the Nuclear Maintenance Applications Center (NMAC), or other organizations within EPRI, they are being included for information. This document was compiled from the EPRI resource catalogs database. The EPRI/NMAC Project Manager Expertise List provides contact phone numbers for selected maintenance subjects. The EPRI Research, Technical Reports, and Products information provides a listing and brief description of EPRI projects and products by selected maintenance subjects. Additionally, listings of EPRI maintenance-related documents are sorted by document number and title. Certain research reports

which contain software have been intentionally omitted due to the lack of (at most utility maintenance departments) computer hardware. The Nuclear Maintenance Applications Center (NMAC) has not reviewed or endorsed the EPRI products listed in this document, except for those produced by NMAC. The planned updating of this cross-reference index is semiannually. Should any further assistance be required for plant-related maintenance problems, contact NMAC at (800) 356-7448.

1860. ENVIRONMENTAL QUALITY STANDARDS: SIGNIFICANCE AND LIMITATIONS. BRUZZI, L.; SABATTINI, A. (Bologna University (Italy)). Energia e Innovazione: Notiziario dell'ENEA (Italy), Volume 36, Numbers 11-12, Nov-Dec 1990, in Italian, pp. 43-59.

Sensitivity and parametric analyses are performed on industrial plant (e.g., fossil-fuel power, nuclear power, chemical, metals working, etc.) materials and waste product risk assessment methodologies employed in the setting up of environmental and human health standards. Comparisons are made between established acceptable environmental and human health limits. Emphasis is on sulfur and nitrogen oxides emissions, ionizing radiations, harmful toxic and noxious materials, dose/response probabilistic estimation techniques. The application of the Alara (as low as reasonably achievable) and risk-benefit (adopted by the ICRP, International Commission for Radiological Protection) principles to environmental impact assessments is also considered to illustrate the uncertainties and complexities involved in the fixing of pollution standards that adequately protect both the environment and human health.

1861. RECENT DEVELOPMENTS IN RADIATION CONTROL TECHNOLOGY. WOOD, C.J.; OCKEN, H. (Electric Power Research Institute, Palo Alto, CA). 5th International Water Conference on Chemistry of Nuclear Reactor Systems, Volume 1, British Nuclear Energy Society, London (United Kingdom) Royal Society of Chemistry, London (United Kingdom) European Nuclear Society, Petten (Switzerland), Institution of Chemical Engineers, London (United Kingdom), 23-27 October 1989, CONF-891026--, 1989, pp. 49-53.

The implementation of technology to control radiation fields at U.S. nuclear power plants is occurring at an accelerating pace and has helped utilities achieve significant reductions in occupational exposures in recent years. This paper describes exposure trends,

the use of improved technology and future developments.

1862. THE NUCLEAR POWER PLANTS DE-COMMISSIONING IN FRANCE: DEFINITIONS AND RESPONSIBILITIES CONCERNING REGULATIONS. LEBRUN, P. (Service Central de Surete des Installations Nucleaires, PARIS (France)). Decommissioning of Nuclear Facilities of the Fuel Cycle. Societe Francaise d'Energie Nucleaire (SFEN), Paris (France), 15 March 1988, CONF-8803283--, 1988, in French, pp. 6-16.

Unlike those in Germany and in the United States, the safety regulations in France, for the decommissioning of the nuclear facilities, are not defined. It is a special application case of a general regulation based on the 11th December 1963 modified decree. The decommissioning operations, for which several possibilities have to be considered, are reviewed and specified. The experience acquired in this field is reviewed.

1863. CORROSION IN NUCLEAR POWER PLANTS MODERATED BY WATER CHEMISTRY. MAEKELAE, K.; AALTONEN, P. (Valtion Teknillinen Tutkimuskeskus, Ydinvoimatekniikan Lab, Otaniemi (Finland)); PAAVOLA, A. (Imatran Voima Oy (IVO), Helsinki (Finland)). 5th International Conference on Water Chemistry of Nuclear Reactor Systems, Volume 1, British Nuclear Energy Society, London (United Kingdom), Royal Society of Chemistry, London (United Kingdom), European Nuclear Society, Petit-Lancy (Switzerland), Institute of Chemical Engineers, London (United Kingdom), 23-27 October 1989, CONF-891026--, 1989, pp. 39-42.

The reliable operation of nuclear power plants requires evaluation and understanding of the long term deterioration mechanisms of critical components. Since the environment has an influence on the degradation of materials through corrosion, it is necessary to use and further develop water chemistry monitoring systems. Successful water chemistry control requires both regular and continuous monitoring of such parameters as dissolved oxygen content, pH, conductivity and impurity contents. Conventionally the monitoring is carried out at low pressures and temperatures which, however, has some shortcomings. The results and operating experiences gained during high temperature high pressure monitoring at Loviisa PWR are reported.

1864. EVALUATION OF SELECTED PARAMETERS ON EXPOSURE RATES IN WESTING-

HOUSE DESIGNED NUCLEAR POWER PLANTS. BERGMANN, C.A. (Westinghouse Electric Corp., Pittsburgh, PA). 5th International Conference on Water Chemistry of Nuclear Reactor Systems, Volume 1. British Nuclear Energy Society, London (United Kingdom), Royal Society of Chemistry, London (United Kingdom), European Nuclear Society, Petit-Lancy (Switzerland), Institution of Chemical Engineers, London (United Kingdom), 23-27 October 1989, CONF-891026--, 1989, pp. 9-16.

During the past ten years, Westinghouse under EPRI contract and independently, has performed research and evaluation of plant data to define the trends of ex-core component exposure rates and the effects of various parameters on the exposure rates. The effects of the parameters were evaluated using comparative analyses or empirical techniques. This paper updates the information presented at the Fourth Bournemouth Conference and the conclusions obtained from the effects of selected parameters namely, coolant chemistry, physical changes, use of enriched boric acid, and cobalt input on plant exposure rates. The trends of exposure rates and relationship to doses is also presented.

1865. OVERVIEW OF ACTIVITY BUILD UP IN THE LATER WEST GERMAN PWRS. RUEHLE, W. (Kernkraftwerk Philippsburg GMBH (Germany)); RIESS, R. (Siemens Ag Unternehmensbereich KWU, Erlangen (Germany)). 5th International Conference on Water Chemistry of Nuclear Reactor Systems, Volume 1, British Nuclear Energy Society, London (United Kingdom), Royal Society of Chemistry, London (United Kingdom), European Nuclear Society, Petit-Lancy (Switzerland), Institution of Chemical Engineers, London (United Kingdom), 23-27 October 1989, CONF-891026--, 1989, pp. 1-8.

The efforts during the recent years to reduce dose rates and man rem in German nuclear power plants led to modifications in the material concept and the primary coolant chemistry. By comparison of three plants which differ in these issues, the effectiveness of the modifications has been evaluated. It can be shown that both measures, substitution of cobalt-based hardfacing alloys and a higher Ph value in the primary coolant, result in lower dose rates at the primary coolant pipes. Results of trace metal determination by stripping voltammetry and cation ion-chromatography are presented to obtain more information about the transport of radioactive cobalt in the primary coolant system.

1866. HYDRAULIC DESIGN OPTIMIZATION FOR HOLLOW FIBRE FILTER SYSTEM. UTAMURA, M.; IBE, E.; UCHIDA, S.; SAWA, T.; ADACHI, T.; SHINDO, T.; OHSUMI, K. (Hitachi Ltd., Ibaraki (Japan)). 5th International Conference on Water Chemistry of Nuclear Reactor Systems, Volume 1, British Nuclear Energy Society, London (United Kingdom), Royal Society of Chemistry, London (United Kingdom), European Nuclear Society Petit-Lancy (Switzerland), Institution of Chemical Engineers, London (United Kingdom), 23-27 October 1989, CONF-891026--, 1989, pp. 329-330.

An analytical model has been developed to describe hydraulic characteristics of a hollow fiber filter (HFF) for the purification of condensate of BWR nuclear power plants. Based on this, a module structure was proposed to minimize pressure drop at the beginning of its operation. The mechanism of iron ion crystallization on the HFF surface was clarified and a countermeasure against it was developed. Taking account of this effect, a simulation code was developed to predict the pressure drop trend in the course of the HFF operation.

1867. OPTIMAL SAFETY IS THE PHILOSOPHY. Modern Power Systems (United Kingdom), Volume 10, Number 12, December 1990, pp. 31-33.

In order to promote international sharing of knowledge of radiation dose reduction at nuclear plants, the OECD's Paris-based Nuclear Energy Agency (NEA) has launched plans and preparations for a new institution, the Information System on Occupational Exposure (ISOE). Eight OECD countries are evaluating a pilot project and the NEA is discussing the establishment of a permanent ISOE. The aim is to have instant answers to varied questions on effective dose-reduction techniques, typical doses for particular jobs, dose rates at standard points in plants and the way they are being affected by dose-reduction techniques. With answers based on data from all OECD countries with experience in nuclear power plant operations. At the centre of the nuclear industry's radiological protection is the principle that all radiation doses should be kept as low as reasonably achievable (alara), economic and social factors being taken into account. If this principle were applied beyond radiological protection to safety in general, for example, road safety, it, would lead to much better overall safety.

1868. THE HTR-2 MODULAR NUCLEAR POWER PLANT. EVALUATION OF THE DESIGN

CONCEPT FROM THE POINT OF VIEW OF SAFETY ENGINEERING. SUMMARY REPORT. Technischer Ueberwachungs-Verein Hannover e.V. (Germany), Bundesministerium fuer Forschung und Technologie, Bonn (Germany), INIS-mf-12195, May 1990, in German, (80 p).

The summary report presents the safety engineering aspects of a HTR design concept with two modules for cogeneration of electricity and process heat, or district heat, respectively. The evaluation is based on a complete and representative list of design-basis accidents, and from the subsequent accident analyses, design criteria and requirements are derived for all plant components and systems. The analyses also yield information on representative radiological events and conditions. The results show that the radioactivity release due to specified normal operation will keep the radiation dose to the environment below the limits defined in section 45 of the Radiation Protection Ordinance.

1869. NUCLEAR POWER AND SAFETY. SAUNDERS, P.; TASKER, A., (AEA Technology, London (United Kingdom)), INIS-GB-348, January 1991, (31 p).

Nuclear power currently provides about a fifth of both Britain's and the world's electricity. It is the largest single source of electricity in Western Europe; in France three quarters of electricity is generated by nuclear power stations. This booklet is about the safety of those plants. It approaches the subject by outlining the basic principles and approaches behind nuclear safety, describing the protective barriers and safety systems that are designed to prevent the escape of radioactive material, and summarizing the regulations that govern the construction and operation of nuclear power stations. The aim is to provide a general understanding of the subject by explaining the general principles of the Advanced Gas Cooled Reactor and setting out the UK AEA strategy for nuclear safety, the objective being always to minimize risk.

1870. RADIATION SAFETY OF NUCLEAR POWER PLANT WORKERS. VILKAMO, O. (Finnish Centre for Radiation and Nuclear Safety (STUK), Helsinki (Finland)). ATS Ydintekniikka (Finland), Volume 19, Number 3, 1990, in Finnish, pp. 9-10.

Doses to workers at the nuclear power plants in Sweden and Finland are generally low. The main causes off this situation include appropriate design

and layout as well as successful operation of and radiation protection at the plants. In a Nordic project it was investigated whether the optimization principle recommended by the International Commission on Radiological Protection is being applied at the Nordic nuclear power plants or can its use be extended. The paper summarizes the main results of the study. The study revealed that many radiation protection actions involve rather high costs. General guidance on the optimization can be given.

1871. MEASURES OF MINIMIZING GASEOUS RADIOACTIVE EFFLUENTS FROM THE TEMELIN NUCLEAR POWER PLANT INTO THE ENVIRONMENT TAKING INTO ACCOUNT THE ALARA AND ALARP PRINCIPLES, RESPECTIVELY. SINGER, J.; NOREK, P.; PILAR, P.; STREDA, I.; SRAGA, L. (Temelin Nuclear Power Plant (Czechoslovakia)); KUEHNE, J.; LANGHAUS, J. (Kernkraftwerk Greifswald (Germany)). 22nd Internationales Symposium "Strahlenschutzphysik, 2-6 April 1990, Kernenergie (German Democratic Republic), Volume 34, Number 4, CONF-9004289--, 1991, in German, pp. 156-159.

Temelin nuclear power plant design effluents were compared with actual effluents from nuclear power plants in the Federal Republic of Germany and France which always are lower so that the comparisons were not quite exact. Even so, technical and organization measures are recommended which concern longer delay of noble gases in the containment and in the activated carbon, better filtration of iodines and aerosols and packing of technological circuits of coolant purification and measurement including special canalization. In this way the noble gas effluents can be brought down by approximately 50 times while iodines and aerosols still more in agreement with ALARA or ALARP principles.

1872. AN EFFECTIVE WAY OF RADIATION PROTECTION: THE APCE DECONTAMINATION PROCESS. OERTEL, K. (Kernkraftwerk Rheinsberg (Germany)). Kernenergie (German Democratic Republic), Volume 34, Number 4, 1991, in German, pp. 147-150.

At the Rheinsberg nuclear power plant the primary circuit of the WWER-70 pressurized water reactor has been completely decontaminated, i.e., core included, for many years. In the last years this radiation protection technology has rendered possible to limit the expenses for developing robots and telecontrol apparatus. In connection with other measures the new

technology essentially contributed to keeping the annual collective dose at an acceptably low level.

1873. ELECTROCHEMICAL ION EXCHANGE OF LOMI DECONTAMINATION SOLUTIONS. BRADBURY, D.; ELDER, G.R. (Bradtec Ltd., Bristol (United Kingdom)). Electric Power Research Inst., Palo Alto, CA, Bradtec Ltd., Bristol (United Kingdom), EPRI-NP-7277, May 1991, (35 p).

The objective of the LOMI Electrochemical Ion Exchange (ELOMIX) process is to reduce the volume of waste arising from LOMI decontamination operations. This is achieved by using the conventional ion exchange resin as an intermediate, rather than a final waste form. Radioactive and metal ion constituents removed during the decontamination are converted by the process to a metallic form. In its simplest form the ELOMIX cell consists of three compartments, (anode, cathode and resin) separated by cation permeable membranes. In the resin compartment, radioactive and chemical components are removed from the flowing solution by the resin. These components are then migrated under the influence of an electric field into the cathode compartment where metallic elements are deposited. After successful laboratory testing a pilot scale ELOMIX unit was constructed and shipped to Commonwealth Edison's Dresden Unit 2 LOMI decontamination in October 1990. The pilot-scale cell was tied in to the decontamination equipment. The cell operated successfully during three LOMI steps, processing a total of 165 resin bed volumes of solution, far exceeding the normal capacity of the ion exchange resin. Removal efficiencies for iron and radioactivity were good throughout the test. The ELOMIX unit was successfully operated in the time, operational and safety constraints of a reactor decontamination. The main benefits of the process are reduced waste volume, and greater chemical and radiation stability of the waste, which is particularly relevant to full system decontamination. Although the testing to date has focused on the LOMI process, the ELOMIX technique could in principle be applied to be other decontamination processes.

1874. LICENSED REACTOR NUCLEAR SAFETY CRITERIA APPLICABLE TO DOE REACTORS. U.S. DOE Assistant Secretary for Nuclear Energy, Washington, DC, Office of Nuclear Safety Policy and Standards, DOE/NE-0100T, April 1991, (208 p).

The Department of Energy (DOE) Order DOE 5480.6, Safety of Department of Energy-Owned Nuclear Reactors, establishes reactor safety

requirements to assure that reactors are sited, designed, constructed, modified, operated, maintained, and decommissioned in a manner that adequately protects health and safety and is in accordance with uniform standards, guides, and codes which are consistent with those applied to comparable licensed reactors. This document identifies nuclear safety criteria applied to NRC (Nuclear Regulatory Commission) licensed reactors. The titles of the chapters and sections of U.S. NRC Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Rev. 3, are used as the format for compiling the NRC criteria applied to the various areas of nuclear safety addressed in a safety analysis report for a nuclear reactor. In each section the criteria are compiled in four groups: (1) Code of Federal Regulations, (2) U.S. NRC Regulatory Guides, SRP Branch Technical Positions and Appendices, (3) Codes and Standards, and (4) Supplemental Information. The degree of application of these criteria to a DOE-owned reactor, consistent with their application to comparable licensed reactors, must be determined by the DOE and DOE contractor.

1875. ALARA PROGRAMS. ARE THEY WORKING. BRITZ, W.L., Proceedings of the ANS International Executive Conference on Maintenance Issues, 6-9 December 1987, American Nuclear Society, CONF-8712101--, 1987, pp. 1-8.

This paper discusses exposure reduction in the U.S. and shows how it compares to other countries. The author presents some thoughts regarding the optimization of collective and individual radiation exposures.

1876. PROCEEDINGS OF THE ANS INTERNATIONAL EXECUTIVE CONFERENCE ON MAINTENANCE ISSUES. 6-9 December 1987, American Nuclear Society, CONF-8712101--, 1987, (305 p).

This proceedings contains articles relating to the ANS International Executive Conference on Maintenance Issues. Included are the following groups: Nuclear Power Plant Maintenance Issues, Emerging Technologies, Effective Outage Management.

1877. REPORT TO CONGRESS ON ABNORMAL OCCURRENCES. Nuclear Regulatory Commission, Office for Analysis and Evaluation of Operations, Washington, DC, NUREG-0090, Volume 13, Number 4, March 1991, (20 p).

Section 208 of the Energy Reorganization Act of 1974 identified an abnormal occurrence as an un-

scheduled incident or event that the U.S. Nuclear Regulatory Commission determines to be significant from the standpoint of public health or safety and requires a quarterly report of such events to be made to Congress. This report covers the period from October 1 through December 31, 1990. The report discusses five abnormal occurrences, none of which involved a nuclear power plant. Two involved significant overexposures to the hands of two radiographers, two involved medical therapy misadministration, and one involved a medical diagnostic misadministration. No abnormal occurrences were reported by the Agreement States. The report also contains information that updates a previously reported abnormal occurrence.

1878. CHARACTERIZATION ACTIVITIES FOR RADIOACTIVE WASTE FROM GARIGLIANO AND ALTO LAZIO NPPS. DE CRESENZO, V.; ELETTI, G. (Italian Directorate for Nuclear Safety and Health Protection, Rome (Italy)). Proceedings of Spectrum '88: International Topical Meeting on Nuclear and Hazardous Waste Management, 11-15 September 1988, American Nuclear Society, CONF-880903--, 1988, pp. 547-549.

The conditioning activities of radioactive wastes in Italy have to comply with the technical specifications established by the Technical Guide n. 26 and the Technical Position 1.26 issued by the Italian Directorate for Nuclear Safety and Health Protection (ENEA-DISP). Such technical specifications range from the Quality Control procedures on the conditioning system up to the physical and mechanical characteristics of the final product. The paper presents some highlights of the ongoing characterization activities for radioactive wastes from Garigliano and Alto Lazio NPP with specific reference to the main technical difficulties which have been faced. The considered wastes (IER, filter sludges, evaporator concentrates) are typical of BWRs.

1879. DEVELOPMENT OF A ROBOT FOR NUCLEAR POWER PLANT OPERATION AND MAINTENANCE. YAMAMOTO, SHINJI (Tokyo Electric Power Co., Inc. (Japan)). Karyoku Genshiryoku Hatsuden (Thermal and Nuclear Power) (Japan), Volume 41, Number 11, November 1990, in Japanese, pp. 1413-1421.

In nuclear power plants, such work as site patrol, repair and maintenance during plant operation and annual outage, are done by personnel or by using an automatic machine for limited work. It is necessary to develop several kinds of multi-function robots.

Based on robot technology investigation and needs analysis for the robots by site survey, ten kinds of robots are considered for site patrol, repair and maintenance work, and the priority and effective plan for the development has been decided. Now, two kinds of robots are being developed, for the "Study for Improvement of Manipulator Operation" and "Development of the Compact Inspection Robot."

1880. ECOLOGICAL ASPECTS OF NUCLEAR POWER PLANT DECOMMISSIONING.

HLADKY, E. (Vyskumny Ustav Jadrovych Elektrarni, Trnava (Czechoslovakia)). Economic and Ecological Aspects of Nuclear Power Development. Ceskoslovenska Vedeckotechnicka Spolecnost, Ceske Budejovice (Czechoslovakia). Dum Techniky, 29-30 March 1990, INIS-mf-12817, CONF-9003260--, 1990, in Slovak, pp. 104-111.

Nuclear power plant decommissioning reduces the risk of accidents but it produces a source of radioactive products which can be released into the environment and brings about radiation hazard for the staff. The principles are outlined whose observance can minimize the environmental impact of nuclear power plant decommissioning. Another aspect associated with the decommissioning is the growth of amount of radioactive waste with increasing extent of disassembly of the technological and building equipment. The overall amount of wastes is comparable with that of wastes produced throughout the entire life of the nuclear power plant. The estimate for the Bohunice A-1 plant is 10 600 tonnes. A considerable part is low-level waste. The problems of decontamination, personnel protection from external irradiation, knowledge of the behavior of the radionuclides present in the materials, and of legal conditions for the release of these materials for further use have not yet been resolved.

1881. THE PLEX ASPECTS OF THE STORAGE OR DISPOSAL OF REPLACED NUCLEAR COMPONENTS. KATZ, L.R. (Westinghouse Electric Corp., Nuclear Technology Systems Div., Pittsburgh, PA). Proceedings of the ANS Topical Meeting on Nuclear Power Plant Life Extension. American Nuclear Society, 31 July - 3 August 1988, CONF-880748--, 1988, pp. 276-288.

One important aspect of plant life extension that must be addressed by the utility industry concerns the disposal of replaced radioactive components. The alternatives for component disposal include on-site storage and shipment to a low-level waste disposal facility. Very little actual experience exists in the

area of component disposal since only a few major changeouts have occurred in the nuclear industry. The precedents include steam generator, core structures, and reactor internals subcomponent replacements. This paper addresses the issues related to the disposal of components that may require replacement for PLEX, evaluate the available disposal options, and recommend a course of action for such aspects as handling and storage, radiological impact and protection, and cost comparison for the major NSSS component disposal alternatives.

1882. CONVERSION OF A MAINFRAME SIMULATION FOR MAINTENANCE PERFORMANCE TO A PC ENVIRONMENT. GERTMAN, D.I. (Idaho National Engineering Laboratory, Idaho Falls, ID). Transactions of the Eighteenth Water Reactor Safety Information Meeting. Compiled by WEISS, A.J. (Brookhaven National Laboratory, Upton, NY). Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Washington, DC, NUREG/CP-0113, October 1990, p. 11.13.

The computer model MAPPS, the Maintenance Personnel Performance Simulation, has been developed and validated by the U.S. NRC (Nuclear Regulatory Commission) in order to improve maintenance practices and procedures at nuclear power plants. This model has now been implemented and improved, in a PC (personal computer) environment and renamed MICROMAPPS. The model is stochastically based and users are able to simulate the performance of 2- to 8-person crews for a variety of maintenance tasks under a variety of conditions. These conditions include aspects of crew actions as potentially influenced by the task, the environment, or the personnel involved. For example, the influence of the following factors is currently modeled within the MAPPS computer code: (1) personnel characteristics include but are not limited to intellectual and perceptual motor ability levels, the effect of fatigue and conversely, of rest breaks on performance, stress, communication, supervisor acceptance, motivation, organizational climate, time since the tasks was last performed and the staffing level available; (2) task variables include but are not limited to time allowed, occurrence of shift change, intellectual requirements, perceptual motor requirements, procedures quality, necessity for protective clothing and essentiality of a procedures quality, necessity for protective clothing and essentiality of a subtask; and (3) environment variables include temperature of the workplace, radiation level, and noise levels. The output describing maintainer performance includes subtask and

task identification, success proportion, work and wait durations, time spent repeating various subtasks and outcome in terms of errors detected by the crew, false alarms, undetected errors, duration, and the probability of success. The model is comprehensive and allows for the modeling of decision making, trouble-shooting and branching of tasks.

1883. OVERVIEW OF THE NUCLEAR REGULATORY COMMISSION'S REVISED 10 CFR PART 20. PETERSON, H.T. JR., Transactions of the Eighteenth Water Reactor Safety Information Meeting, Compiled by WEISS, A.J. (Brookhaven National Laboratory, Upton, NY). Nuclear Regulatory Commission, Washington, DC, Office of Nuclear Regulatory Research, NUREG/CP-0113, October 1990, p. 3.3.

The principal U.S. Nuclear Regulatory Commission (NRC) regulations covering radiation protection are in Part 20 of Title 10 of the Code of Federal Regulations (10 CFR Part 20). These regulations apply to almost all NRC licensed activities including nuclear power and research reactors, nuclear medicine, industrial radiography and the use of most radioactive isotopes in research. The revised 10 CFR Part 20 is based on the 1977 recommendations of the International Commission on Radiological Protection (ICRP) in ICRP Publication No. 26, implements the 1987 Federal Radiation Guidance on Occupational Exposure and is generally consistent with the 1987 recommendations of the U.S. National Council on Radiation Protection and Measurements (NCRP) in NCRP Report No. 91. There are a number of areas where the Revised Part 20 employs new and more up-to-date scientific information and concepts. One major change is in the use of revised lung and GI tract models and more recent metabolic retention data to calculate Annual Limits on Intake (ALIs) and Derived Air Concentration limits (DACs). The Appendix B in the Revised Part 20 employs new and more up-to-date scientific information and concepts. One major change is in the use of revised lung and GI tract models and more recent metabolic retention data to calculate Annual Limits on Intake (ALIs) and Derived Air Concentration limits (DACs). The Appendix B in the Revised Part 20 contains data on occupational ALIs and DACs and radionuclide concentration limits for releases to the general environment for over 750 radionuclides. The final rule is expected to be published early in September with an implementation date of January 1, 1992.

1884. GUIDELINE FOR THE PROOF OF EXPERT KNOWLEDGE OF NUCLEAR POWER STATION PERSONNEL. AS OF DECEMBER 10, 1990. Bundesministerium fuer Umwelt-, Naturschutz und Reaktorsicherheit, Bonn (Germany), Gemeinsames Ministerialblatt, A (Germany), Volume 42, Number 4, February 1991, in German, pp. 50-56.

The guideline allows the examination of particulars on the state of knowledge and instruction of persons responsible for the erection, management, and monitoring of a plant which applicants in accordance with section 7, subsection (2) No. 1 AtG are required to provide. They apply uniformly to all of the Federal Republic of Germany. They comprise principles, evidence of expert knowledge, rules for the conduct of examinations and instructions for updating knowledge. This guideline supersedes the version of November 13, 1987.

1885. EXPERT KNOWLEDGE OF THE RADIATION PROTECTION OFFICERS IN NUCLEAR POWER PLANT AND OTHER INSTALLATIONS FOR THE FISSION OF NUCLEAR FUELS. BMU CIRCULAR LETTER DATED DECEMBER 10, 1990 - RS II 3 - 15040/1. Bundesministerium fuer Umwelt-, Naturschutz und Reaktorsicherheit, Bonn (Germany), Gemeinsames Ministerialblatt, A (Germany), Volume 42, Number 4, February 1991, in German, pp. 56-60.

The guideline is to be applied for examining and ascertaining the required expert knowledge of radiation protection officers, regarding their activities for the commissioning phase onwards, including first reactor core fueling with nuclear fuel and the operation of nuclear power plant and other installations for the fission of nuclear fuels, of a thermal output greater than 50 kW.

1886. WATER CHEMISTRY DIAGNOSIS SYSTEM FOR NUCLEAR POWER PLANTS. IGARASHI, H.; KOYA, H. (Hitachi Engineering Co. Ltd., Ibaraki (Japan)); OSUMI, K. Hitachi Hyoron (Japan), Volume 72, Number 8, August 1990, in Japanese, pp. 775-782.

The water quality control for the BWRs in Japan has advanced rapidly recently, and as to the dose reduction due to the decrease of radioactivity, Japan takes the position leading the world. In the background of the advanced water quality control like this and the increase of nuclear power plants in operation, the automation of arranging a large quantity of water quality control information and the heightening of its

reliability have been demanded. Hitachi group developed the water quality synthetic control system which comprises the water quality data management system to process a large quantity of water quality data with a computer and the water quality diagnosis system to evaluate the state of operation of the plants by the minute change of water quality and to carry out the operational guide in the aspect of water quality control. To this water quality diagnosis system, high speed fuzzy inference is applied in order to do rapid diagnosis with fuzzy data. The trend of development of water quality control system, the construction of the water quality synthetic control system, the configuration of the water quality diagnosis system and the development of algorithm and the improvement of the reliability of maintenance are reported.

1887. ALARA ENGINEERING AT DEPARTMENT OF ENERGY FACILITIES; BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA. DIONNE, B.J.; KHAN, T.A.; LANE, S.G.; BAUM, J.W., (Brookhaven National Laboratory, Upton, NY), BNL-43228, Volume 2, March 1991, (48 p).

This report is the second in the series of bibliographies supporting the efforts at the Brookhaven National Laboratory ALARA Center on dose reduction at U.S. Department of Energy (DOE) facilities. The BNL ALARA Center was originally established in 1983 under the sponsorship of the U.S. Nuclear Regulatory Commission to monitor dose-reduction research and ALARA activities at nuclear power plants. This effort was expanded in 1988 by the DOE's Office of Environment, Safety and Health to include DOE nuclear facilities. Abstracts for this bibliography were selected from proceedings of technical meetings, journals, research reports, searches of the DOE Energy Data Base, and reprints of published articles provided by the authors. Information that the reader feels should be included in the next volume of this bibliography may be submitted to the BNL ALARA Center. These abstracts, which have a bearing on dose reduction, consolidates information from publications pertinent to Radiological Engineers and Operational Health Physicists. Volume 2 contains 127 abstracts numbered from 69 through 195 as well as author and subject indices. The subject index contains the abstract numbers from both the previous volume and the current volume, the latter being indicated in boldface.

1888. RADIATION EXPOSURE REDUCTION TECHNOLOGY FOR BWRS. OSUMI, K.; MITANI, S. (Hitachi Ltd., Hitachi Works, Ibaraki (Japan)); KATAOKA, I.; TSURUOKA, R.; ASAKURA, Y. Hitachi Hyoron (Japan), Volume 72, Number 10, October 1990, in Japanese, pp. 1051-1058.

Accompanying the increase of the nuclear power stations in operation, in the heightening of the efficiency of operation and maintenance works and the improvement of working environment, it has become increasingly important to keep the radiation dose to which workers are exposed low. Since the introduction of Japanese reactors, Hitachi Ltd. grasped the reduction of radiation dose as an important subject, and the technical development centering around the research on materials and water quality has been promoted. In No. 4 plant in Fukushima No. 2 Nuclear Power Station, Tokyo Electric Power Co., Inc. and No. 2 plant in Shimane Nuclear Power Station, Chugoku Electric Power Co., Inc., in which the latest technology of radiation dose reduction was adopted, the radiation dose equivalent in the ordinary regular inspection works in the first regular inspection was recorded as less than 0.2 man-Sv which was very low value in the world, in this way, the technology of reducing radiation dose in BWRs was established. The progress of the technical development of plant radioactivity reduction, the application of exposure dose reduction technology and the results, the results in Shimane No. 2 plant and so on are reported.

1889. LOW-LEVEL RADIOACTIVE WASTE CLASSIFICATION, WASTE FORM STABILITY, FAILURE MECHANISMS, AND RADIO-NUCLIDE RELEASES. REED, P.R. Transactions of the Eighteenth Water Reactor Safety Information Meeting, Compiled by WEISS, A.J. (Brookhaven National Laboratory, Upton, NY). Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Washington, DC, NUREG/CP-0113, October 1990, pp. 12.9-12.10.

Regulation of low-level radioactive waste (LLW) disposal facilities requires that the NRC staff have an understanding of the complex scientific and technical variables associated with classifying LLW for radionuclide and chemical content and ensuring the waste forms produced by the nuclear power stations and disposed in licensed burial sites maintain stability for at least 300 years. This research paper discusses the low-level radioactive research plan for conducting research on issues of concern to the U.S. Nuclear

Regulatory Commission associated with LLW classification, waste form stability, failure mechanisms of waste forms, radionuclide and chemical releases from cement solidified waste forms, and past experiences involving unsuccessful cement solidification of radwaste. The topics discussed include the regulatory need for performing the research studies, research strategies, current research activities, applications of the research results to NRC's licensing and regulatory environment, and new research studies expected to begin in fiscal year 1991.

1890. ANALYSIS OF REPLIES TO AN IAEA QUESTIONNAIRE ON INSPECTION AND ENFORCEMENT BY THE REGULATORY BODY FOR NUCLEAR POWER PLANTS. International Atomic Energy Agency, Vienna (Austria), IAEA-TECDOC-589, March 1991, (67 p).

At a Special Session of the IAEA General Conference in September 1986, it was suggested that the IAEA could play a role in assisting Member States in the enhancement of their regulatory practices with the objective of increasing the confidence of the international community in the safety of nuclear power programs. As the first stage of this assistance, the Agency initiated an IAEA Questionnaire on Regulatory Practices in Member States with Nuclear Power Programs and summarized the results of an analysis of replies to the questionnaire, issued as IAEA-TECDOC-485 (October 1988). The IAEA Questionnaire on Inspection and Enforcement by the Regulatory Body for Nuclear Power Plants, drafted by a group of consultants and IAEA staff members in August 1989, was initiated as a follow-up to the general questionnaire on regulatory practices. This questionnaire on inspection and enforcement was sent on 3 October 1989 to 31 Member States in which nuclear power plants are under construction and/or in operation. Replies to the questionnaire received from 26 Member States were analyzed by IAEA staff members with the assistance of consultants in order to identify the main differences in approach and important aspects of inspection and enforcement by the regulatory body for nuclear power plants. This report is the summary report on the results of the analysis of the replies to the questionnaire from 26 Member States.

1891. CONSIDERING JOB-RELATED DOSES IN EUROPE. Nuclear Engineering International (United Kingdom), November 1990, Published under the title: Dosimetry and Radiation Protection, 1990-1991 Product Review, pp. 14-16.

As part of its role of safeguarding nuclear workers the Commission of the European Community has created a European data bank on job-related doses. The data bank is intended to correlate jobs and workers doses and is a useful instrument to observe the collected doses of workers in nuclear power plants. Set up in 1980, the database covers doses in PWRs and BWRs. To create it a questionnaire is distributed to plant operators. The database responses provide information on (i) Trends in total collective doses, including those for some ancillary jobs. It is possible to see trends by calendar year and by fuel cycle number. (ii) Other trends such as dose in terms of installed power or dose in terms of power station design. (iii) Dose differences between power stations, normalized against the differences in dose rates. This is only possible for PWRs that monitor dose rates using the EPRI standard monitoring program. (vi) The relative dose rate in each plant for a defined job. The questionnaire is not perfectly adapted to all the working criteria in nuclear power stations but it is an effective tool for implementing radiation protection.

1892. STANDARDIZATION LEADS TO DOSE REDUCTION IN JAPAN. REDUCED RADIATION EXPOSURE AT NUCLEAR POWER PLANTS. Nuclear Engineering International (United Kingdom), published under the title: Dosimetry and Radiation Protection, 1990-1991, Product Review, November 1990, pp. 8-10.

In 1975 the Japanese government agency, MITI, and the Japanese nuclear power industry drew up plans for standardized nuclear power plants. The plans included specific targets for reducing radiation exposures and some detailed recommendations as to how those targets were to be met. The effect of the standardization program is visible in the reduction in exposure at Japanese sites during the last 13 years, and the effect is most dramatic in the BWRs.

1893. REVIEWING OCCUPATIONAL EXPOSURE TRENDS AT PWRs. RADIATION PROTECTION AT PRESSURIZED WATER REACTORS. BURHOLT, G.D. (Associated Nuclear Services, Epsom (United Kingdom)). Nuclear Engineering International (United Kingdom), Volume 35, Number 435, October 1990, pp. 95-98.

Workers at nuclear power plants are monitored to ensure that individual doses remain within limits set by the regulatory authorities. The collective dose is used to describe the overall radiological impact of the plant and, in some countries, is also subject to regulatory control. The major part of the collective dose at

a PWR comes from maintenance and inspection activities during the annual shutdown. As part of an ongoing exercise, collective dose data have been obtained from PWR operators in a number of countries for the period up to and including 1988.

1894. PROCEEDINGS OF THE 21ST DOE/NRC NUCLEAR AIR CLEANING CONFERENCE. Edited by FIRST, M.W. (Harvard University, Harvard Air Cleaning Lab, Boston, MA). Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Washington, DC, 13-16 August 1990, NUREG/CP-0116, Volume 2, CONF-900809, Volume 1, February 1991, (560 p).

Separate abstracts have been prepared for the papers presented at the meeting on nuclear facility air cleaning technology in the following specific areas of interest: air cleaning technologies for the management and disposal of radioactive wastes; Canadian waste management program; radiological health effects models for nuclear power plant accident consequence analysis; filter testing; U.S. standard codes on nuclear air and gas treatment; European community nuclear codes and standards; chemical processing off-gas cleaning; incineration and vitrification; adsorbents; nuclear codes and standards; mathematical modeling techniques; filter technology; safety; containment system venting; and nuclear air cleaning programs around the world.

1895. DOSE-REDUCTION TECHNIQUES FOR HIGH-DOSE WORKER GROUPS IN NUCLEAR POWER PLANTS. KHAN, T.A.; BAUM, J.W.; DIONNE, B.J. (Brookhaven National Laboratory, Upton, NY). Nuclear Regulatory Commission, Division of Regulatory Applications, Washington, DC, NUREG/CR-5139, BNL-NUREG-52278, March 1991, (77 p).

This report summarizes the main findings of a study of the extent of radiation dose received by special work groups in the nuclear power industry. Work groups which chronically get large doses were investigated, using information provided by the industry. The tasks that give high doses to these work groups were examined and techniques described that were found to be particularly successful in reducing dose. Quantitative information on the extent of radiation doses to various work groups shows that significant numbers of workers in several critical groups receive doses greater than 1 and even 2 rem per year, particularly contract personnel and workers at BWR-type plants. The number of radiation workers whose lifetime dose is greater than their age is much less.

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Although the techniques presented would go some way in reducing dose, it is likely that a sizeable reduction to the high-dose work groups may require development of new dose-reduction techniques as well as major changes in procedures.

1896. A UNIVERSAL BILATERAL MANUAL CONTROLLER UTILIZING A UNIQUE PARALLEL ARCHITECTURE. BEVILL, P.J. (University of Texas, Austin, TX); LOVETT, J.T., American Nuclear Society (ANS) Annual Meeting, 10-14 June 1990, Transactions of the American Nuclear Society, Volume 61, CONF-900608--, June 1990, pp. 409-410.

Since 1987, the Advanced Technology Division of the U.S. Department of Energy has sponsored a team composed of four universities and Oak Ridge National Laboratory to pursue research leading to the development and deployment of an advanced robotic system. The tasks to be performed by this system will be those tasks that are hazardous to humans, that generate significant occupational radiation exposure, and those which can be performed more rapidly and more accurately by an automated system. An essential component of the program plan is the annual technology demonstration performed at the Center for Engineering Systems Advanced Research (CESAR) at Oak Ridge National Laboratory. The scenario selected for the 1989 technology demonstration involved the cleanup of a spill of a simulated hazardous material. The demonstration utilized the seven-degrees-of-freedom CESAR manipulator, which is mounted aboard the HERMES III transporter. The transporter traveled to the site of the simulated spill through an obstacle-strewn environment both under direct human control and autonomously, navigating by reference to the computer-stored world model. After arriving at the site of the spill, the vision system scanned and located the spill, thus determining its position in local and global coordinate frames. This information was used to generate a manipulator sweep pattern that resulted in the removal of the spilled material by a vacuum device.

1897. FASTER-THAN-REAL-TIME ROBOT SIMULATION FOR PLAN DEVELOPMENT AND ROBOT SAFETY. CRANE, C.D. III; DALTON, R.; OGLES, J.; TULENKO, J.S.; ZHOU, X. (University of Florida, Gainesville, FL). American Nuclear Society (ANS) Annual Meeting, 10-14 June 1990, Transactions of the American Nuclear Society, Volume 61, CONF-900608--, June 1990, pp. 408-409.

The University of Florida, in cooperation with the Universities of Texas, Tennessee, and Michigan and

Oak Ridge National Laboratory (ORNL), is developing an advanced robotic system for the U.S. Department of Energy under the University Program for Robotics for Advanced Reactors. As part of this program, the University of Florida has been pursuing the development of a faster-than-real-time robotic simulation program for planning and control of mobile robotic operations to ensure the efficient and safe operation of mobile robots in nuclear power plants and other hazardous environments.

1898. REGULATORY RESEARCH AND SUPPORT PROGRAM FOR 1990-91: PROJECT DESCRIPTIONS. INFORMATION BULLETIN NO. 90-2. ANNUAL PUBLICATION. Atomic Energy Control Board, Ottawa, Ontario (Canada), MIC-90-07090, 1990, (206 p).

The Regulatory Research and Support Program is intended to augment and extend the Atomic Energy Control Board's regulatory program beyond the capability of in-house resources. The program covers the safety of nuclear facilities, radioactive waste management, health physics, physical security and the development of regulatory processes. This document contains a list of the projects with a brief description of each, as well as additional supporting information. A list of review panels and funding by mission object is included.

1899. AECB STAFF ANNUAL REPORT ON POINT LEPREAU G.S. FOR THE YEAR 1988. REPORT NO. INFO-0325. ANNUAL PUBLICATION. Atomic Energy Control Board, Ottawa, Ontario (Canada), MIC-90-06890, 1989, (14 p).

Review by AECB staff of the operation of Point Lepreau Nuclear Generating Station during 1988. The review is based on information contained in the documents submitted by the New Brunswick Electric Power Commission and on information gathered by AECB staff via inspections, audits and rounds. The review covers compliance, radiation protection, effluents and environmental monitoring, process systems, performance of special safety systems, significant events, quality assurance, maintenance, chemistry control, management, training, emergency exercises and drills, security, and measures of performance. Significant licensing matters and activities include license renewal, containment leak rate testing, automatic trip of heat transport pumps, follow-up to the Chernobyl accident, and steam and feedwater failures in the turbine building.

1900. ENVIRONMENTAL HARDENING OF ROBOTS FOR NUCLEAR MAINTENANCE AND SURVEILLANCE TASKS. HINTENLANG, D.E.; TULENKO, J.S.; WHEELER, R.; ROY, T. (University of Florida, Gainesville, FL). American Nuclear Society (ANS) Annual Meeting, 10-14 June 1990, Transactions of the American Nuclear Society, Volume 61, CONF-900608--, June 1990, pp. 410-411.

The University of Florida, in cooperation with the Universities of Texas, Tennessee, and Michigan and Oak Ridge National Laboratory, is developing an advanced robotic system for the U.S. Department of Energy under the University Program for Robotics for Advanced Reactors. As part of this program, the University of Florida has been pursuing the development of environmentally hardened components so that autonomous robotic systems can successfully carry out their tasks under the most extreme expected environmental conditions. This requirement means that the designed robotic system with its onboard computer-based intelligence must be able to successfully complete tasks in toxic, radioactive, wet, temperature extremes, and other physically impairing environments. As part of this program, a study was carried out to determine the environmental conditions that should be set as the design criteria for robotic systems to maintain reasonable operations for nuclear plants in the course of maintenance, testing, and surveillance under all conditions, including plant upset. It was decided that Florida would build a combined environmental testing facility to test specific devices in high-radiation/high-temperature combined environments. This environmental test chamber has been built and successfully tested to over 250°F. This facility will provide some of the first combined temperatures/radiation data for many large-scale integrated components.

1901. DOE/NE UNIVERSITY PROGRAM IN ROBOTICS FOR ADVANCED REACTORS RESEARCH. TRIVEDI, M.M., Tennessee University, Dept. of Electrical and Computer Engineering, Knoxville, TN, DOE/NE/37968-T1, 1990, (44 p).

The document presents the bimonthly progress reports published during 1990 regarding the U.S. Department of Energy/NE-sponsored research at the University of Tennessee Knoxville under the DOE Robotics for Advanced Reactors Research Grant. Significant accomplishments are noted in the following areas: development of edge-segment based stereo matching algorithm; vision system integration in the CESAR laboratory; evaluation of algorithms for surface

characterization from range data; comparative study of data fusion techniques; development of architectural framework, software, and graphics environment for sensor-based robots; algorithms for acquiring tactile images from planer surfaces; investigations in geometric model-based robotic manipulation; investigations of non-deterministic approaches to sensor fusion; and evaluation of sensor calibration techniques.

1902. MULTISENSOR-BASED ROBOTIC SYSTEMS FOR INSPECTION AND MANIPULATION TASKS. TRIVEDI, M.M.; ABIDI, M.A.; GONZALEZ, R.C. (University of Tennessee, Knoxville, TN). American Nuclear Society (ANS) Annual Meeting, 10-14 June 1990, Transactions of the American Nuclear Society, Volume 61, CONF-900608--, June 1990, pp. 412-413.

The Advanced Technology Division of the U.S. Department of Energy is sponsoring research at four universities (Florida, Michigan, Tennessee, and Texas) and the Oak Ridge National Laboratory (ORNL) directed toward the development of advanced robotic systems for use in nuclear environments. The primary motivation for using robotic technology in such hazardous environments is to reduce exposure and costs associated with performing tasks such as surveillance, maintenance, and repair. The main focus of research at the University of Tennessee has been to contribute to the development of autonomous inspection and manipulation systems, which utilize a wide array of sensory inputs in controlling the actions of a stationary robot. This involves several important research issues, including selection of sensor modalities, processing and analysis of data acquired by individual sensors, integration of sensory information, control of the robotic arm and end effectors, and efficient computational architectures for implementing various algorithms. Experimental research effort is directed toward design and evaluation of new methodologies using a laboratory-based robotic test bed. A unique feature of this test bed is a multisensor module that is useful in the characterization of the robots work space. In this paper, the authors describe the development of a robotic vision system for automatic spill detection, localization, and cleanup verification and the development of efficient techniques for analyzing range images using a parallel computer. The simulated spill cleanup scenario allows demonstration of the applicability of robotic systems to problems encountered in nuclear environments.

1903. EXTENDED KALMAN FILTERING FOR MODEL-BASED SENSOR FUSION IN ROBOT-

ICS. FUJII, Y.; WEHE, D.K.; LEE, J.C. (University of Michigan, Ann Arbor, MI). American Nuclear Society (ANS) Annual Meeting, 10-14 June 1990, Transactions of the American Nuclear Society, Volume 61, CONF-900608--, June 1990, pp. 411-412.

Remote surveillance and maintenance in advanced nuclear power plants will benefit from the increased utilization of mobile robotic systems. For these robotic systems to function most effectively in hazardous environments, they should be able to make decisions and take necessary actions with minimal human supervision. To accomplish this, the robot must be able to construct an accurate model of the power plant environment from diverse sensory data and a priori maps. In this paper, the authors demonstrate how a recursive parameter estimation technique known as Kalman filtering can integrate noisy data from various sensors to construct a consistent representation of the sensed environment.

1904. STATEMENTS ON THE SAFETY OF THE PLANNED GORLEBEN REPOSITORY FOR RADIOACTIVE WASTE, AND ON THE SAFETY OF NUCLEAR POWER PLANT AND REPROCESSING FACILITIES, PRESENTED ON THE OCCASION OF HEARINGS HELD BY THE BUNDESTAG COMMITTEE FOR THE ENVIRONMENT, NATURE CONSERVATION AND REACTOR SAFETY. GRIMMEL, E.; HAHN, L., Oeko-Institut, Institute fuer Angewandte Oekologie e.V., Published by Freiburg im Breisgau (Germany), Volume 53 of a collective series Werkstattreihe, April 1989, in German, (58 p).

The two contributions in this report deal with statements and opinions of experts relating to a number of issues all concerned with plant or site safety. These issues include ultimate storage of radioactive waste, waste reprocessing, and the nuclear power phaseout. The reservations heard in debates about the safety of the Gorleben repository have recently been aggravated and enlarged by an accident in a mine belonging to currently planned and examined sites. The Wackersdorf reprocessing project, having been abandoned in favor of a cooperation agreement with COGEMA, adds to the topicality of debates. The issues of reactor safety are discussed with a view to the risks and hazards involved in continuing the operation of nuclear power stations and reprocessing facilities, the international risk assessment procedures and debates, and the state-of-the-art in waste management and disposal, which all seem to call for a nuclear power phase-out.

1905. JUDGEMENT OF LUENEBURG HIGHER ADMINISTRATIVE COURT OF JUNE 28, 1989 ON BINDING EFFECTS OF PARTIAL PERMIS- SIONS IN LICENSING PROCEDURES UNDER THE ATOMIC ENERGY LAW. Recht der Elektrizitaetswirtschaft (Germany), Volume 51, Number 5, May 1990, in German, pp. 92-107.

The judgment is concerned with the second partial operation permit for the Brokdorf Nuclear Power Plant. The questions discussed are: the binding effect of previous partial permits; binding effects extending beyond the authorized part; the binding effect of temporary approval for the whole scheme and reservations pronounced to this. In this connection, questions are treated concerning: off-site disposal; limit values under section 45 of the German Radiation Protection Regulation (Strahlen-schutz-verordnung, StrlSchVO); design basis accident plan- ning doses according to section 28, subsection 3, StrlSchVO; the relevance of scientific knowledge of the risks involved in nuclear engineering for the licensibility of an installation; and questions concern- ing radiation exposure through design basis accidents and severe accidents in nuclear installations with regard to the background contamination within the framework of section 45 StrlSchVO.

1906. RADIATION SITUATION IN THE DUKOVANY NUCLEAR POWER PLANT AND ITS ENVIRONS. KOPEC, J.; JOHN, A.; HAKL, J.; LONDYN, P.; ZELENKA, Z. (Jaderna Elektrarna, Dukovany (Czechoslovakia)). Operation and Maintenance of Nuclear Power Plant, Jaderna Elektrarna, Dukovany (Czechoslovakia), 24-26 April 1990, INIS-mf-12796, CONF-9004282--, April 1990, in Czech, pp. 71-74.

In dosimetric monitoring of the radiation situation, particular attention is paid to the activities of the primary coolant circuit and of the steam generator surface blowdown. Gamma dose rates are perma- nently monitored in the working environment. Personnel dosimetry uses film dosimetry comple- mented by thermoluminescence dosimetry. Data are computer-processed. The environmental impacts are monitored by measuring releases into water and air and by determining concentrations of artificial radio- nuclides in environmental samples. Tritium bulk activity increased in the Mohelno Dam and in the Jihlava River. The environmental radiological im- pacts are very low. The radiological protection and safety levels can be improved by improving the material and technological base.

1907. EXPERIENCE WITH IN-SERVICE IN- SPECTIONS AND REPAIRS OF STEAM GEN- ERATORS AND PRESSURIZERS. BEDNAREK, L. (Vedecko Vyzkumny Uhelny Ustav, Ostrava-Radvanice (Czechoslovakia)). Operation and Maintenance of Nuclear Power Plant. Jaderna Elek- trarna, Dukovany (Czechoslovakia), 24-26 April 1990, INIS-mf-12796, CONF-9004282--, April 1990, in Czech, pp. 167-172.

A separate unit responsible for in-service inspections and repairs of nuclear power plants was established in the Vitkovice Steel Works. Decontamination is recommended prior to steam generator repairs. The technology of repairs of degraded sealing surfaces has been mastered. A device is being completed for eddy current testing of thread nests. Some defects have so far not been repairable in Czechoslovakia; for in- stance, several dozens of heat exchange tubes had to be capped for steam generator leaks. A Siemens manipulator has been purchased for tube inspection.

1908. DEVELOPMENT OF A MAINTENANCE SUPPORT SYSTEM FOR NUCLEAR POWER PLANTS. YOKOJIMA, T.; TAKEHARA, T. (To- shiba Corp., Kawasaki, Kanagawa (Japan)). Toshiba Rebyu (Toshiba Review) (Japan), Volume 45, Num- ber 10, October 1990, in Japanese, pp. 782-784.

Proper maintenance management is a key factor in ensuring the stable operation of a nuclear power plant. To optimize plant maintenance, Toshiba has devel- oped a maintenance support system so that personnel involved in maintenance activities can accomplish their work correctly and efficiently. This system provides the information and technical knowledge required by such personnel at each stage of mainte- nance activities. The main elements of the system are the annual inspection management system, radiation control and engineering support system, and equip- ment maintenance information control system.

1909. ENERGY SUPPLY PERSPECTIVES IN THE 1990S. REQUIREMENTS FOR RESEARCH AND DEVELOPMENT IN THE FIELD OF POW- ER PLANT ENGINEERING. HLUBEK, W. (RWE Energie AG, Essen (Germany)). VGB Kraftwerkstechnik (Germany), Volume 70, Number 11, November 1990, in German, pp. 890-894.

In the power plant engineering field, research and development work initiated today determines the future technical structures for a long time in the future. In the field of fossil-fuel-fired power stations, efforts are being made to adhere to existing environ-

mental protection regulations in a more cost-effective way than before, to save fuel by increasing efficiency and also to reduce capital and operating costs even further. In nuclear engineering, the focus of efforts is on the safety of nuclear power stations, environmental and radiation protection and also future disposal concepts. Renewable forms of energy will be developed wherever indications towards significant improvement in economic results are perceived.

1910. DECONTAMINATING METHOD. KUSAKA, K.; SATO, Y., (Toshiba Corp., Kawasaki, Kanagawa (Japan)), Japanese Patent Number JP 2-222899 A, Toshiba Corp., Kawasaki, Kanagawa (Japan), G21F9/28, 5 September 1990, in Japanese, (4 p).

In a decontaminating method for equipments and pipelines for conducting pre-decontamination, oxidative pre-treatment and post-decontamination successively, sulfuric acid is used throughout the entire processing steps and the oxidative treatment is carried out by adding tetravalent cerium ions thereto. Then, after the completion of the oxidative pre-treatment, oxidizing power is eliminated by redox reaction and then the post decontamination is carried out by adding a post-decontaminating agent. Accordingly, since liquid wastes are not formed on every steps and it is not required for changing the kind of acid, the decontaminating procedures can be simplified through out the steps and the amount of liquid wastes formed is decreased. In this way, it is possible to decrease the amount of wastes formed upon decontamination and the wastes themselves can be converted so as to be processed easily, as well as the time required for the decontamination can be shortened.

1911. SAFETY OF NUCLEAR POWER PLANTS WITH PWRS IN THE FRG. OTCENASEK, P., Jaderna Energie (Czechoslovakia), Volume 36, Number 7, July 1990, in Czech, pp. 272-278.

The safety principles of nuclear power plants in the FRG are described in detail. The items discussed include, e.g., the effect of independent separate barriers to prevent radioactivity leak, or steps taken to secure safe reactor shutdown and heat removal, including the building of facilities for this. Attention is also paid to the study of risks, not only at a national level but also within international cooperation.

1912. INFRASTRUCTURE AND REGULATORY CONTROL FOR RADIATION PROTECTION IN THE NUCLEAR INDUSTRY IN INDIA.

KRISHNAMURTHI, T.N.; SOMASUNDARAM, S.; DE, A.K. (Atomic Energy Regulatory Board, Bombay (India)). Radiation Protection Infrastructure. Proceedings Series. International Atomic Energy Agency, Vienna (Austria), International Symposium on Radiation Protection Infrastructure, 7-11 May 1990, CONF-900526--, 1990, pp. 115-129.

The Atomic Energy Regulatory Board (AERB), constituted in November 1983 by the Government of India, carries out certain safety and regulatory functions envisaged under the Atomic Energy Act, 1962. The paper describes the legal provisions, responsibilities, structure and organization of AERB, with special emphasis on radiation protection. The responsibilities of the Board cover nuclear, radiological and industrial safety within the installations of the Department of Atomic Energy and radiological safety in the manifold applications of radioisotopes and radiation sources within the country. An adequate and competent infrastructure has been built to cater to the diverse radiation protection requirements of all nuclear installations in India, arising at different stages from site selection to day-to-day operation. The paper describes the multilevel review which is carried out by the Board prior to issue of authorization for nuclear installations. A number of safety codes, guides and manuals have been prepared to provide specific guidance to design, construction and operating organizations of nuclear power plants. The Regulatory Board receives considerable support from the Health and Safety Group of Bhabha Atomic Research Centre in discharging its responsibility with regard to radiation protection surveillance. The Safety Review Committee of AERB monitors the safety status of all nuclear fuel cycle facilities in operation. In addition, periodic inspections of the installations are carried out. AERB has conducted an in-depth study of specific areas of operational safety of nuclear power plants. AERB ensures that the radiation exposures of workers and radioactive releases to the environment are kept as low as reasonably achievable (ALARA) and that the installations maintain at all times a capability to handle local, site and off-site emergencies.

1913. DECOMMISSIONING AND JOBS. JOHN, B.S. (Arizona State University, Tempe, AZ), Nuclear Decommissioning and Society, Edited by PASQUALETTEI, M.J., Natural Environment - Problems and Management Series. Published by Routledge, London (United Kingdom), 1990, pp. 87-112.

One aspect of the decommissioning web is its effect on socioeconomics, particularly jobs. What will reactor retirement mean to jobs, especially in rural communities where power plant operations may be the most reliable and dominant source of direct and indirect employment in the area. The problems which any plant closure produces for job security are generally understood, but the decommissioning of nuclear power plants is different because of the residual radioactivity and because of the greater isolation of the power plant sites. For example, what will be the specific employment effects of several possible decommissioning scenarios such as immediate dismantlement and delayed dismantlement. The varying effects of decommissioning on jobs is discussed. It is concluded that the decommissioning of nuclear power plants in some areas such as Wales could bring benefits to the surrounding communities.

1914. CONTROL ROOM HABITABILITY SYSTEM REVIEW MODELS. GILPIN, H. (Science Applications International Corp., McLean, VA). Nuclear Regulatory Commission, Div. of Safety Issue Resolution Science Appl., Washington, DC, NUREG/CR-5659, SAIC--90/1054, December 1990, (142 p).

This report provides a method of calculating control room operator doses from postulated reactor accidents and chemical spills as part of the resolution of TMI Action Plan III.D.3.4. The computer codes contained in this report use source concentrations calculated by either TACT5, FPPF, or EXTRAN, and transport them via user-defined flow rates to the control room envelope. The codes compute doses to six organs from up to 150 radionuclides (or 1 toxic chemical) for time steps as short as one second. Supporting codes written in Clipper assist in data entry and manipulation, and graphically display the results of the FORTRAN calculations.

1915. AECB STAFF REVIEW OF PICKERING NGS OPERATIONS FOR THE YEAR 1988. RESEARCH REPORT NO. INFO-0310. ANNUAL PUBLICATION. Atomic Energy Control Board, Ottawa, ON (Canada), MIC-90-01127, 1989, (36 p).

The operation of Pickering Nuclear Generation Station (NGS) A Units 1-4 and Pickering NGS B Units 5-8 are monitored to ensure compliance with licensing requirements by the Atomic Energy Control Board (AECB) Pickering project office staff. This report presents the AECB staff's review of major licensing issues and of the operational performance of Pickering NGS during 1988. The report is limited to those

aspects that AECB staff considers to have particular safety significance, such as the process system, chemistry, performance of special safety systems, reportable significant events, measures of station performance, radiation protection, station effluents and environmental monitoring, quality assurance audits, training, emergency exercises and drills, security, and maintenance and management.

1916. KTA SAFETY RULE: MEASURING AND MONITORING OF GASEOUS AND AEROSOL-BOUND RADIOACTIVE DISCHARGES. Kerntechnischer Ausschuss (KTA), Koeln (Germany), OA-Trans-4048, 1990, (21 p).

(1) To meet the requirements of paragraph 46 section 1 no. 3 of the Radiation Protection Directive, equipment for measuring and monitoring radioactive discharges must be installed and operated in nuclear power stations. (2) The two parts of this rule deal with equipment for monitoring stack air: part 1: "Measurement and monitoring of radioactive matter discharged with stack air during normal operation," part 2: "Measurement and monitoring of radioactive matter discharged with stack air during malfunctions." This part of the rule applies to equipment for measuring and monitoring gaseous and aerosol-bound radioactive matter discharged with stack air from stationary nuclear power stations with light water reactors and high temperature reactors during normal operation.

1917. CONTROL ROD FOR USE IN REACTOR. YASUDA, Y. (Toshiba Corp., Kawasaki, Kanagawa (Japan)), Japanese Patent Number 2-140692 A, Toshiba Corp., Kawasaki, Kanagawa (Japan), May 1990, in Japanese, (4 p).

In a combination of a guide roller, a limit roller and a pin mounted to a reactor control rod, the roller is constituted with bearings. For the material of the bearing, iron-base alloys are used instead of cobalt-base alloy or nickel-base alloy for reducing radiation exposure. By constituting the roller with the bearing, the control rod can be inserted and extracted smoothly. Accordingly, the frictional resistance or the frictional abrasion of the pin of the roller can be reduced and the control rod can be driven stably at high accuracy for a long period of time. Further, since the material of the bearing is made of the iron base alloy, increase of radiation dose due to cobalt 58 or cobalt 60 can be suppressed.

1918. DECONTAMINATION TECHNIQUES FOR PWR POWER GENERATION PLANT. IWAI, M. (Kansai Electric Power Co., Inc., Osaka

(Japan)). Genshiryoku Kogyo (Nuclear Engineering) (Japan), Volume 36, Number 8, August 1990, in Japanese, pp. 34-39.

The report outlines the current trends in development of decontamination techniques to reduce the dose equivalent rate in work environments. Applications of these techniques to PWR power generation plants are also described. The aims of decontamination are briefly explained first. As major measures generally taken in PWR plants, the report outlines features of chemical decontamination of reactor coolant pumps, physical decontamination of the water rooms of steam generators, and chemical decontamination of bypass piping for resistance temperature detectors, focusing on their purposes, experiences with them, structure and material of the components to be decontaminated, decontamination techniques, their mechanisms, their procedures, and their performance. For chemical decontamination, future efforts should include the development of decontamination agents that permit stepwise treatment and reduction in the size of required equipment. For physical decontamination, high-performance hard polishing agents should be developed which would not cause troubles in the primary system.

1919. DECONTAMINATION TECHNIQUES FOR BWR POWER GENERATION PLANT. YOSHIKAWA, S. (Tokyo Electric Power Co., Inc. (Japan)). Genshiryoku Kogyo (Nuclear Engineering) (Japan), Volume 36, Number 8, August 1990, in Japanese, pp. 28-33.

The present report describes various techniques used for decontamination in BWR power generation plants. Objectives and requirements for decontamination in BWR power plants are first discussed focusing on reduction in dose, prevention of spread of contamination, cleaning of work environments, exposure of equipment parts for inspection, re-use of decontaminated resources, and standards for decontamination. Then, the report outlines major physical, chemical and electrochemical decontamination techniques generally used in BWR power generation plants. The physical techniques include suction of deposits in tanks, jet cleaning, particle blast cleaning, ultrasonic cleaning, coating with special paints, and flushing cleaning. The chemical decontamination techniques include the use of organic acids etc. for dissolution of oxidized surface layers and treatment of secondary wastes such as liquids released from primary decontamination processes. Other techniques are used for removal of penetrated contaminants, and soft and hard cladding

in and on equipment and piping that are in direct contact with radioactive materials used in nuclear power generation plants.

1920. RADIATION PROTECTION STANDARDS IN THE UNITED STATES. MILLS, W.A.; FLACK, D.S. (Oak Ridge Associated Universities, Washington, DC), Transactions of the American Nuclear Society, Volume 60, November 1989, Winter Meeting of American Nuclear Society, San Francisco, CA, November 26-30, 1989, CONF-891103-, pp. 529-530.

Standards to protect workers and members of the general public against any harmful effects of ionizing radiation are numerous and complex in the United States. Many federal agencies have protection responsibilities. Congress limits the discretionary authority given to these agencies in providing for this protection, and courts appear at times to render judgments that are illogical to our sense of the degree of radiological protection required. To many, U.S. standards appear to be over-protective in that they have, at best, marginal health benefits and without question are costly to implement. Government agencies, the Congress, industry, professional organizations, and others have expressed such concerns. It is against this background that the Committee on Interagency Radiation Research and Policy Coordination (CIRRPC) undertook a project to enhance its knowledge and understanding of the principal standards in the United States that limit and control radiation exposures. Review of the fact sheets shows no clear intent to be consistent in either the statutory language or the explicit or implicit protection objective of the standard. A more detailed evaluation of U.S. standards seems warranted, especially since the trend is towards lower standards and more restrictive regulations.

1921. PRESENT STATUS AND DEVELOPMENT OF RADIATION PROTECTION STANDARDS. ZI, X. (Academia Sinica, Beijing, (China)). Nuclear Techniques (China), Volume 13, Number 12, December 1990, in Chinese, pp. 708-712.

The present status and development of radiation protection standards, the features of radiation protection in the nuclear energy age and the purposes of complete revision of ICRP publication No 26 by the Commission are described. Some recent debates or fights in this field are also reviewed.

1922. ANSI (AMERICAN NATIONAL STANDARDS INSTITUTE) STANDARDS IN THE AGE

OF RADIATION PROTECTION. DAVIS, J.P. (General Physics Corp., San Diego, CA), Transactions of the American Nuclear Society, Volume 60, November 1989, Winter Meeting of the American Nuclear Society, San Francisco, CA, November 26-30, 1989, CONF-891103-, pp. 530-531.

The American National Standards Institute (ANSI) is the organization responsible for promulgating voluntary consensus standards in a large number of fields. Although nongovernmental, it serves a quasi-official function in representing the United States in international standards activities. ANSI carries out its mandate through liaison with a variety of expert committees, sponsored by professional organizations that recommend consensus standards for adoption as American National Standards in accordance with processes specified by ANSI. Adherence to these processes is important to avoid legal difficulties. Committee N13, sponsored by the Health Physics Society is authorized to recommend consensus standards in the field to ANSI for their approval as American National Standards for Radiation Protection. The committee also reviews existing standards on a 5-year cycle to ensure they are up to date. The goal of Committee N13 is the development of a coherent set of National Radiation Protection Standards. Recent progress and activities are discussed.

1923. RADIATION PROTECTION - HOW SUCCESSFUL ARE WE? GUIMOND, R.J.; O'CONNOR, D. (Environmental Protection Agency, Washington, DC). Transactions of the American Nuclear Society, Volume 60, November 1989, Winter Meeting of the American Nuclear Society, San Francisco, CA, November 26-30, 1989, CONF-891103-, pp. 531-532.

Often scant attention is paid to the context within which standards are set. This is a mistake because it is important to understand that the regulatory process takes place within a complex social milieu. This paper addresses some of the nontechnical concerns that must be considered. The preconditions for regulations, regulatory approaches, and the increasingly lower risk levels applied in other disciplines must be considered. Likewise, the Environmental Protection Agency (EPA) itself is contemplating new directions spawned by public concerns and newly emerging issues.

1924. RADIATION PROTECTION STANDARDS: THE NRC (U.S. NUCLEAR REGULATORY COMMISSION) PERSPECTIVE. CONGEL, F.J. (Nuclear Regulatory Commission,

Washington, DC). Transactions of the American Nuclear Society, Volume 60, November 1989, Winter Meeting of the American Nuclear Society, San Francisco, CA, November 26-30, 1989, CONF-891103-, pp. 532-533.

The purpose of this paper is to summarize the role and viewpoint of the U.S. Nuclear Regulatory Commission (NRC) on radiation protection standards, specifically the major revision of 10CFR20 and the criteria for below regulatory concern. The NRC relies heavily on consensus standards and (especially in radiation protection) the recommendations of authoritative organizations such as the International Commission on Radiological Protection (ICRP), the International Commission on Radiation Units and Measurements, and the National Committee on Radiation Protection and Measurements (NCRP). The NRC is modifying its radiation protection requirements to be consistent with ICRP/NCRP recommendations and to avoid undue controls on trivial risks. The problems are formidable and final resolution of many issues may not occur this calendar year.

1925. A NEW RADIATION PROTECTION DOSE LIMIT FOR OCCUPATIONALLY EXPOSED PERSONS IN THE FEDERAL REPUBLIC OF GERMANY. MARTIGNONI, K.; NITSCHKE, J. (Bundesamt fuer Strahlenschutz, Neuherberg (Germany)). *Radiologe*, Volume 31, Number 5, May 1991, in German, May 1991, pp. 235-239.

In the Federal Republic of Germany, as in most other countries, the dose limit is 50 mSv per year for occupationally exposed persons, as recommended by the ICRP. A reduction in the annual dose limit was recently considered in connection with a discussion of a higher estimation of radiation risk. In the recent amendment of the German Radiological Protection Ordinance, however, the dose limit of 50 mSv per year was retained and additionally a new occupational lifetime dose limit of 400 mSv was introduced. The possible consequences of the 400 mSv dose limit are discussed on the basis of the previous results of dose monitoring in the Federal Republic of Germany.

1926. THE IRPA (INTERNATIONAL RADIATION PROTECTION ASSOCIATION) PERSPECTIVE. LAKEY, J.R.A. (International Radiation Protection Association, London (United Kingdom)) Transactions of the American Nuclear Society, Volume 60, November 1989 Winter Meeting of the American Nuclear Society, San Francisco, CA, November 26-30, 1989, CONF-891103-, p. 528

The development of guidelines, recommendations, standards, and criteria in radiation protection is the task of many government and non-government organizations. These standards are used in all applications of ionizing and nonionizing radiations and also relate to natural sources as well as accidental sources of radiation. The primary purpose of the International Radiation Protection Association (IRPA) is to serve as a medium for international communication and cooperation in radiation protection with the goal of advancing sound and effective radiation protection in all parts of the world. The IRPA is an association of 30 independent associate societies active in 35 countries, and the organization is governed by its 13,000 individual members who elect the general assembly. To encourage this process, IRPA has set up formal links with the international organizations and, for example, occupies an observer position on committees of the International Commission on Radiological Protection (ICRP) and the Nuclear Energy Agency of the Organization for Economic Cooperation and Development (OECD).

1927. RETHINKING BASIC CONCEPTS IN ICRP'S SYSTEM OF DOSE LIMITATION. MILLS, W.A. (Oak Ridge Associated Universities, Washington, DC); MOSSMAN, W.A. (Arizona State University, Tempe, AZ). Implementation of Current NCRP and ICRP Guidance and Revised 10 CFR Part 20: Proceedings of the 24th Midyear Topical Meeting of the Health Physics Society, Raleigh, NC, January 20-25, 1991, CONF-910137-24, pp. 24-31.

The present criterion for radiation protection appears to be exposure reduction rather than adequate protection of health. The 1990 ICRP draft recommendations for a system of dose limitation would further implement this more restrictive criterion by implementing certain academic concepts and assumptions. These concepts and assumptions are discussed and the suggestion is made that the radiation protection community needs to carefully examine the need for the complex system proposed.

1928. THE RADIOLOGICAL SAFETY OF NUCLEAR POWER STATIONS. PASKA, J. (Politechnika Warszawska, Warsaw (Poland)). Gospodarka Paliwami i Energia, Volume 36, Number 2, 1988, in Polish, 1988, pp. 8-12.

Radiological safety of nuclear power engineering is described. The type of dangers and the ways of radiological influence of nuclear power stations and of the objects of nuclear fuel cycle on people are discussed. The admissible doses and the limits of

release of radionuclides in the light of existing regulations and recommendations are given as well as the real value of doses and releases during normal operation conditions. Two greatest breakdowns in nuclear power stations are briefly characterized.

1929. RADIATION PROTECTIVE CLOTHING. IJIRI, Y.; FUGINUMA, T.; ASO, T. (Mitsubishi Cable Industries Ltd., Tokyo (Japan)). Japanese Patent Number JP 3-63598 A, in Japanese, March 19, 1991.

The present invention concerns radiation protective clothing suitable for medical protective clothing, aprons, etc. A primary sheet comprises a lead-incorporated organic polymer layer having a less frictional layer on one side and a contamination-resistant layer on the other side. A secondary sheet comprises a lead incorporated organic polymer layer having a less frictional layer on one side and a comfortable skinfeeling layer on the other side. The less frictional layers of the primary and the secondary layers are laminated so as to be in contact with each other. They are formed so that the comfortable skin-feeling layer of the secondary sheet is on the inside, in other words, on the side of the wearer, and the contamination-resistant layer of the primary sheet is on the outside. With such a combination, although it involves the heavy lead-incorporated organic polymer sheets, it is comfortable to wear because of excellent flexibility and causes less fatigue even when worn for long periods.

1930. NCRP COMMENTS ON RADIATION PROTECTION RELATED REPORTS, PROPOSED STANDARDS, AND MEETINGS. National Council on Radiation Protection and Measurements, Bethesda, MD; U.S. Department of Energy, Washington, DC, DOE/ER/60317-T2, 1991, 5 p.

Two NCRP reports, two annual meeting proceedings and one commentary have been published during this period. NCRP Report No. 105, Radiation Protection for Medical and Allied Health Personnel is a rewrite and update of NCRP Report No. 48, which had the same title and which it supercedes. The primary objective of the new report is to update the material to include new radiation sources used in medicine. NCRP Report No. 106, Limit for Exposure to "Hot Particles" on the Skin was prepared in response to a request from the Nuclear Regulatory Commission. The report addresses the potential biological effect of microscopic radioactive particles on the skin and reviews the presently available information on the subject. Proceedings No. 10, Radon, is the

proceedings of the 24th Annual Meeting of the NCRP. Proceedings No. 11, Radiation Protection Today - The NCRP at Sixty Years, is the proceedings of the 25th Annual Meeting of the NCRP. Commentary No. 5, Review of the Publication, Living Without Landfills, was recently released. Also included is a list of NCRP committee reports which were released in the current reporting period.

1931. NATIONAL COUNCIL ON RADIATION PROTECTION AND MEASUREMENT SEMI-ANNUAL TECHNICAL PROGRESS REPORT. MARCH 1989-AUGUST 1989. NEY, W.R. (National Council on Radiation Protection and Measurements, Bethesda, MD), U.S. DOE, Washington, DC, DOE/ER/60317-T3, 1991, 13 p.

The semi-annual technical progress report is for the period 1 March 1989 to 31 August 1989. This National Council on Radiation Protection and Measurements (NCRP) program is designed to provide recommendations for radiation protection based on scientific principles. During this period several reports were published covering the topics of occupational radiation exposure, medical exposure, radon control, dosimetry, and radiation protection standards. Accomplishments of various committees are also reported; including the committees on dental X-ray protection, radiation safety in uranium mining and milling, ALARA, instrumentation, records maintenance, occupational exposures of medical personnel, emergency planning and others.

1932. A SUMMARY OF THE 1990 RECOMMENDATIONS OF ICRP. SMITH, H. (International Commission on Radiological Protection, Didcot (United Kingdom)), Nuclear Engineering International (United Kingdom), January 1992, pp. 2-4.

Since publication of the 1977 Recommendations, the Commission has periodically reviewed the biological basis of its recommendations and its method of defining the human health detriment associated with exposure to ionizing radiations. It is now convinced that the risks of radiation-induced cancer are greater by a factor of 3 or 4 than those assumed in 1977. Thus it is necessary to consider the appropriateness of the currently recommended annual dose limits as part of the proposed System of Radiological Protection. The system of dose limitation recommended in 1977 was intended to be applied to controllable activities in normal operations. The Commission now wishes to extend its advice to clarify its position in those situations where exposures are not certain to

occur; and to define the principles of protection necessary to reduce dose in situations where the source of radiation is not under control. The 1990 Recommendations are intended to be of help to regulatory and advisory agencies and to management bodies and their professional staff. They deal only with ionizing radiations and with the protection of man.

1933. THE NEW ICRP RECOMMENDATIONS: THE EUROPEAN DIMENSION. LUYKX, F. (Commission of the European Communities (Luxembourg)), Journal of Radiological Protection (United Kingdom), Volume 12, Number 1, March 1992, pp. 29-32.

Currently the EC Commission is preparing a new version of the Basic Safety Standards to take into account the 1990 recommendations of ICRP and also the new conditions which will be created at the end of 1992 by the completion of the EC internal Common Market. The main items for which changes are proposed are discussed in this note.

1934. RADIOLOGICAL DECISION AIDING USING MULTI-ATTRIBUTE ANALYSIS TECHNIQUES. AUMONIER, S.; BARRACLOUGH, I.M. (National Radiological Protection Board, Chilton (United Kingdom)), Radiological Protection Bulletin (United Kingdom), Volume 119, February 1991, pp. 23-26

Multi-attribute analysis techniques for aiding complex decision making and their use in the optimization of radiological protection problems is briefly discussed. The analysis begins with the identification of the set of options or available solutions to the problem. Options are compared by evaluating their performance with reference to a number of factors or attributes which are relevant to the decision. In order that options may be compared, scores are assessed in terms of how valued each score is to the decision maker. Changes in preference along a particular scale, or value functions, vary between individuals or preference groups. Weighting the values associated with each attribute then leads to the preferred option.

1935. 1990 RECOMMENDATIONS OF THE INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION ADOPTED BY THE COMMISSION IN NOVEMBER 1990. ANNALS OF THE ICRP. (International Commission on Radiological Protection, Sutton (United Kingdom)), ICRP Publication 60, Published by Oxford (United Kingdom) Pergamon Press, 1991, 208 p.

This publication represents a completely new set of basic recommendations, outdating ICRP Publication 26. They aim to: a) take account of new biological information and of trends in setting safety standards; b) improve the presentation of the recommendations; c) maintain as much stability in the recommendations as is consistent with the new information. The recommendations cover quantities, biological aspects, the conceptual framework of radiological protection, proposed and continuing practices (occupational, medical and public exposure) and implementation of the Commission's Recommendations.

1936. ALARA IN INDUSTRY. CROFT, J.R. (National Radiological Protection Board, Leeds (United Kingdom)), Radiation Protection Optimization. Advances in Practical Implementation, 3rd European Scientific Seminar, Madrid, Spain, 12-14 September 1988, EUR-12469, CONF-8809506-, pp. 259-277.

This paper reviews how the general frameworks of the ALARA procedures, ALARA Audits and Predictive ALARA Plans can be used in the non-nuclear power sector to give practical effect to the concept of optimization. The development of the use of ALARA Audits as an integral part of a Radiation Protection Adviser service and our experience of this structured approach is described. Particular attention is drawn to the use of site specific reference levels to trigger graded levels of response to increasing doses. Also several examples of the use of Cost Benefit Analysis within the ALARA Procedure are briefly described.

1937. OPTIMIZATION AND DECISION-MAKING IN RADIOLOGICAL PROTECTION. (International Commission on Radiological Protection, Sutton (United Kingdom)), Published by Oxford (United Kingdom) Pergamon Press, ICRP Publication No. 55, 1989, 69 p.

A report of a Task Group of Committee 4 of the ICRP on "Optimization and Decision-Making in Radiological Protection" is given. Topics covered include optimization of protection, the optimization procedure applied to design, the optimization procedure applied to operations, quantitative decision-aiding techniques, quantification of the cost of protection efforts and dose distributions and the application of the optimization procedure.

1938. ALARA - A GENERAL PERSPECTIVE FROM THE UKAEA. Edited by BROWN, M.L.; BLACKMAN, T.E.; JONES, P.M.S.; MC KEAGUE, R.; LAKEY, J.R.A.; LEWINS, J.D., (UKAEA Safety

and Reliability Directorate, Culcheth (United Kingdom)), ALARA: Principles, Practice and Consequences, Published by Adam Hilger, Bristol, England, Proceedings of the Symposium on ALARA - Quantitative Optimization Techniques for Radiation Protection in the Nuclear Industry, London, United Kingdom, 18 September 1986, pp. 82-103.

The UKAEA has to apply ALARA in many different situations including operation, maintenance, refurbishment or plant modification, waste management, decommissioning and new plant design. Some aspects of the UKAEA's experience of ALARA are described with particular reference to the design and modification of process plant and the operation of Winfrith SGHWR reactor. The application of ALARA to accidents involving the release of radioisotopes is also discussed together with a brief comment on the implications of the use of cost-benefit approaches in the implementation of ALARA.

1939. ALARA/ALARP: WORKING WELL BEFORE ICRP 60. CRIPPS, S. (National Nuclear Corp. Ltd., Knutsford (United Kingdom)), Nuclear Engineering International, Volume 36, Number 448, November 1991, pp. 31-33.

The 1991 British Nuclear Energy Society conference on Occupational Radiation Exposure is reported. The papers presented suggest that the exposure criterion, ALARA/ALARP (as low as reasonably achievable and/or practicable) appears to be working well, even before national regulations reflect the new International Commission on Radiological Protection (ICRP) recommendations. Increasing reliance is being placed on integrated management systems for controlling exposure.

1940. KEY ISSUES IN THE IMPLEMENTATION OF ALARA IN OPERATIONS. LOCHARD, J. (Centre d'Etude sur l'Evaluation de la Protection dans le Domaine Nucleaire, Fontenay-aux-Roses (France)); CROFT, J.R. (National Radiological Protection Board, Leeds (United Kingdom)), Journal of Radiological Protection (United Kingdom), Volume 8, Number 2, June 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.

1941. CASE OF ALARA: AN INCIDENT AT SELLAFIELD, NOVEMBER 1983. CHARLESWORTH, F.R., Journal of the Society of Radiological Protection, Volume 7, Number 1, Spring 1987, pp. 5-14.

The principle of ALARA has long been an important concept to health physicists in their work in radiological protection. However, there is seldom an opportunity to test their judgement and advice against public opinion. Such an opportunity was provided in June 1985 when a trial was held before a jury, as representatives of the public, in the Crown Court in Carlisle in Cumbria. One of the six charges brought by the Crown against British Nuclear Fuels Ltd. alleged a breach of an ALARA condition in respect of a discharge of radioactive liquids into the Irish Sea in November 1983. The events leading to the discharges, and the principal points raised in the case are described. Some comments on the significance of the verdicts to health physicists and the nuclear industry are presented.

1942. SOCIETY'S VALUATION OF LIFE SAVING IN RADIATION PROTECTION AND OTHER CONTEXTS. COHEN, B.L. (Pittsburgh University, PA; Argonne National Laboratory, IL), Health Physics, Volume 38, Number 1, January 1980, pp. 33-51.

Various situations are described in which societal action may be interpreted as a dollar value placed on averting a human fatality, and numerical values are derived in each case. Situations included are a variety of medical screening and medical care programs and of automobile and highway safety measures, food for overseas relief, air pollution control, fire prevention, industrial safety, and several radiation-related activities including standards for radium in drinking water, radwaste systems in nuclear plants, and defence and civilian high-level waste management. Values varying from a few thousand dollars to hundreds of million of dollars per fatality averted are obtained. An attempt to derive data of this type from polling is described. The problem of discounting when money is spent now to save lives far in the future (as with nuclear waste) is discussed. It is concluded that nearly all of the vast variation in the results is unjustified and represents a need for educating the public, especially in the area of radiation protection.

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