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

Prepared by T. A. Khan, H. Tan, J. W. Baum, B. J. Dionne

Brookhaven National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

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PREFACE

One of the functions of the ALARA Center is to collect and disseminate information on dose reduction at nuclear power plants. This is the fifth report in the series of bibliographies of selected readings in radiation protection and ALARA that the Center publishes periodically. The abstracts in this bibliography were selected from proceedings of technical meetings, journals, research reports, searches of information data bases and reprints of published articles provided to us by the authors. The abstracts relate in one way or another to dose reduction at nuclear power plants, whether it is through good water chemistry, improvements in nuclear materials, better control of corrosion, robotics and remote tooling or good operational health physics.

The report contains 278 abstracts. Subject and author indices are provided. The subject index covers all previous volumes in this series. The material from the present volume is shown in boldface in the subject index.

The information presented here is also available on our on-line service, which provides up-to-date information. Apart from the bibliography, a number of other databases are available on the service. Among these databases are RESEARCH, which contains information on research projects which have a bearing on dose reduction, HPTECH, which covers operational health physics at nuclear power plants, and NEWS, which gives brief information on what is happening internationally in the area of ALARA at nuclear power plants. These databases may be accessed by means of a personal computer and a modem. Passwords and a manual on how to access the user-friendly system may be obtained from the ALARA Center. Access to the on-line system is given to organizations that provide information for one or more of the databases.

ACKNOWLEDGMENTS

We would like to thank Alan K. Roecklein of the NRC and Charles B. Meinhold of Brookhaven National Laboratory for their support and encouragement. We wish to acknowledge the fine software support provided by Stuart W. Daniel for this series of reports. We thank our secretary Maria Beckman and our librarian Sandra Lane for their assistance in producing this volume.

ABSTRACTS

1129. INDUSTRIAL SAFETY IN POWER PLANTS (in German). VGB Technische Vereinigung der Grosskraftwerksbetreiber e.V. (Essen, Germany, F. R.), Jan 21 1987, 164 pp.

The proceedings of the VGB conference 'Industrial safety in power plants' held in the Gruga-Halle, Essen on January 21 and 22, 1987, contain the papers reporting on: Management responsibility for and legal consequences of industrial safety; VBG 2.0 Industrial Accident Prevention Regulation and the power plant operator; Operational experience gained with wet-type flue gas desulphurization systems; Flue gas desulphurization systems: Industrial-safety-related requirements to be met in planning and operation; the effects of the Hazardous Substances Ordinance on power plant operation; Occupational health aspects of heat-exposed jobs in power plants; Regulations of the Industrial Accident Insurance Associations concerning heat-exposed jobs and industrial medical practice; The new VBG 30 Accident Prevention Regulation 'Nuclear power plants'; Industrial safety in nuclear power plants; safe working on and within containers and confined spaces; Application of respiratory protection equipment in power plants.

1130. INDUSTRIAL SAFETY PRACTICE IN NUCLEAR POWER PLANTS (in German). K.H. SPANGENBERG (Kernkraftwerk Obrigheim G.m. b.H. (Germany, F.R.)) VGB Technische Vereinigung der Grosskraftwerksbetreiber e.V., Essen (Germany, F.R.) ; Kernkraftwerk Obrigheim G.m.b.H.. Proceedings of a Symposium - Symposium for wind power technology and the application of wind energy, Hamburg, DE, Jan 1987, pp. 1-5.

The author focusses on industrial safety practice in the field of radiation protection while quoting examples from the Obrigheim NPP, like shielding wall between steam generator and main coolant pump, shielding of the concentrate reheater of the evaporator system, shielding cover for the main coolant pump housing and shielding of the reactor pressure vessel. In addition, he describes the machining of the sealing surfaces of the flanged connections of pressure nozzles for control rod drives.

1131. OVERVIEW OF 'BASIC SAFETY PRINCIPLES FOR NUCLEAR POWER PLANTS' (in Japanese). MASAO NOZAWA and KUNIHISA SODA (Japan Atomic Energy Research Inst., Tokai, Ibaraki. Tokai Research Establishment). Nippon Genshiryoku Gakkaishi (Japan), Vol. 30, Oct 1988, pp. 889-896.

International Nuclear Safety Advisory Group (INSAG) was established in 1985 within IAEA to discuss the matters related to and to exchange informations concerning nuclear safety. INSAG has published several reports based on their activities, one of which was a summary report on the Post-accident Review Meeting on the Chernobyl Accident. 'Basic Safety Principles for Nuclear Power Plants (BSP)' is the third report of the series. BSP summarizes the safety principles which are to be followed by all of personnels working in nuclear community including designers and owners of the plants and regulators. Special emphasis is given to the concept of 'Safety Culture' which forms the basis of assuring the safety of the plant. Severe accident is also given a special attention since an accident leading to release of a large amount of radioactive materials must be prevented and measures to mitigate such accident are provided. The background and objectives of BSP are explained in the present document.

1132. DECONTAMINATION TECHNIQUES AND THEIR APPLICATION (in German). H. WILLE and H.P. BERTHOLDT (Siemens A.G. Unternehmensbereich KWU, Erlangen (Germany, F.R.). Fachbereich Zentrale Entwicklung und Technik). ATW, Atomwirtschaft, Atomtechnik (F.R. Germany), Vol. 34, Feb 1989, pp. 81-83.

The decontamination of components and systems in nuclear power plants is becoming more and more important in inspection and repair work. The local dose rates in nuclear power plants, which rise as a function of the length of operation, make decontamination an important initial method in reducing personnel exposure doses. Decontamination techniques are defined not only on the basis of the time available and the decontamination factor to be achieved, but also in the light of the waste produced and the way in which it can be further processed. Using decontamination techniques with low chemicals concentrations can greatly reduce waste arisings. This was proved, for instance, in the further development of the wellknown chemical MOPAC process into the MOPAC 88 process, in the course of which waste arisings were cut back to 5-10% of the original volume.

1133. **REDUCTION OF RADIATION EXPOSURE AND RADIOACTIVE WASTES ON KARIWA KASHIWAZAKI UNIT 1 (in Japanese).** TACHIMORI OHBA, TADAMASA YAMAZAKI, YOSHIO SUNAMI, and TOSHIO TAMAI (Tokyo Electric Power Co., Inc. (Japan)). Karyoku Genshiryoku Hatsuden (Japan), Vol. 39, Aug 1988, pp. 873-881.

The first regular inspection of Kashiwazaki Kariwa Unit 1 (K-1) was completed. The total radiation exposure during the inspection was 18.7 person center dot rem and the amount of wastes generated was 473 200 l-drums. These figures are far less than generally expected. Various measures were taken to ensure appropriate design,plant operation management and regular inspection. Special measures have also been taken to reduce the amount of wastes. A great part of the radiation exposure is attributed to the primary system and equipment. Radiation sources in the primary system are roughly divided into two groups: substitution type and deposit type. 'Pre-filming' was performed to produce oxide film over the inner walls of the piping under the condition where the reactor water was free from radioactive substances. This was carried out by first charging the fuel for trial operation, and then operating the reactor and primary system at the rated temperature and pressure for about two weeks before the start of heating by nuclear fuel. Operation is being performed with the ratio of nickel to iron crud in water maintained below 0.2 to control the Co concentration. Various devices have been installed to reduce radiation exposure during regular inspection."

1134. **PLANT MONITORING DEVICE (in Japanese).** HITOSHI MIYAHARA (Toshiba Engineering Co. Ltd., Tokyo (Japan)). Toshiba Corp., Kawasaki, Kanagawa (Japan); Toshiba Engineering Co. Ltd., Tokyo (Japan), Jun 27 1988, 5 pp.

Purpose: To properly and rapidly take countermeasures depending on the situations by totally monitoring atmospheres in a power plant. Constitution: A device for totally monitoring atmospheres in a nuclear power plant has not yet been known. For instance, temperature and humidity are measured only within a required range for the monitor and the control of building ventilation and air conditioning facilities, while radioactive rays are measured only for the atmosphere to the limited areas by area radiation monitors. According to the present invention, plant atmospheres are measured respectively by a plurality of measuring means disposed at each of the positions in the plant. These measurement results are subjected to calculation processing and then displayed. Accordingly, the atmospheres in the plant can be recognized totally by monitoring the contents of the display. Thus, it is possible for early finding of abnormality in the operation and saving patrolling labors of operators in the nuclear power plant.

1135. **THREE DECADES OF NUCLEAR POWER SAFETY.** J.K. WRIGHT (Central Electricity Generating Board, London (UK). Nuclear Health and Safety Dept.). Safety Practitioner (UK), Vol. 6, Dec 1988, pp. 11-17.

The change over the last 30 years in safety management and relations with the licensing authority is explained. During that time the reactor design has changed from Magnox, to Advanced Gas Cooled Reactors, to the first Pressurized Water Reactor to be built in Britain. The following topics are discussed in turn: the licensing process, the development of standards, developments in technology, the man-machine interface and how to deal with ageing plant. The safety of the reactors is judged by the risks to the public from radiation releases. Over the past 30 years a vast amount of experience has been gained and, together with the advances in technology, will enable identification of safety gains in the future to be achieved cost effectively. The risks are assessed in the context of the principle of 'As Low as Reasonably Practicable' and the document 'Tolerability of Risk from Nuclear Power Stations'.

1136. **METHODS FOR IMPROVING WATER CHEMISTRY AT FOREIGN NPPS WITH LWRS (in Russian).** V.N. BELOUS and KARAKHAN'YAN, L.N. (comps.). Atomnaya Tekhnika za Rubezhom (USSR), Jun 1988, pp. 19-24.

Methods for improving water chemistry (WC) at foreign NPPs with LWRs are described. The methods are aimed at radiation situation improvement, prevention of intercrystalline corrosion crack propagation in 304 steel under strains. WC with constant high-temperature pH value at the level of 7.3-7.4 is recommended for NPPs with PWRs. For NPPs with BWRs it is recommended to add zinc (5-15 mg/kg) to reactor water and hydrogen-to feed water.

1137. NONDESTRUCTIVE TESTING IN SERVICE - STATE OF DEVELOPMENT AND PRACTICAL EXAMPLES (in German). K. FISCHER (Siemens A.G. Unternehmensbereich KWU, Erlangen (Germany, F.R.). Geschaeftsbereich Nukleare Energieerzeugung Inland, Service, Montage). ATW, Atomwirtschaft, Atomtechnik (F.R. Germany), Vol. 34, Feb 1989, pp. 77-79.

The nondestructive in-service inspections of the primary system, which are important in revision procedures and for the safety of nuclear power plants, employ mechanized ultrasonic and eddy current tests. New manipulators allow reactor pressure vessels and steam generators to be inspected faster and with less radiation exposure. New inspection equipment and faster data processing systems make for further reduced inspection times, clearer inspection findings, and reduced inspection expenditure.

1138. EXPERIMENTAL EVALUATION OF COBALT BEHAVIOR ON BWR FUEL ROD SURFACE. H. KARASAWA, Y. ASAKURA, M. SAKAGAMI, and S. UCHIDA (Energy Research Lab., Hitachi Ltd., 1168 Moriyama, Hitachi Ibaraki 316 (JP)). Corrosion (Houston) (USA), Vol. 44, Jun 1988, pp. 371-375.

Cobalt behavior on the boiling water reactor (BWR) fuel rod surface was experimentally evaluated at 285 C and with various pH values. Adsorption of cobalt ions on hematite particles proceeded via the exchange reaction of cobalt ion with the surface hydroxyl of the hematite. The equilibrium constant for the adsorption at 285 °C was found to be ~ 570 times as large as that at 20 °C. The adsorbate formed cobalt ferrite at the rate of 3.4×10^{-2} g-Co adsorbed/h. The dissolution rates of cobalt ferrite and cobalt oxide particles were found to depend on $H^{-1.1}$ and $H^{-1.2}$,respectively, where H^{+} means the H^{+} concentration. Cobalt ions were released from these oxides when O^{2-} ions in them combined with two aqueous protons to form water at the oxide-water interface. Cobalt behavior on the fuel rod surface under BWR conditions was discussed using the experimental results.

1139. PRIMARY WATER CHEMISTRY CONTROL AND EXPERIENCE IN JAPANESE PWR. WATER CHEMISTRY FOR CONTROL OF RADIOACTIVE CORROSION PRODUCT.

KOZO HOSHIKAWA (Shikoku Electric Power Co., Inc., Takamatsu (Japan)) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP, Apr 13 1988, v. 1.

The control of the radioactive corrosion product is an essential countermeasure for radiation exposure reduction. In this paper, primary water chemistry control was discussed at the coolant chemistry conditions of power operation and shutdown operation based on the basic studies and plant survey. And our present water chemistry control has been recognized as effective for the reduction of radiation level. Recent topics relating to the crud bahavior in plant were also presented.

1140. OPERATIONAL EXPERIENCE OF PWR PLANTS IN JAPAN. HISTORY OF WATER CHEMISTRY. TETSUJI KISHIDA (Kansai Electric Power Co., Inc., Osaka (Japan)) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP, Apr 13 1988, v. 1.

No abstract available.

1141. STATUS OF ZINC INJECTION PASSIVATION AT U.S. BWRS. W.J. MARBLE and R.L. COWAN (General Electric Co., San Jose, CA (USA)) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP, Apr 13 1988, v. 1.

Over the past several years, operating plant correlations and laboratory experiments have shown that the presence of trace quantities of soluble zinc in the Boiling Water Reactor (BWR) reactor water will suppress Co-60 buildup on recirculation system piping. This paper reports on the first year of zinc injection at Hope Creek, a new reactor, and presents the results of an initial mid-cycle gamma scanshowing average dose rates a factor of 2 to 3 lower than would have been expected without zinc present. It also presents the results of a successful demonstration test and subsequent permanent implementation at Millstone Pt. 1, a mature reactor. Data from these two reactors suggest possible unanticipated benefits from zinc injection in . . areas of reactor water soluble Co-60 reduction and lower curies of Co-60 in waste shipments.

1142. OPERATION EXPERIENCES OF ONAGAWA UNIT 1 FOR REDUCTION OF RADIATION SOURCES. Y. SUTOH, N. SAKATA, K. KOBAYASHI, M. KOBAYASHI, T. FUKUSHIMA, and K. YAMAZAKI (Tohoku Electric Power Co. Ltd., Sendai (Japan)) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium-21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP, Apr 13 1988, v. 1.

In order to control radiation exposure in nuclear power reactors, measures should be taken at design phase, construction and startup test phases and operational phase. Based on operation experiences of many other BWRs,latest technologies concerning radiation reduction program were adopted in Onagawa Nuclear Power Station unit-1, 524 MWe BWR, of Tohoku Electric Company Ltd. As the result of these measures, radiation exposures were 70, 67 and 48 man-rem in first three annual inspection outages. Water chemistry experiences are described along with design improvements and operating practices.

1143. CONTINUOUS ON-LINE ANALYSIS OF SOLUBLE IMPURITIES IN PWR PRIMARY COOLANT BY AUTOMATED ION CHROMATOGRAPHY. M.D.H. AMEY and G.R. BROWN (UKAEA Atomic Energy Establishment, Winfrith). Proceedings of a Symposium - International conference on ion exchange processes (ION-EX '87), Wrexham, GB, Apr 13 1987, pp. 180-187.

Two ion chromatographs have been developed and evaluated for the measurement of soluble transition metals, particularly cobalt, in PWR primary coolant. Complete automation has been achieved, including automatic injection of standards and analysis from multi-sampling points together with continuous and unattended operation. Data processing of the chromatograms and automatic control has been possible by interfacing the instruments to microprocessor control systems. An absolute detection limit of 0.1 ng cobalt has been achieved which, with an on-line sample preconcentration (100 ml), has allowed measurements to be made down to the one part per trillion level (0.001 ppb). Experience with the long term stability and performance under continuous operation to monitor experimental PWR loops and operating reactors is reviewed and further development identified.

1144. CANDU WATER CHEMISTRY. 200 REACTOR YEARS OF EXPERIENCE. D. BARBER and J.P. VAN BERLO (Atomic Energy of Canada Ltd., Sheridan Park, Ontario (Canada). CANDU Operations) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP, Apr 13 1988, v. 1.

There are currently 20 CANDU nuclear power plants with electrical capacities greater than 500 MW(e), operating worldwide. Their net capacity is close to 13,000 MW(e). A further nine plants are under construction. When these plants are completed the total net capacity will be just over 19,500 MW(e). In addition, AECL has been involved in the design, construction and early operation of four PHWR plants having a total electrical output of about 750 MW(e). This paper presents the results of 30 years of development of CANDU water chemistry from that used for the 25 MW(e) Nuclear Power Demonstration (NPD) plant to the water chemistry requirements for plants currently being designed. The combination of improved heat transport system water chemistry and construction materials has resulted in significant reductions in in-plant radiation fields and collective operator doses. An extensive joint AECL/OH research and development program which began in 1971 on understanding the production transport and removal of radioactive materials played a significant role in this achievement. The impact of the environmental releases from CANDU plants located on a typical inland site in Canada meet the Canadian regulatory requirement of not exceeding one percent of 5 mSv/a (500 mrem/a). Typical releases from individual CANDU 600 plants contribute a radiation dose to the critical member of public of less than 15 μSv/a (1.5 mrem/a).

1145. EXPOSURE RATE AND DOSE TRENDS IN WESTINGHOUSE-DESIGNED PLANTS AND TECHNIQUES TO REDUCE THEM. MANN, C.A. BERG and F.L. LAU (Westinghouse Electric Corp., Pittsburgh, PA (USA)) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP, Apr 13 1988, v. 1.

Since 1979, Westinghouse Electric Corporation has conducted a radiation control program with the goal of reducing plant personnel exposure. As part of this program, plant design features, materials of construction, and operational techniques have been evaluated for their effect on radiation fields in and around PWR components. As a result of work performed during the last 10 years, the following conclusions can be made regarding trends of exposure rates and plant doses in Westinghouse-designed plants and the development of techniques to reduce them. The long-term exposure rate trends of the steam generator channel head can be grouped into three broad categories. In all of them, there is a decrease in exposure rates after about five EFPYs (effective full-power years) of operation. The short-term generator exposure rate trend in plants also appears to be divisible into three similar groups. Within a few EFPYs, the exposure rate in the steam generators after replacement will reach at least the same level as before replacement. The absolute level appears to be a function of whether or not the channel head bowl was decontaminated. There is a direct relationship between steam generator channel head exposure rates and plant doses. In spite of the decreasing exposure rate trend, plant annual doses are still increasing. This indicates a possible need for complete system decontamination.

1146. DEPOSITION OF CRUD IN BWR WATER ON VARIOUS STEELS EXPOSED IN THE DODEWAARD NUCLEAR POWER PLANT. W.M.M. HUIJBREGTS and P.J.C. LETSCHERT (Keuring van Elektrotechnische Materialen N.V., Arnhem (Netherlands)) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP, Apr 13 1988, v. 2.

A rack composed of samples of different materials with various surface treatments was exposed in the reactor vessel of the GKN boiling-water reactor (BWR) in Dodewaard for one reactor cycle in order to study the CRUD deposition. Ferritic, austenitic and ferritic-austenitic steels and Inconel 600 were mounted in the rack. The ^{60}Co and ^{54}Mn activity of the CRUD and of the adherent oxide layer were measured. A correlation was found between deposited CRUD and the ^{60}Co activity in the adherent layer. An extensive report of this research has been published .

1147. DEPOSITION BEHAVIOR OF CORROSION PRODUCTS ON FUEL SURFACE IN SIMULATED LOW CRUD BWR CONDITION. Y. URUMA, K. YAMAZAKI, T. BABA, H. NAGAO, K. SHI-MIZU, and S. YOSHIKAWA (Nippon Atomic Industry Group Co. Ltd., Kawasaki, Kanagawa. Nuclear Research Lab.) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP, Apr 13 1988, v. 2.

Intentional iron crud injections were carried out in recent Japanese BWRs aiming at suppression of ionic ^{58}Co ^{60}Co increases at early stage of plant operations. Drastical reductions of these species were successfully observed and these reductions had been predicted under the explanation of chemical interaction between injected iron crud and nickel ion on heating fuel claddings. Loop experiment was conducted to demonstrate these assumptions. In order to simulate the actual plant phenomenon regarding iron crud injection, fundamental model equations are proposed based on experimental results.

1148. COOLANT CHEMISTRY STUDIES AT THE BELGIAN PWRS, DOEL 3 AND DOEL 4. G.C.W. COMLEY, P. CAMPION, K. DE RANTER, and R. ROOFTHOOFT (UKAEA Atomic Energy Establishment, Winfrith) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP, Apr 13 1988, v. 2.

Collaborative studies of coolant chemistry on the Doel 3 and 4 reactors have benefitted from standardised sampling equipment to provide self-consistent and integrated data. Modern methods of analyses have been applied for the in-line assessment of particulates in coolant and soluble corrosion products. Refined techniques for the measurement of soluble and insoluble active species have provided useful comparative data at Doel 3 and 4. While our analyses of the data suggest that the soluble component of coolant-borne activity is most important in setting early doserate trends around the external coolant circuit, the contribution of particulates cannot be ignore in the longer-term. The properties and composition of particles in suspension in the coolant have been explored in depth during steady reactor operation, during transients and at shutdown and reactor start-up. At similar stages of operation our studies have covered the role of all metallic and active species carried by the coolant. In two similar reactor systems where the main variables were coolant pH_T and Zircaloy or Inconel 718 gridded fuel, the advantages of higher alkalinity and the absence of an in-core component source of ^{60}Co on external doserates were most evident. This work was funded by the CEGB in the UK.

1149. DECONTAMINATION AND MATERIALS CORROSION CONCERNS IN THE BWR. B.M. GORDON and G.M. GORDON (General Electric Co., San Jose, CA (USA)) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium-21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP, Apr 13 1988, v. 2.

The qualification of chemical decontamination processes to decontaminate complete systems or individual components in essential if effective inspection, maintenance, repair or replacement of plant components is to be achieved with minimum exposure of workers to ionizing radiation. However, it is critical that the benefits of decontamination processes are not overshadowed by deleterious materials/ corrosion side effects during the application of the process or during subsequent operation. This paper discusses such potential corrosion/materials problems in the BWR and presents relevant available corrosion data for the various commercial decontamination processes.

1150. MONITORING OF RADIOACTIVITY IN AIR IN ROOMS OF NUCLEAR POWER STATIONS (in German). Ueberwachung der Radioaktivitaet in der Raumluft von Kernkraftwerken. T. 2. Kernkraftwerke mit Hochtemperaturreaktor. Entwurf. Kerntechnischer Ausschuss (KTA), Koeln (Germany, F.R.). Heymanns (Koeln, Germany, F.R) , 1988, 20 pp.

The monitoring of radioactive materials in the air of rooms with correct operation should make a contribution to fulfilling the requirements of Para. 28 section 1 and Para. 46 section 1 no. 2 of StrlSchV, by: (a) Automatically emitting signals if warning thresholds are exceeded, in order to recognise increased activity concentrations in the air and to introduce the necessary measures. (b) Identifying the group of rooms concerned, in which increased activity concentration led to a rise of activity in the exhaust air from the chimney. (c) Giving notice of leaking systems or components, which carry radioactive materials (monitoring leakage of parts of plant). (d) Detecting increased concentrations in the air with regard to protecting the personnel. The devices required for the tasks are divided into: Solidly fixed measuring devices and sample collectors, particularly for monitoring radioactive materials in the air of waste air ducts and mobile measuring equipment and sample collectors for monitoring the air in rooms, particularly with regard to the radiation protection monitoring of workplaces.

1151. CERTIFICATION OF RADIATION PROTECTION INSPECTOR QUALIFICATION (in Portuguese). Comissao Nacional de Energia Nuclear, Rio de Janeiro, RJ. CNEN (Rio de Janeiro, Brazil) , 1988, 10 pp.

The norm which establishes the requirements related to the certification of radiation protection inspector qualification is presented. The norm is applied to physical person candidate, or holding the office, to the certification of radiation protection inspector qualification in radiactive installations or nuclear installations or transport of radioactive materials.

1152. TMI-2 SHOWS THE BENEFITS OF AN INNOVATIVE APPROACH [TO REACTOR MAINTENANCE]. Nuclear Engineering International (UK), Vol. 33, Oct 1988, pp. 14-15.

Innovative approaches developed in the cleanup of Three Mile Island 2 reactor are outlined. Whilst the work at TMI-2 is largely related to decontamination and defueling, many of the practices are directly applicable to maintaining an operating plant. Techniques briefly covered include: remote supervision; use of training mock-ups; heat stress control; and battery powered respirators.

1153. EFFECT OF COOLANT CLEANING DURING REACTOR SHUTDOWN ON DOSE RATE PRODUCED BY PRIMARY CIRCUIT EQUIPMENT (in Czech). J. BURCLOVA, J. MORAVEK, D. BACEK, R. REPAS, L. DOBIS, and J. TISCHLER (Vyskumny Ustav Jadrovych Elektrarni, Trnava (Czechoslovakia); Atomova Elektraren Bohunice, Jaslovske Bohunice (Czechoslovakia)). Jaderna Energie (Czechoslovakia), Vol. 34, Oct 1988, pp. 372-376.

The effect was studied of coolant cleaning on the magnitude of dose rate produced by the primary circuit equipment during several shutdowns of the WWER-440 reactors of the 3rd and the 4th units of the Jaslovske Bohunice nuclear power plant. It was assumed that coolant cleaning during a shutdown could significantly reduce the activity of corrosion products not only in the coolant but also on the surfaces of primary circuit components. The SOV-2 equipment was designed for coolant cleaning featuring its own pumps so that it can operate during the unit shutdown. The results showed that short-term coolant cleaning using the SOV-2 equipment led to significant reduction in coolant activity. However, with the exception of the first reactor duty time, it did not lead to observable reduction in the activity of the inner surfaces of the primary circuit components. Dose rates were thus only reduced for the components for which the coolant volume to the inner equipment surface ratio was sufficiently high. The use of the SOV-2 equipment for removal of corrosion products can only be recommended as efficient in the period of heating up the primary circuit prior to the first or repeat reactor start-up. In the period of reactor cooling and prior to the inspection of fuel cladding for hing, the SOV-2 equipment can only be used for reducing the fission product activity in the coolant.

1154. INTEGRATION OF OPTIMIZATION IN THE REGULATORY PROCESS. R.E. CUNNINGHAM, D.A. COOL, and R.L. O'CONNELL (US Nuclear Regulatory Commission, Washington, DC (USA)). Optimisation of radiation protection, Paris. Nuclear Energy Agency, 75 - Paris (France). Report No. INIS-XN--149, Mar 14 1988, pp. 87-112.

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

1155. CONSIDERATION OF RADIATION PROTECTION OF WORKERS IN THE DESIGN AND OPERATION OF NUCLEAR POWER PLANTS. PT. 2. OPERATION. DRAFT (in German). Berruecksichtigung des Strahlenschutzes der Arbeitskraefte bei Auslegung und Bertrieb von Kernkraftwerken. T. 2. Betrieb. Entwurf. Kerntechnischer Ausschuss (KTA), Koeln (Germany, F.R.). Heymanns (Koeln, Germany, F.R), 19[illegible], 5 pp.

The standard has to be applied for fixing protective measures for the people working in the nuclear power station with regard to exposure to radiation. It concerns the measures which are necessary for activities during operation as authorized and the planning of measures with regard to the incidents and accidents defined according to appendix I StrlSchV. With regard to the protection of the people working in the nuclear station against exposure to radiation it has the objective to ensure that the radiation protection rules are transformed into appropriate organizing and technical protective measures according to paragraph 28 section 1 StrlSchV. This draft modifies the rule KTA-1301.2, version 6/1982.

1156. PROBABILISTIC IDENTIFICATION OF DEFECTS IN BWR PIPINGS (in Japanese). ASAO OKAMOTO, MASATSUNE AKASHI, and MASAKI, KITAGAWA (Ishikawajima-Harima Heavy Industries Co. Ltd., Tokyo (Japan)). Ishikawajima-Harima Giho (Japan), Vol. 28, Sep 1988, pp. 293-298.

In order to evaluate the remaining life of a BWR plant, an estimation method is developed to obtain the probability of a cracking due to fatigue or stress corrosion in pipings, in which easily obtainable plant construction data and the operational history are used as inputs. The period to the occurrence of stress corrosion and fatigue cracking is expressed in a stochastic form according to experimental data, considering field experience. For effective factors of the cracking, available convenient estimation methods are studied and stochastic expression is also given to them. The procedure is coded into a FORTRAN program which calculates the probability by the Monte Calro method. Trial calculation with the data in a cracking incident of an operating plant is shown that this program is an effective tool for predicting failure life.

1157. 130-YEAR SHUTDOWN. THOM, DIBDIN. Environment Now (UK), Nov 1988, pp. 36-37.

The policy on decommissioning nuclear power stations is discussed. This is an urgent problem now that the Magnox stations such as Berkeley, are being closed down having come to the end of their service life. Decommissioning has three stages. Stage 1 is to remove the nuclear fuel and coolant. Stage 2 is to remove the external plant and buildings but leave the reactor and shielding in the smallest volume possible. Stage 3 is to dismantle completely and remove all the radioactive material so the site can be returned to a green field state. A decision has to be made whether to go to stage 3 in 17 years or leave the reactor at stage 2 for 130 years. The problem of dismantling completely now is the disposal of a relatively large amount of high-level radioactive waste. It could be transferred to another site or left in-situ. The second option is much cheaper especially if discounted cash flow is used which only needs a small outlay now to cover the cost of stage 3 in 130 years time. (U.K.).

1158. A REVIEW OF PLANT DECONTAMINATION METHODS: 1988 UPDATE: FINAL REPORT. J.F. REMARK. Electric Power Research Inst., Palo Alto, CA (USA) ; Applied Radiological Control, Inc., Marietta, GA (USA). Report No. EPRI-NP--6169, Jan 1989, 60 pp.

This document updates the state-of-the-art in decontamination technology since the publication of the previous review (EPRI NP- 1128) in May 1981. A brief description of the corrosion-film characteristics is presented as well as corrosion film differences between a BWR and PWR. The generation transportation, activation,and deposition of the radioisotopes found throughout the reactor coolant system is also discussed. Successful,well executed, decontamination campaigns are always preceded by meticulous planning and careful procedure preparation which include contingency operations. The Decontamination Planning and Preparation Section describes the technical planning steps as well as the methodology that should be followed in order to select the optimum decontamination technique for a specific application. A review of a number of the decontamination methods commercialized since 1980 is presented. The basic mechanism for each process is described as well as specific applications of the technology in the fields. Where possible, results obtained in the field are presented. The information was obtained from industry vendors as well as personnel at the plant locations that have utilized the technology.

1159. RADIOLOGICAL CONDITIONS AND EXPERIENCES IN THE TMI-2 AUXILIARY BUILDING. P.E. RUHTER and W.G. ZURLIENE. EG and G Idaho, Inc., Idaho Falls (USA). Report No. EGG-M--88203, Oct 30 1988, 21 pp.

Although the radiological conditions following the TMI-2 accident were extraordinary, those that had a potential impact on personnel were largely confined to the Auxiliary and Fuel Handling Building. The most significant pathway was the Letdown and Make-Up and Purification System. Dose rates in some locations in the Aux/Fuel Handling Buildings were in excess of 3 mSv/s (1000 R/h) during the first few days following the accident. They decreased after three to four days and stabilized after about one week. Airborne radioactivity levels were initially due to the release of noble gases, and subsequently due to resuspension of surface contamination. During the first month, the mixture of fission products in the reactor coolant change from one of largely cesium to where the strontium and cesium were about equal in radiological importance. This created some very high beta radiation levels. The significant strontium levels caused the contamination control limit to be reduced to one-half of the pre-accident limit.

1160. CHINA NUCLEAR SCIENCE AND TECHNOLOGY REPORTS. ABSTRACTS (in Chinese). China Nuclear Information Centre, Beijing, BJ. Report No. CNIC--00202, 1987, 77 pp.

114 abstracts of nuclear science and technology reports, which were published in 1986-1987 in China, are collected. The subjects included are: nuclear physics, nuclear medicine, radiochemistry, isotopes and their applications, reactors and nuclear power plants, radioactive protection, nuclear instruments etc. They are arranged in accordance with the INIS subject categories, and a report number index is annexed.

1161. INTERGRANULAR CORROSION RESISTANCE OF 304 STAINLESS STEEL WELDED JUNCTIONS FOR BWR PIPING (in Italian). V. REGIS and R. PASCALI (Ente Nazionale per l'Energia Elettrica, Milan (Italy). Centro Termica e Nucleare; Centro Informazioni Studi Esperienze, Milan (Italy)) ENEA, Rome (Italy). Proceedings of a Symposium - Welding of nuclear power plant components, Genoa, IT, May 19, 1987, pp. 163-177.

Intergranular corrosion of stainless steel pipings has been noticed in welded areas in almost every boiling water nuclear power plants so as to attract the attention of safety authorities. The research activity developed during more than one decade allowed to apply and qualify a series of provisions aiming at assuring the tensio-corrosion resistence and satisfying the strigent limitations imposed by control authorities. Results achieved by ENEL in cooperation with CISE concerning the study of origin materials, made on mockups of 4" to 8" welded tubings and weldings of BWR plants.

1162. NEA ACTIVITIES IN 1987. Nuclear Energy Agency, 75 - Paris (France). Report No. INIS-XN--146, 1988, 46 pp.

This report presents the main features of the Agency work during 1987. It deals with trends in nuclear power. Nuclear development and the fuel cycle; nuclear safety and licensing; radiation protection; radioactive waste management; legal affairs; nuclear science; joint undertakings and other NEA joint projects; information programmes; organization and administration.

1163. EXPERIENCE WITH GEZIP AT HOPE CREEK GENERATING STATION. J.R. LOVELL, T.W. VANNOY, R.L. WOLD, and R.F. GROUSER (Public Service Electric and Gas Co., Newark, NJ (USA)) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP, Apr 13 1988, v. 2.

Hope Creek Generating Station is a single 1067 MWe BWR plant situated at the mouth of the Delaware River in southern New Jersey. Its commercial operation began in January, 1987. During the late stage of construction, a number of the new methods to reduce worker dose in BWR plants were investigated. Because Hope Creek has a Mark 1 containment, where dry well working conditions are very restrictive, special emphasis was placed on reducing drywell dose rate by minimizing the amount of radioactive materials deposited on piping system. Based on this review, two dose reduction methods were incorporated into the Hope Creek design. The first is the replacement of Stellite pins and rollers of control rod blades with a non-Stellite material. The second method adopted was the installation of a zinc addition system. The decision to provide zinc injection was based on the data developed by General Electric and the Electric Power Research Institute that the plants with higher zinc level in reactor water have on average lower recirculation pipe dose rate. The zinc addition system adopted and the operation with zinc injection are reported. The extreme spike in zinc concentration was experienced in the recent plant shutdown. The GEZIP program is working extremely well.

1164. PROCEDURES FOR REDUCING SHUTDOWN DOSE RATE AT NO.1 AND NO.4 UNITS OF FUKUSHIMA DAIICHI NUCLEAR POWER STATION AND CURRENT EXPERIENCE WITH DOSE RATE AND OCCUPATIONAL EXPOSURE AT THE PLANTS. N. USUI, R. TSURUOKA, K. OHSUMI, S. UCHIDA, M. NAGASE, H. MOCHIZUKI, and Y. HIRAHARA (Hitachi Engineering Co. Ltd., Ibaraki (Japan)) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium-21. JAIF annual conference on water chemistry in nuclear power plants , Tokyo, JP, Apr 13 1988, v. 1.

The No. 1 and No. 4 Units of Fukushima-Daiichi Nuclear Power Station (1F-1 460MWe BWR-3; and F1-4 784MWe BWR-4) started commercial operation in 1970 and 1978, respectively, and are now operating the 12th and 9th fuel cycles, respectively. A Radiation reduction program for 1F-1 and 1F-4 has been implemented since 1979. This report describes the effectiveness of radiation level improvement that has been executing for 1F-1 and 1F-4. The following improvements have been made to reduce the radiation level: (1) increase in frequency of condensate demineralizer backwash, (2) low liner velocity of condensate demineralizer (only in 1F-1), (3) replacement of condensate demineralizer (only for anion resin in 1F-1, cation resin used continuously), (4) installation of hallow fiberfilter (50 % of condensate flow rate in 1F-1, 33 % of condensate flow rate in 1F-4), (5) improvement of lay-up during plant shutdown, and (6) replaced low cobalt materials. As a result, iron concentration in feedwater, radioactive crud concentration in reactor water and hot spots of radiation level have been reduced. But the level of 1F-1 and 1F-4 are still higher than that of new plants. It is necessary to investigate the effective radiation level improvement further.

1165. 18 YEARS OF EXPERIENCE WITH WATER CHEMISTRY AT THE DODEWAARD NUCLEAR POWER PLANT. D.A. KERS (Gemeenschappelijke Kernchergiecentrale Nederland N.V., Dodewaard) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP, Apr 13 1988, v. 1.

The Dodewaard nuclear power plant is of the thermal boiling water reactor type with natural circulation. This type uses water for the moderation of neutrons and for cooling the core. In the primary circuit various types of stainless steel is used as construction material. A special operation for the condensate polishing system (CPS) was created in 1982. Once a month the stand-by filter of the CPS was put into service and the other one switched off and thoroughly backwashed. This created a large decrease in the suspended iron and cobalt amounts which were transported toward the reactor. The decrease of the iron and cobalt input had a direct influence on the crude layers in the auxiliary system. Also, the amount of crud on the control rod drives has been lowered. Oxygen can cause inter granual attack followed by inter granual stress corrosion cracking (IGSCC). The conductivity of the reactor-water is very low, and this allows the e.c.p. value to be as low as -25 mV (SHE) before IGSCC will occur. Thus HWC is not implemented. To separate solids resulting from errosion, etc., a plate-filter with precoat powder has been installed. A sedimentation and flocculation tank is also used. A great number of modifications permitted water transport to and from the various (waste) water systems,but this caused serious cross-contamination. Efforts are under way to solve this problem.

1166. OPERATING EXPERIENCE ON RADIATION REDUCTION IN THE LATEST BWRS. K. OHSUMI, S. UCHIDA, M. AIZAWA, K. TAKAGI, O. AMANO, and K. YAMASHITA (Hitachi Ltd., Ibaraki (Japan). Hitachi Works) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP, Apr 13 1988, v. 1.

In Japan, BWR plants have been operated commercially since 1970, and the reduction of radiation exposure has been an important concern. The application of the procedure for reducing occupational exposure is incorporated in Japanese Improvement and Standardization Program for LWRs. No.2 and No.4 plants in Fukushima No.2 Nuclear Power Station were designed and constructed as the latest 1,100 MWe BWRs in conformity with the Improvement and Standardization Program. No.2 plant began the commercial operation in February, 1984, and experienced three times of the scheduled annual maintenance outage. No.4 plant began the commercial operation in September, 1987, and the first annual maintenance is scheduled from September, 1988. In this paper, discussion is focused on recent radiation reduction measures, that is the control of iron and nickel in primary coolant for reducing the radiation dose rate in primary systems, based on the experience with No.2 and No.4 plants. The design concept of a low radiation dose rate nuclear power plant, the experience on water chemistry in No.2 plant, the control of iron and nickel in No.4 plant operation and so on are reported. It is believed that these operation experiences contribute to the reduction of occupational exposure in BWR plants currently in operation and in future.

1167. FIRST UNIT OF SHIMANE N.P.S.-WATER CHEMISTRY IMPROVEMENT BASED ON LONG TERM OPERATING EXPERIENCE. YASUNORI MATUSIMA, KHOICHI YONEZAWA, FUMIO MIZUNIWA, HISAO ITOW, KATSUMI OHSU-MI, and MOTOHIRO AIZAWA (Chugoku Electric Power Co. Inc., Hiroshima (Japan)) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP, Apr 13 1988, v. 1.

The first unit of Shimane Nuclear Power Station, Shimane-1, is the first Japan's domestically fabricated 460 MWe BWR. Shimane-1 is owned by Chugoku Electric Power Co., Ltd. Its commercial operation started in March 1974. It was constructed by Hitachi Ltd. as the prime contractor, and is owned by Chugoku Electric Power Co., Ltd. Shimane-1 has since been operating stably. The present fuel cycle is the 13 th. This paper describes the relationship between the feed water iron crud concentration and radioactivity concentration in the reactor water on the basis of water chemistry experience,and also addresses the development of methods for managing iron exchange resin for the condensate demineralizer. Operation of Shimane-1 by Chugoku Electric Power Co., Ltd. gave a dynamic moment toward establishing a concept for low radiation build-up. The point of importance was to maintain the highest attainable water quality. Feed water iron concentration is kept at less than 1 ppb. Data on the break through capacity of the condensate demineralizer ion exchange resin are used to evaluate the backwash treatment and chemical regeneration treatment. Radwaste from the condensate demineralizer is expected to be reduced to a half of the present volume.

1168. VARIATIONS IN RADIATION DOSES TO PERSONNEL OF NPP WITH WWER DEPENDING ON OPERATION TIME (in Russian). V.P. ROMANOV. Radiation safety and radiation protection of NPP. Issue 12. Collection of papers , 1987, pp. 6-9.

The problem concerning changing of dose costs (DC) to personnel of NPP with WWER-440 depending on operation time is considered. It is shown, that many factors, the basic of which are the precipitation activity for the equipment, washed by radioactive liquid media and the repair scope, effect on the formation of personnel dose costs during NPP operation. Published results on DC investigation are analyzed. It is recommended to use NPP operation effective time instead of calendar one to investigate character of DC dependence on time.

1169. RADIATION DOSES TO PERSONNEL OF NPP WITH LWGR (in Russian). YU.A. EGOROV, A.A. NOSKOV, and A.F. SHAMA-SHOV. Radiation safety and radiaiton protection of NPP. Issue 12. Collection of papers, 1987, pp. 3-6.

Arithmetical means of annual individual doses and dose costs for personnel of NPP with different type reactors are considered. Comparison of these characteristics has revealed, that individual dose to the personnel of NPP with SWQR practically does not differ from its value at NPPs with WWER reactors or at all the US NPPs with LWR, while personnel dose costs at different NPPs differ essentially. Dose costs per unit of the output energy (specific dose costs) are rather low and per energy output at all the NPPs with the given type reactor are of most minimal value.

1170. CONTROL OF NICKEL RELEASE FROM INCONEL X-750 BY SURFACE TREATMENT. K. TADA, T. FUJIWARA, M. YAMAMOTO, T. BABA, and H. NAGAO (Toshiba Corp., Kawasaki, Kanagawa (Japan)) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP, Apr 13 1988, v. 2.

Occupational radiation exposure becomes very low in recent BWRs as the result of the system improvement and material development to reduce the radiation level. One of the current major sources of activated corrosion products responsible for radiation level buildup is considered to be Inconel X-750 used in a neutron field. Cobalt-58 and cobalt-60 are generated directly from nickel-58 and cobalt-59, respectively, in the alloy, and dissolve into reactor water. Therefore, the inhibition of such release by surface treatment is the most effective means to accomplish further reduction of radiation level. In this study, the authors selected oxidation in the air as the surface treatment, and examined its effect on the metal release into high temperature pure water. Moreover, the authors characterized the oxide layer formed on the surface, and examined the relationship between metal release and the oxide layer. The Inconel X-750 tested, the immersion test, the oxide layer analysis, the level of nickel release and so on are reported. The level of nickel release from Inconel X-750 decreased by preoxidation in the air. An oxide layer covered the base metal, and inhibited metal release. The inhibition effect was larger as oxidation was carried out at higher temperature for longer time.

1171. COBALT DEPOSITION STUDIES IN GE VALLECITOS TEST LOOPS, (2). C.C. LIN, F.R. SMITH, Y. URUMA, T. TANEICHI, and N. ICHIKAWA (General Electric Co., Pleasanton, CA (USA). Vallecitos Nuclear Center) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP, Apr 13 1988, v. 2.

The buildup rate of a radiation field during shutdown in nuclear power reactors is an important factor to maintain occupational exposure as low as reasonably achievable. The major radiation source in BWRs that causes personnel exposure in shutdown maintenance has been identified to be activated corrosion products, mainly Co-60, deposited on primary system piping walls. In the present cobalt deposition studies, a test program has been designed specifically to provide a better understanding of Co-60 deposition process. The main objectives of this program are shown. Some early test results including the effects of water conductivity, pH, dissolved oxygen and surface pretreatment have been already published. The effect of Zinc ions in water was also reported. In this paper, the latest test results including the effect of metallic ions on the Co-60 deposition of stainless steel surfaces are summarized. The experiment using a test loop and the results on the effect of metallic ions and Co-60 deposition mechanism are reported. Direct reaction/crystallization, adsorption/recrystallization, and ion exchange have been hypothesized as the major mechanism.

1172. EXPOSURE DURING ROUTINE MAINTENANCE OF THE REACTOR IN NUCLEAR POWER PLANT (in Serbo-Croat). D. KUBELKA, DJ. HORVAT, and A. LAKOVSKI (Institut za medicinska istrazhivanja i medicinu rada, Zagreb (Yugoslavia); Filozofski fakultet Skopje, Institut za ONO (Yugoslavia)) Yugoslav Radiological Protection Association, Belgrade. Proceedings of a Symposium - 14. Yugoslav symposium on radiation protection, Novi Sad, YU, Jun 8 1987, pp. 331-334.

The purpose of this issue was to analyse structural chromosome damages in lymphocytes of 7 persons working on a specific job of periodical service in nuclear power plant. The results of pre-expositional chromosomal aberration analyses, physical dosimetric approximation and chromosomal aberration analyses after the exposure, were mutually compared. There is a certain difference between biodosimetric and physical estimation.

1173. CORROSION OF INCONEL X750 IN SIMULATED BWR CORE ENVIRONMENT. Y. MORIKAWA, T. BABA, H. NAGAO, K. SHIMIZU, S. YOSHIKAWA, N. ICHIKAWA, N. SAITO, and Y. HEMMI (Toshiba Corp., Kawasaki, Kana-gawa (Japan)) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP, Apr 13 1988, v. 2.

The corrosion tests of Inconel X750 in the simulated core environment by using an irradiated test loop were performed. From the experiments, the followings were clarified; 1. corrosion rate was proportional to minus cubic root of the time and the value was evaluated to be 19 + -4 mdm after one year, 2. the value was markedly high compared to other reported experimental data, but it had a good coincidence with the value evaluated from activity balance data in actual BWR plants, 3. the most important chemical specie which affected the corrosion behavior was identified to be hydrogen peroxide, and 4. the rate of specimens pre-filmed in the high temperature atmosphere at 700 °C for 5 hrs was reduced to be one third or less than that of specimens pre-oxidized in steam at 390 °C for 13 hrs which was the same condition for actual BWR fuel springs.

1174. DEMONSTRATION OF ALKALINE PREFILMING PROCESS FOR PRIMARY PIPING IN A NEW JAPANESE BWR. T. HONDA, K. OHSUMI, K. TAKAGI, O. AMANO, K. YAMASHITA, and M. AIZAWA (Hitachi Ltd., Ibaraki (Japan). Hitachi Works) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP, Apr 13 1988, v. 1.

No abstract available.

1175. PERIODICAL INSPECTION AND ITS AUTOMATION IN NUCLEAR INDUSTRY (in Japanese). YOSHIYUKI SHINOHARA (Japan Atomic Energy Research Inst., Tokai, Ibaraki. Tokai Research Establishment). Keisoku To Seigyo (Japan), Vol. 27, Jun 1988, pp. 489-492.

The major element of regular inspection of nuclear facilities is the inspection of equipment in a radioactive environment. It is important, therefore, to reduce or prevent radiation exposure sufferred by inspection person. Taking nuclear power plants as an example, the present article describes major techniques used in inspection of nuclear facilities in Japan. Regular inspection of nuclear power plants in Japan is generally performed on the in-service basis and according to a plan that specifies the scope, procedure and frequency of regular inspection. Such inspection covers the disassembling and detailed checking of equipment to ensure the soundness of the facilities. In addition to visual examination, TV cameras and fiberscopes are used to check for flaws, wear, cracks or corrosion. Magnetic powder inspection, liquid penetrant examination, etc., are performed to check for flaws on surfaces. Radiations, ultrasonic waves and eddy currents are used to check for defects within bulk parts of equipment. Th recent use of automatic inspection apparatus and remote-controlled devices has reduced the radiation exposure dose to about 1/2 and the plant shutdown period to 2/3. Some automatic inspection apparatus are outlined.

1176. ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. W.J. SHACK, T.F. KASSNER, P.S. MAIYA, J.Y. PARK, and W.E. RUTHER. Argonne National Lab., IL (USA). Report No. CONF-8810155--25, Oct 24 1988, 29 pp.

Research during the past year focused on (1) stress corrosion cracking (SCC) of austentitic stainless steels (SS), (2) fatigue of Type 316NG SS, and (3) SCC of ferritic steels used in reactor piping, pressure vessels,and steam generators. Stress corrosion cracking studies on austentitic SS explored the critical strains required for crack initiation, the effects of crevice conditions on SCC susceptibility, heat-to-heat variations in SCC susceptibility of Type 316NG and modified Type 347 SS, the effect of heat treatment on the susceptibility of Type 347 SS, threshold stress intensity values for crack growth in Type 316NG SS, and the effects of cuprous ion and several organic salts on the SCC of sensitized Type 304 SS. Crevice conditions were observed to strongly promote SCC. Significant heat-to-heat variations were observed in SCC susceptibility of Types 316NG and 347 SS. No correlation was found between SCC behavior and minor variations in chemical composition. A significant effect of heat treatment was observed in Type 347 SS. A heat that was extremely resistant to SCC after heat treatment at 650 °C for 24 h was susceptible to transgranular stress corrosion cracking (TGSCC) in the solution-annealed condition. Although there was no sensitization in either condition, the presence or absence of precipitates and differences in precipitate morphology appear to influence the SCC behavior.

1177. CALCULATIONAL MODEL OF A RADIATION SOURCE FOR FORECASTING GAMMA RADIATION FIELDS NEAR THE WWER-440 REACTOR PRIMARY COOLANT CIRCUIT COMPONENTS (in Russian). M.M. AMOSOV and L.P. KHAM'YANOV. Radiation safety and radiation protection of NPP. Issue 12. Collection of papers, 1987, pp. 23-25.

The problem concerning the calculation of radiation situation near the WWER-440 reactor primary coolant circuit components and personnel additional irradiation due to fission product release from defective fuel elements into the coolant is considered. The formula for the calculation of fission product specific activities in the primary coolant circuit water is obtained on the base of relation describing fission product accumulation within a fuel element. Calculational results for iodine isotope limiting specific activities in the coolant of reactor, operating at nominal power level are presented. It is possible to determine γ-radiation sources connected with fission product release into the coolant having these data and the values of equipment volumes filled with coolant.

1178. ACTIVITY BUILD-UP ON THE CIRCULATION LOOPS OF BOILING WATER REACTORS: BASICS FOR MODELLING OF TRANSPORT AND DEPOSITION PROCESSES (in German). B. COVELLI and H.P. ALDER (Tecova A.G., Wohlen (Switzerland); Paul Scherrer Institut, Labor fuer Werkstoffe und nukleare Verfahren, Wuerenlingen (Switzerland)). Paul Scherrer Inst. (PSI), Wuerenlingen (Switzerland). Lab. fuer Werkstoffe und Nukleare Verfahren. Report No. PSI--2, Mar 1988, 48 pp.

In the past 20 years the radiation field of nuclear power plant loops outside the core zone was the object of investigations in many countries. In this context test loops were built and basic research done. At our Institute PSI the installation of a LWR-contamination loop is planned for this year. This experimental loop has the purpose to investigate the complex phenomena of activity deposition from the primary fluid of reactor plants and to formulate analytical models. From the literature the following conclusions can be drawn: The principal correlations of the activity build-up outside the core are known. The plant specific single phenomena as corrosion, crud-transport, activation and deposit of cobalt in the oxide layer are complex and only partially understood. The operational experience of particular plants with low contaminated loops (BWR-recirculation loops) show that in principle the problem is manageable. The reduction of the activity build-up in older plants necessitates a combination of measures to modify the crud balance in the primary circuit. In parallel to the experimental work several simulation models in the form of computer programs were developed. These models have the common feature that they are based on mass balances, in which the exchange of materials andhe sedimentation processes are described by global empirical transport coefficients. These models yield satisfactory results and allow parameter studies; the application however is restricted to the particular installation. All programs lack models that describe the thermodynamic and hydrodynamic mechanisms on the surface of deposition layers. Analytical investigations on fouling of process equipment led to models that are also applicable to the activity build-up in reactor loops. Therefore it seems appropriate to combine the nuclear simulation models with the fundamental equations for deposition.

1179. COST ESTIMATES: STRESS IMPROVEMENT REMEDIES FOR SAVANNAH RIVER REACTOR PROCESS WATER PIPE WELDS. G.R. CASKEY, P.R. VORMELKER, and W.S. EHRHART. Du Pont de Nemours (E.I.) and Co., Aiken, SC (USA). Savannah River Lab.. Report No. DPST--88-951, Nov 15 1988, 5 pp.

The process water pipes in the Savannah River reactors were made of Type 304 stainless steel. This steel becomes susceptible to intergranular stress corrosion cracking (IGSCC) when sensitized as by welding. IGSCC may initiate in locations under residual or applied tensile stress in the presence of impure water such as the moderator. One recommendation from the 1986 ES & H audit was for further evaluation of the application of stress improvement (SI), either induction heating stress improvement (HSI) or heat sink welding (HSW), to preclude further cracks and leaks in susceptible welds in the process water piping. In our response to this ES & H recommendations, we stated that there was no safety reason to consider such action, but that we would evaluate the operational cost-benefit of remedial treatment of the welds. Mechanical stress improvement (MSIP), weld overlay (WO) and Pipelock were evaluated also. All of the welds in the process water systems which join Type 304 stainless steel are potential sites for IGSCC. Therefore, any of the approved treatments would be of potential benefit by eliminating or reducing the tensile stress necessary for IGSCC.

1180. GUIDELINE ON QUALIFICATION REQUIREMENTS AND ON FURTHER TRAINING OF RESEARCH REACTOR PERSONNEL TO ENSURE ATOMIC SAFETY AND RADIATION PROTECTION (in German). Staatliches Amt fuer Atomsicherheit und Strahlenschutz, Berlin (German Democratic Republic). Report No. SAAS-Mitt--88-06, 1988, 10 pp.

The guideline which entered into force on 3 June 1988 governs the qualification and further training requirements to be met by persons who are responsible for ensuring atomic safety and radiation protection during commissioning, operation, maintenance, and decommissioning of research and training reactors as well as subcritical assemblies.

1181. GUIDELINE ON RADIATION PROTECTION REQUIREMENTS FOR IONIZING RADIATION SHIELDING IN NUCLEAR POWER PLANTS (in German). Staatliches Amt fuer Atomsicherheit und Strahlen-schutz, Berlin (German Democratic Republic). Report No. SAAS-Mitt--88-04, 1988, 10 pp.

The guideline which entered into force on 1 May 1988 stipulates the radiation protection requirements for shielding against ionizing radiation to be met in the design, construction, commissioning, operation, and decommissioning of nuclear power plants.

1182. SUMMARY ON PRE-FEASIBILITY STUDY OF HIGH TEMPERATURE GAS-COOLED REACTOR APPLICATION IN CHINA (in Chinese). XU JIMING. Hedongli Gongcheng (China), Vol. 9, Feb 1988, pp. 19-23.

The position and role of HTGR in China's nuclear energy development are outlined. The inquiry is tentatively made for potential market of HTGR in China. As an example, the adaption design and cost estimation of a small HTGR nuclear power station (HTR100) have been included. The results show that from the point of view of technical aspect, it is feasible to build the small HTGR nuclear power station in China, and the generating cost is comparable with coal-fired power station for the same capacity in some region..

1183. UNUSUAL WORK IN LOVIISA'S FUEL POND (in Finnish). B. WAHLSTROEM (Imatran Voima Oy, Helsinki (Finland)). ATS Ydintekniikka (F.R. Germany), Vol. 17, 1988, pp. 9-11.

At the first unit of Loviisa NPS and unusual job was performed during the spring 1988. All fuel-elements and the water was completely evacuated from the fuel pond. Thereafter inspection and cleaning work, going on for several weeks in the contaminated pool, was started. The work was successful and only minor radiation doses were received.

1184. COMPLETE DECONTAMINATION OF THE LINGEN PRIMARY STEAM DUCT (in German). W. AHLFAENGER (Kernkraftwerk Lingen G.m.b.H., Darme/Lingen (DE)). Commission of the European Communities, Luxembourg. Report No. EUR--11435, 1988, 44 pp.

In a previous work performed within the framework of the research programme of the European Community concerning the decommissioning of nuclear power plants, a procedure for the complete decontamination of austenitic materials has been developed in laboratory tests. In the present study, the application on a technical scale is described. A pipe piece of the primary steam duct of the Lingen Nuclear Power Station with a nominal diameter of 50 mm and a length of about 20 m was chosen as test object. As shown by the examination results, the pipe piece could be decontamined below the tolerance limits of the German Radiation Protection Regulation.

1185. OCCUPATIONAL RADIATION DOSE STATISTICS FROM LIGHT-WATER POWER REACTORS OPERATING IN WESTER EUROPE. I.R. BROOKES and T. ENG (Central Electricity Generating Board, Gloucester (UK). Oldbury Nuclear Power Station; Swedish State Power Board,Vaellingby (SE)). Commission of the European Communities, Luxembourg. Report No. EUR--10971, 1987, 228 pp.

Since the early days of nuclear power, collective and individual doses for people engaged in the maintenance and operation of nuclear power plants have been published by regulatory authorities. In 1979 a small working party whose members were drawn from Member States operating light-water reactors (LWRs) in the European Community was convened. The Working party decided that only by collection of data under a unified scheme would it ever be possible to properly compare plant performance and for this reason a questionnaire was drawn up which attempted to elicit the maximum of information with the minimum inconvenience to the plant staff. Another decision made by the working party was to broaden the data base from 'European Community LWRs' to 'West European LWRr' to try to take advantage of the considerable experience being built up in Sweden, in Finland and in Switzerland. All the data available to the Commission up to the end of 1984 are presented and commented on. The deductions are not exhaustive but are believed to represent the limits of what could sensibly be done with the data available. Results are presented separately for BWR and PWR but no other subdivision, say by country or maker, is made. Where interpretation can be enhanced by graphical presentation, this is done. In general, doses for each job category are expressed in various ways to reveal and afford comparisons.

1186. FUTURE DIRECTIONS IN RADIATION PROTECTION IN NUCLEAR POWER PLANTS. L. LEWIS (Duke Power Co. (US)). Proceedings of a Symposium - Topical conference on theory and practice in radiation protection and shielding, Knoxville, TN, US, Apr 22 1987, pp. 291-294.

Our visions of the future are often very optimistic and hopeful, representing the best imaginings of the human mind. The authors are inclined to think of the future as filled with new and better things. Some people even visualize the future as a science fiction perfection, but, in reality, it will also contain elements of the past and the present, both good and bad. With respect to radiation protection, a guess would tell us that the future holds the implementation of some version of ICRP-26 in one revision or another of the NRC 10 CFR20 regulations. But many of the technical problems of today may likely be 'solved' by the public, the politicians, the sociologists and the bureaucrats of the future. For example, two such 'solutions' may possibly appear in such grotesque forms as drastically lowered allowable annual doses or as engineered facilities for the disposal of low-level radioactive waste above ground on seismic stilts. All of these aspects - the good, the bad,the new, the old, and the indifferent - are all touched upon in this vision of the future.

1187. NEUTRON EXPOSURE ANALYSIS IN REACTORS. FINAL REPORT (in German). W. JACOBI, A. WITTMANN, J. KOLLERBAUR, A. MORHART, H. SCHRAUBE, and G. BURGER. Bundesministerium fuer Umwelt, Naturschutz und Reaktorsicherheit, Bonn (Germany, F.R.) ; Gesellschaft fuer St. Report No. BMU--1988-179, 1988,139 pp.

The purpose of this study was to make a radiation risk analysis for occupational exposure to neutrons of reactor personnel. The neutron exposure was analysed at selected, typical places of work by comparing theoretical and experimental dosemeter readings with particular regard to albedo dosemeters, and by theoretical calculation of body doses (and risks) on the basis of theoretical spectra and spectra determined indirectly by means of the multi-sphere method.

1188. RESULTS OF OPERATION OF BWRS IN JAPAN (in Japanese). KENSUKE FUEKI (Tokyo Electric Power Co., Inc. (Japan)) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - Japan-U.S.S.R seminar on safety of LWR nuclear power plants, Tokyo, JP, Oct 27 1987, pp. 5/1-5/22.

It is considered that the development of BWR plants in Japan has been advanced relatively smoothly though sometimes there were complications. As of the end of fiscal year 1986, the BWR plants in operation were 16, and the total power output amounted to 12,917 MW, which was equivalent to 8 % of 153 GW of the total power source facilities in Japan. Reflecting the excellent operational result of BWR plants, the generated electric power in fiscal year 1986 reached 85.9 TWh and about 15 % of the total, exceeding hydroelectric power. This means that about 18 million tons of petroleum import was reduced. At the initial stage, BWR plants suffered the stress corrosion cracking of stainless steel pipings in the reactor primary system. This trouble was successfully solved by the efforts of the government, electric power companies, plant manufacturers and research institutes. In fiscal year 1983, the capacity factor of all BWR plants in Japan recovered to more than 70 %, and in fiscal year 1986, it has reached 75.9 %. In order to improve the capacity factor further, it is necessary to prevent troubles by the development of diagnostic techniques and preventive maintenance, and to shorten regular inspection period. The change of accidents and troubles, the measures to reduce regular iperiod, the reduction of radiation exposure, and the reduction of wastes are reported.

1189. ANNOUNCEMENT OF DRAFT VERSIONS OF SAFETY ENGINEERING RULES OF THE KTA. AS OF SEPTEMBER 21, 1988 (in German). B undesanzeiger (F.R. Germany), Vol. 40, Oct 4 1988, pp. 4400-4401.

Publication of contents of KTA 1301.2 draft (proposal for amendment), concerning personnel radiation protection aspects in design and operation of nuclear power plant. Part 2, Operation. Publication of contents of draft rule KTA 1404, methodology of documentation during construction and operation of nuclear power plant. Publication of contens of draft rule KTA 1502.2, radioactivity monitoring of room air in nuclear power plant, Part 2: Nuclear power stations with HTR. Publication of contents of draft rule KTA 3205.3, component support systems with non-integral connections, Part 3: Series produced standard supports. Publication of contents of draft rule KTA 3605, treatment of contaminated gases in nuclear power plant with LWR. The attached documentation ... for establishment of the rule presents the composition of the responsible working group of the competent working group of the competent KTA sub-committee, as well as the time schedule. Comments on the full text of the rule explain the conceptual approach and technical facts and data.

1190. COBALT DEPOSITION CONTROL BY PRETREATMENT OF STAINLESS STEEL SURFACE. K. YAMAZAKI and Y. MORIKAWA (Nippon Atomic Industry Group Co. Ltd., Kawasaki, Kanagawa. Nuclear Research Lab.) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP, Apr 13 1988, v. 2.

Deposition tests of radioactive cobalt on stainless steel surface under a boiling water reactor condition were performed for study of control of radiation buildup on the surface by prefilming. Oxide filmes formed by prefilming were examined to evaluate relationship between characteristics of oxides and beneficial factor. It is found that the condition more oxidizing than normal primary coolant is beneficial and one of most important factors to make protective oxide film is initial high reaction rate of oxide crystalization.

1191. PRECONDITIONING AND PASSIVATION OF REACTOR MATERIALS TO REDUCE RADIATION FIELD BUILDUP. H. OCKEN and C.J. WOOD (Electric Power Research Inst., Palo Alto, CA (USA)) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP, Apr 13 1988, v. 2.

Future reductions in radiation fields will make use of preconditioning and passivation techniques to reduce the incorporation of ^{60}Co into the oxides that form on reactor construction materials. This paper describes ^{60}Co activity measurements following exposure of prefilmed stainless steel specimens in both simulated BWR and PWR coolant chemistry. An electroless layer of palladium performed best in both environments. Loop test results of a preoperational cleaning technique that holds promise for replacement PWR steam generators also are discussed.

1192. FULL SYSTEM DECONTAMINATION FEASIBILITY STUDIES. R.P. DENAULT, J.E. LESURF, and F.W. WALSCHOT (London Nuclear Ltd., Niagara Falls, Ontario (Canda)) Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP, Apr 13 1988, v. 2.

Many chemical decontaminations have been performed on subsystems in light water reactors (BWRs and PWRs) but none on the full system (including the fuel) of large, (500 MWe) investor owned reactors. Full system decontaminations on pressure-tubed reactors have been shown to facilitate maintenance, inspection, repair and replacement of reactor components. Further advantages are increased reactor availability and plant life extension. A conceptual study has been performed for EPRI (for PWRs) and Commonwealth Edison Co (for BWRs) into the applicability and cost benefit of full system decontaminations (FSD). The joint study showed that FSDs in both PWRs and BWRs, with or without the fuel included in the decontamination, are feasible and cost beneficial provided a large amount of work is to be done following the decontamination. The large amounts of radioactive waste generated can be managed using current technologies. Considerable improvements in waste handling, and consequent cost savings, can be obtained if new techniques which are now reaching commercial application are used.

1193. CONSIDERATIONS CONCERNING THE REDUCTION OF PRIMARY CIRCUIT CONTAMINATION IN PRESSURIZED WATER REACTORS (in German). W. AHLFAENGER and E. SCHUSTER (Kernkraftwerke Lippe-Ems G.m.b.H., Lingen (Germany, F.R.). Kernkraftwerk Emsland; Siemens A.G., Erlangen (Germany, F.R.). Unternehmensbereich KWU). VGB Kraftwerkstechnik (F.R. Germany), Vol. 68, Oct 1988, pp. 1043-1047.

As operating experience from PWR plants shows the activiation of Co-59 to Co-60 is a significant source of primary circuit contamination. Following intensive discussions with the constructor of the plant it was basically decided to minimize the cobalt quantities in the materials in the primary circuit. An important step in this direction was the replacement of cobalt alloys by other materials. The deciding parameters of the chemical treatment were considerably more difficult to determine. The paper explains that a substantial reduction in contamination can also be achieved through primary water chemistry. In this respect the paper discusses in detail the transport mechanism of the corrosion products, the influence of plant design concepts and the build-up of oxide films.

1194. IMPROVEMENT IN RADIATION EXPOSURE REDUCTION TECHNOLOGY AT BWR PLANTS. PRESENT STATUS AND PROSPECT (in Japanese). KATSUMI OSUMI, NOBORU JIBU, RYOZO TSURUOKA, and MOTOHIRO AIZAWA (Hita-chi Ltd., Ibaraki (Japan). Hitachi Works). Hitachi-Hyoron. A Monthly Magazine for Electrical and Mechanic, Vol. 70, Apr 1988, pp. 417-422.

In nuclear power plants, accompanying the increase of the number of plants in operation, in order to carry out the maintenance and checkup works smoothly, it becomes important to reduce radiation dose. In Hitachi Ltd., in order to lower radiation dose, the design technology for low dose plants was established, and has been applied to new plants. In this paper, the state of application of radioactivity reduction technology, such as Fe/Ni ratio control, prefilming and so on, which were developed on the basis of the operational experience of No.2 plant in Fukushima No.2 Nuclear Power Station, Tokyo Electric Power Co., Inc., to No.4 plant in the same power station and its effect are reported. The radiation dose to which workers are subjected is the product of the dose rate at working places and working hours. Therefore, the reduction of dose rate in working environment by reducing the radioactivity in plants is the most effective method for improving working efficiency. The reduction of radioactivity concentration in reactor water, the restriction of the accumulation of radioactivity in equipment and piping, and the countermeasures for improving radiation dose control are described.

1195. METHOD AND DEVICE FOR SUPPRESSING DEPOSITION OF RADIOACTIVE MATERIAL (in Japanese). MOTOHIRO AIZAWA, KATSUMI OSUMI, HISAO ITO, and TAKASHI. HONDA (Hitachi Engineering Co. Ltd., Ibaraki (Japan)), May 9 1988, 10 pp.

Purpose: To suppress the deposition of radioactive materials to primary coolant circuit components of a nuclear reactor. Method: During nuclear heating operation of a BWR type nuclear power plant, alkali chemicals are injected in a slight amount to the reactor water. Then, pH value in the water for pre-treatment is adjusted to a weakly alkaline range and the water is supplied circulately the reactor primary coolant circuits. This forms a great amount of dense oxide films to the surface of primary coolant circuit components of the nuclear reactor. The films prevent the dissolved oxygen in the reactor water from diffusing to the surface of the components and also prevents corrosion products leached from the metal matrix upon corrosion from diffusing into the reactor water. Accordingly, growing of new films formed under the reactor water circumstance after the pretreatment can effectively be suppressed to reduce the radioactivity of plants.

1196. DEPOSITION BEHAVIOR OF METAL ELEMENTS ON FUEL CLADDING. YUTAKA URUMA, KENJI YAMAZAKI, TAKAO BABA, YOSHITAKE MORIKAWA, HIROYUKI NAGAO, TADAO KOAKUTSU, and YUTAKA SUZUKI (Nippon Atomic Industry Group Co. Ltd., Kawasaki, Kanagawa (Japan). Nuclear Research Lab.). NAIG (Nippon Atomic Industry Group) Annual Review (Japan), May 1988, pp. 64-66.

No abstract available.

1197. STRESS CORROSION CRACKING OF NUCLEAR PRESSURE VESSEL STEELS IN PRESSURIZED HIGH TEMPERATURE WATERS (in Japanese). TETSUO SHOJI, HIDEAKI TAKAHASHI, and SYUJI AIZAWA (Tohoku Univ., Sendai (Japan). Faculty of Engineering). Nippon Kikai Gakkai Ronbunshu, A Hen (Japan), Vol. 54, Jun 1988, pp. 1251-1257.

The quantitative evaluation of the subcritical crack growth behavior of nuclear pressure vessels and piping steels in pressurized high temperature water has been receiving extensive attention to ensure the structual integrity of nuclear pressure vessels and piping. Recent research progress in this field suggests the significance of the sulfur content of steels in assessing susceptibility to environmentally assisted cracking of these steels in high temperature water from the view point of solution chemistry at a crack tip, where dissolution of MnS, non-metallic inclusion, results in an enrichment of S anion, such as SO_4^{2-}. Hence, it is neccesary to simulate the solution chemistry at a crack tip for the slow strain rate test (SSRT) by use of a smooth specimen which has been commonly used as a useful tool to see stress corrosion cracking characteristics in a short time. In this study, SSRT tests are performed in simulated BWR or PWR environments, and specimen electrochemical potentials are controlled chemically or potentiostatically. The cracking potential of this material/environments systems are examined quantitatively as a function of temperature, sulfur content in steels and also SO_4^{2-} contents in solution, simulating crack tip solute potential showed a lower value of about 250 °C, rather than 288 °C. Sulfur content of steel and SO_4^{2-} concentration in solution can drastically influence the cracking potential and some amount of sulfur in steels or SO_4^{2-} anion at a crack tip make it possible to crack at a BWR or PWR condition.

1198. FUGEN OPERATING EXPERIENCE, 10. DEVELOPMENT OF OPERATION AND MAINTENANCE TECHNOLOGY OF FUGEN (in Japanese). Donen Giho , Mar 1989, pp. 65-73.

In order to advance operation and maintenance technology of Fugen, research and development works are being carried out: (a) research and development of H_2 H_2O isotope exchange reaction method using a hydrophobic catalyst and the construction of heavy water upgrade facilities applying this method. (b) confirmation of SCC suppression technique by hydrogen injection, which practically used at Fugen, is the first case in Japan. (c) development and performance of remote-controlled inspection equipments for pressure tubes of inlet/outlet tubes. (d) development of chemical decontamination techniques and application to decontamination of components and of a primary coolant loop with the view of saving exposure dose. (e) application of Fuzzy logic to a reactor feed water controller. (f) development of a new type in-core monitor (Long Lived Local Power Monitor).

1199. FUGEN OPERATING EXPERIENCE, 7. PRIMARY COOLANT CHEMISTRY CONTROL OF FUGEN (in Japanese). Donen Giho (Japan), Mar 1989, pp. 54-57.

Primary coolant chemistry control of Fugen is conducted to suppress radionuclide build up on the inner surface of primary circuit pipeworks and to reduce susceptibility of materials against stress corrosion cracking. Owing to the oxygen injection, improvement of back wash method for condensate demineralizers, hot-drain-off technique, and clean up operation prior to plant power up, the amounts of iron and cobalt carried to the reactor core are markedly reduced and radiation levels are kept low. Hydrogen injection method is applied to protect stainless steel from stress corrosion cracking.

1200. WATER CHEMISTRY EXPERIENCE OF NUCLEAR POWER PLANTS IN JAPAN (in English). KENKICHI ISHIGURE, KENJI BE, NOBUO AKAJIMA, HIROYUKI AGAO, and SHUNSUKE CHIDA. Journal of Nuclear Science and Technology (Japan), Vol. 26, pp. 145-156.

Japanese LWRs have experienced several troubles caused by corrosions of structural materials in the past ca. 20 years of their operational history, among which are increase in the occupational radiation exposures, intergranular stress corrosion cracking (IGSCC) of stainless steel piping in BWR, and steam generator corrosion problems in PWR. These problems arised partly from the improper operation of water chemistry control of reactor coolant systems. Consequently, it has been realized that water chemistry control is one of the most important factors to attain high availability and reliability of LWR, and extensive researches and developments have been conducted in Japan to achieve the optimum water chemistry control, which include the basic laboratory experiments, analyses of plant operational data, loop tests in operating plants and computer code developments. As a result of the continuing efforts, the Japanese LWR plants have currently attained a very high performance in their operation with high availability and low occupational radiation exposures. A brief review is given here on the R and D of water chemistry in Japan.

1201. OVERVIEW ON DEVELOPMENT OF HIGH RELIABILITY NUCLEAR POWER PLANT IN JAPAN WITH PARTICULAR EMPHASIS ON FUEL AND MATERIALS (in English). YOSHITSUGU MISHIMA. Journal of Nuclear Science and Technology (Japan), Vol. 26, Jan 1989, pp. 115-117.

To begin with the concept of nuclear safety in Japan, efforts to achieve power reactor operation with higher reliability and safety have been described. More strict regulatory criteria, standards as well as the better quality control with excellent workmanships have led to this world's best performance today. Sucessful development work of fuel and materials, in fundamental work as well as in practical work through the co-operation among research organizations, university people, reactor, fuel and cladding tube manufacturers together with electric utilities has contributed a great deal to achieve this. Improvement and standardization work followed by the development of ALWR and then next-generation LWR have been going on by reflecting our experiences during 18 years of operation as well as the results of our technological development in fuel and materials.

1202. DATA BASE ON DOSE REDUCTION RESEARCH PROJECTS FOR NUCLEAR POWER PLANTS (in English). T.A. KHAN and J.W. BAUM. Nuclear Regulatory Commission, Washington, DC (USA). Div. of Regulatory Applications ; Brookhaven National Lab., Upton, NY. NUREG/CR--4409-Vol.3, May 1989, 181 pp.

This is the third 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 data base maintained by Brookhaven National Laboratory's ALARA Center for the Nuclear Regulatory Commission. This report presents information on 80 new projects, covering a wide area of activities. Projects on steam generator degradation, decontamination, robotics, improvement in reactor materials, and inspection techniques, among others, are described in the research section. The section on health physics technology includes some simple and very cost-effective projects to reduce radiation exposures. Collective dose data from the United States and other countries are also presented. In the conclusion, we suggest that although new advanced reactor design technology will eventually reduce radiation exposures at nuclear power plants to levels below serious concern, in the interim an aggressive approach to dose reduction remains necessary.

1203. DEVELOPMENT OF HPLC TECHNIQUES FOR THE ANALYSIS OF TRACE-METAL SPECIES IN THE PRIMARY COOLANT OF A PRESSURIZED-WATER REACTOR. K.R.P. BARRON. Council for National Academic Awards, London (UK) , 1988, 267 pp.

The need to monitor corrosion products in the primary circuit of a pressurized water reactor (PWR), at a concentration of 10pg ml^{-1} is discussed. A review of trace and ultra-trace metal analysis, relevant to the specific requirements imposed by primary coolant chemistry, indicated that high-performance liquid chromatography (HPLC), coupled with preconcentration of sample was an ideal technique. A HPLC system was developed to determine trace metal species in simulated PWR primary coolant. In order to achieve the desired detection limit an on-line preconcentration system had to be developed. Separations were performed on Aminex A9 and Benson BC-X10 analytical columns. Detection was by post-column reaction with Eriochrome Black T and Calmagite Linear calibrations of 2.5-100ng of cobalt (the main species of interest), were achieved using up to 200ml samples. The detection limit for a 200ml sample was 10pg ml^{-1}. This study in conjunction with work carried out at Winfrith, resulted in a monitoring system that could follow changes in coolant chemistry, on deposition and release of metal species in simulated PWR water loops, On-line detection of cobalt at 11pg ml^{-1} was recorded.

1204. RADIATION MONITORING TRAINING SYSTEM - AN AID TO PERFORMANCE-BASED TRAINING. C.L. BOUDREAUX. Oak Ridge National Lab., TN (USA). Report No. CONF-890470--, CONF-890470--, Apr 23-27 1989, pp. V.B.5.. V.B.5.12.

This paper covers the use of Waterford-3 plant-specific equipment used for hands-on training of plant personnel. Specifically, the Radiation Monitoring Training System is discussed; however, the application applies to any component or system utilized to support a performance-based training program.

1205. EVOLUTION OF TECHNICIAN AND MAINTENANCE TRAINING. D.R. CLIFTON. , Gatlinburg, TN, USA, Apr 23-27 1989. Oak Ridge National Lab., TN (USA). Report No. CONF-890470--, pp. V.B.4.1-V.B.4.22.

In the past five years Maintenance Training at Diablo Canyon Power Plant has evolved into a sophisticated and strongly supported organization. The key elements in the successful transition have been management support and plant specific hands-on training equipment.

1206. ANNUAL CRITIQUE: AN EFFECTIVE PROGRAM EVALUATION TOOL. T.J. WALL and L.J. CKENZIE. Oak Ridge National Lab., TN (USA). Report No. CONF-890470--, Apr 23-27 1989, pp. IV.B.3.1-IV.B.3.16.

Accreditation by the Institute of Nuclear Power Operations (INPO), indicates a Utility has made a formal commitment to a systematic approach to training. Duke Power Company had implemented INPOs Training System Development (TSD) model to achieve accreditation of its programs in 1986 The last phase of the five step model includes a systematic evaluation of the effectiveness of the training program. This evaluation relies on data collected from the client group, and the training group's ability to respond and affect needed changes to the training program. This paper will discuss using an annual critique to accomplish specialization of on-the-job (OJT) training requirements for the Health Physics discipline at Duke Power Company. The discussion details the feedback process that lead to specialization,the process involved to get changes made, and the cost savings and results of implementing these changes. The paper also addresses the client-training group relationship that created the ability to make this happen.

1207. TEAM MAINTENANCE AS A RESULT OF TEAM TRAINING. S.K. STONE, D.R. LAVENDER, and P.C. MCANULTY. Oak Ridge National Lab., TN (USA). Report No. CONF-890470--, Apr 23-27 1989, pp. III.B.6.1-III.B.6..

Nuclear station maintenance crews have long confronted complex jobs and adverse conditions. Crews often consist of diverse station and vendor personnel who may never have worked together as a team. They are given highly technical tasks that are further complicated by radiological controls, interfacing station groups, and shift turnovers. These complications, combined with the requirements of high quality and job execution standards,tend to result in extended down-time, increased radiation exposures and considerable stress on maintenance crews. A task involving these complications is the replacement of reactor coolant pump (RCP) seals. The advantages of reactor coolant pump seal team training are noted as follows: (1) improves job performance and reduces time required to execute seal replace; (2) improves maintenance team cooperation and minimizes inter-personnel conflicts; (3) improves interfacing group relationships and defines responsibilities; (4) improves communication and minimizes non-productive time during shift turnovers; (5) validates procedures; and (6) identifies special tooling.

1208. OCCUPATIONAL DOSE REDUCTION AT NUCLEAR POWER PLANTS: ANNOTATED BIBLIOGRAPHY OF SELECTED READINGS IN RADIATION PROTECTION AND ALARA (in English). T.A. KHAN and J.W. BAUM. Nuclear Regulatory Commission, Washington, DC (USA). Div. of Regulatory Applications ; Brookhaven National Lab., Upton, NY. NUREG/CR--3469-Vol.4, Jun 1989, 121 pp.

This report is the fourth in the series of bibliographies supporting the efforts at the Brookhaven National Laboratory on dose reduction at nuclear power plants. Abstracts for this bibliography were selected from proceedings of technical meetings, journals, research reports and searches of the DOE's Energy Data Base. The abstracts included in this report to operational health physics as well as other subjects which have a bearing on dose reduction at nuclear power plants, such as stress corrosion, cracking, plant chemistry, use of robotics and remote devices, etc. Material on improved design, materials selection, planning and other topics which are related to dose reduction efforts are also included. The report contains 327 abstracts as well as subject and author indices. All information in the current volume is also available from the ALARA Center's bulletin board service which is accessible by personal computers with the help of a modem. The last section of the report explains the features of the bulletin board. The bulletin board will be kept up-to-date with new information and should be of help in keeping people current in the area of dose reduction.

1209. ROLE OF THE CHEMISTRY IN THE OCCUPATIONAL DOSE CONTROL IN NUCLEAR POWER PLANTS (in Spanish). M.A. BLESA. Boletin de la Socidad Argentina de Radioproteccion (Argentina), Jul 1988, pp. 5-17.

The safety and radioprotection problem in nuclear power plants does not only concern the plants in operation, but it also includes the design, building and decommissioning stages. The factors that determine the radiation field development and the possibility of diminishing them when they reach critical values are presented. Here are considered pressure vessel-heavy water reactors and particularly the radionucleides coming from the products of structural material corrosion. These products are removed by decontaminating compounds and primary circuit systems in general. In accordance with ALARA criterium, the factors that make the decontamination process advisable are analyzed. Firstly, the number of collective doses is discused. In case of heavily contaminated components there is another limitation to the ALARA criterion: the limits of individual dose in a fixed period of time (year, trimester, etc). Among the various decontamination processes-physical or chemical - the stages to follow just in chemical procedures are stated. As Atucha I and II Power Plants are uniques it is necessary to be ready to solve problems. The research and development programs of National Atomic Energy Commission have produced very valuable results such as the 'in situ' activation model and the HERO (highefficient electrochemical removal of oxides) decontamination procedure.

1210. CONTROL OF BWR RADIATION BUILDUP WITH SOLUBLE ZINC (in English). W. MARBLE and C.J. WOOD. Proceedings of a Symposium - National Association of Corrosion Engineers annual meeting and materials performance and corrosion show, Boston, Massachusetts, USA, CONF-850311--, Mar 25-29 1985, 37 pp.

Recent analysis of plant data and laboratory experiments have demonstrated that the presence of dissolved zinc in BWR reactor water minimizes radiation buildup in the primary system. The soluble zinc acts as a corrosion inhibitor for the stainless steel and thus limits the thickness of the oxide film on BWR piping surfaces.

1211. OPTIMIZATION OF A PWR DECONTAMINATION PROCESS (in English). J. TOROK and J.L. SMEE (Atomic Energy of Canada Ltd., Chalk River, Ontario (CA)). Proceedings of a Symposium - National Association of Corrosion Engineers annual meeting and materials performance and corrosion show, Boston, Massachusetts, USA, CONF-850311--, Mar 25-29 1989, 38 pp.

PWR decontamination involves oxidation of the oxide film resulting in a partial removal of chromium, followed by the reductive dissolution of the remainder of the film using the CAN-DECON process. Three oxidizing reagents were assessed. The application of several oxidizing-reducing treatments resulted in a very effective removal of activity from PWR system surfaces.

1212. PROCEEDINGS OF THE TOPICAL CONFERENCE ON THEORY AND PRACTICES IN RADIATION PROTECTION AND SHIELDING. (in English). Topical conference on theory and practice in radiation protection and shielding. American Nuclear Society (La Grange Park, IL), Apr 22-24 1989, 648 pp.

This book contains 75 selections. Some of the titles are: A useful guideline for design of the labyrinth extension wall and/or the shield door; Radiation protection management at Indian Point 3; Verification of a Monte Carlo code system for analysis of ionization chamber responses; Skyshine study for next generation of fusion devices; Future directions in shielding methods and analysis; and The role of shielding for SDI space systems.

1213. METHOD OF DECONTAMINATING EQUIPMENT IN NUCLEAR POWER PLANT (in Japanese). YASUAKI GUNJI, TETSUNORI YASU, and HIRO-SHI. KUNIHARA. Patent No. JP 63-304199/A/, Dec 12 1988, 3 pp.

Physical decontamination is impossible for equipments of complicated shapes such as pipeways, large equipments or those equipments being in use. In view of the above, equipments to be decontaminated are immersed in a decontaminant to which a foaming agent forming fine bubbles are added. In the liquid decontaminant, bubbles are formed and bursted repeatedly and ultrasonic waves are generated upon bursting of bubbles. When the liquid decontaminant is caused to flow recyclically, the bubbles are bursted in the boundary layer and the radioactive corrosion products are defoliated by the vibrations of supersonic waves generated thereby. Thus, deposited radioactive corrosion products can be defoliated and discharged without imparing the durability or the strength of the equipments.

1214. CONTROL ROD FOR USE IN NUCLEAR REACTOR (in Japanese). NORIYUKI NAKAJO, KAORU TA-DA, and SHIN-ICHI ISHII. Toshiba Corp., Kawasaki, Kanagawa (Japan); Nippon Atomic Industry Group Co. Ltd., Tokyo, Sep 30 1988, 5 pp.

Purpose: To reduce the abrasion of rollers and suppress the increase of radiation doses. Constitution: A guide roller is disposed to the upper hand, while a limiter roller is disposed therebelow to the lower hand of a control rod supported vertically movably between fuel assemblies. Each of the guide roller and the limiter roller is made of iron-based alloy, and a support pin for each of the rollers is applied with nitridation at the surface of the iron-based alloy substrate to form a nitride layer thereon. Since the pin substrate is made of the iron-based alloy, the nitridation can be applied easily, abrasion resistance in water is excellent and the abrasion loss is reduced as compared with conventional cobalt or nickel-based alloy. In addition, since each of the rollers is made of iron-based alloy, abrasion with the pin is also reduced. Accordingly, products caused by frictional abrasion are reduced to suppress the increase in radiation doses.

1215. RM-10A ROBOTIC MANIPULATOR SYSTEM. J.R. WHITE, J.B. COUGHLAN, H.W. HARVEY, and R.G. UPTON. Transactions of the American Nuclear Society (USA), Vol. 57, Oct 20 to Nov 4 1988, pp. 335-336.

The REMOTE RM-10A is a man-replacement manipulator system that has been developed specifically for use in radioactive and other hazardous environments. It can be teleoperated, with man-in-the-loop, for unstructured tasks or programmed to perform routine tasks automatically much like robots in the automated manufacturing industry. The RM-10A is a servomanipulator utilizing a closed-loop, microprocessor-based control system. The system consists of a slave assembly, master control station, and interconnecting cabling. The slave assembly is the part of the system that enters the hostile environment. It is man-like is size and configuration with two identical arms attached to a torso structure. Each arm attaches to the torso using two captive screws and two guide pins. The guide pins position and stabilize an arm during removal and reinstallation and also align the two electrical connectors located in the arm support plate and torso. These features allow easy remote replacement of an arm, and commonality of the arms allow interchangeability. The water-resistant slave assembly is equipped with gaskets and O-ring seals in the torso and arm and camera assemblies. In addition, each slave arm's elbow, wrist, and tong are protected by replaceable polyurethane boots. An upper camera assembly, consisting of a color television (TV) camera, 6:1 zoom lens, and a pan/tilt unit, mount to the torso to provide remote viewing capability.

1216. PRACTICABLE ASSAY SYSTEM FOR RADIONUCLIDE QUANTIFICATION OF DISPOSAL PACKAGES (in English). R.G. POST. Proceedings of a Symposium - Waste management '88, Tucson, AZ , Feb 28 to Mar 3 1988, pp. 429-436.

The requirements for land disposal of LLW from nuclear power plants and facilities have necessitated a more complete and accurate analysis of the radionuclide contents of waste packages. JGC has developed a new direct assay technique based on gamma-ray spectroscopy and total gamma-ray counting, combined with a scaling factor methodology for difficult-to-measure nuclides. The system consists of an HpGe detector, a plastic scintillator, a microcomputer and a waste package handling system such as a turntable. The radioactivity concentrations of Co-60 and Cs-137, which are key nuclides for difficult-to-measure nuclides, are calculated from the activity ratio of Co-60 to Cs-137 measured by using an HpGe detector and total radioactivity is measured by using a plastic scintillator. The concentrations of difficult-to-measure nuclides in a waste package are calculated by combining the radioactivity concentrations of Co-60 and Cs-137 with the waste package data and scaling factors. The system is simple and enables a complete analysis of all nuclides specified prior to shipment and disposal of waste packages, and also ensures that not only homogeneous solidified waste but also nonhomogeneous DAW (dry active waste) can be measured within a short time.

1217. EFFECTIVENESS AND SAFETY ASPECTS OF SELECTED DECONTAMINATION METHODS FOR LWRS RECONTAMINATION EXPERIENCE 1988. S.W. DUCE. , Gaithersburg, MD (USA). Nuclear Regulatory Commission, WA(USA). Office of Nuclear Regulatory Research, BNL, Upton, NY (USA). NUREG/CP--0097-Vol.1, Oct 1988, pp. 33-46.

This paper presents information on the recontamination of recirculation piping in commercial boiling water reactors following successive chemical decontaminations or pipe replacement. Several types of pipe pre-treatments have been used at different facilities where the recirculation pipe were replaced to reduce the rate at which radionuclides were incorporated into the oxide films on the inner pipe surfaces. These pipe treatments are briefly discussed and net contamination control effects of the treatments are compared. Net contamination control effects of successive chemical decontaminations on non-replaced pipe is also discussed.

1218. OPERATION OF FINNISH NUCLEAR POWER PLANTS. QUARTERLY REPORT, 4. QUARTER, 1988 AND THE ANNUAL SUMMARY(in English).

C. OTTOSSON. Finnish Centre for Radiation and Nuclear Safety, Helsinki (Finland). Dept. of Nuclear Safety. Report No. STUK-B-YTO--58, May 1989, 30 pp.

This general review of the operation of the Finnish nuclear power plants concentrates on such events and discoveries related to nuclear and radiation safety as the regulatory body, the Finnish Centre for Radiation and Nuclear Safety, regards as noteworthy. The report also includes a summary of the radiatic.. safety of the personnel and the environment, as well as tabulated data on the production and load factors of the plants. In the report period, no event essentially degraded plant safety nor posed a radiation hazard to the personnel or the environment.

1219. OPERATION OF FINNISH NUCLEAR POWER PLANTS. QUARTERLY REPORT, 1. QUARTER 1988 (in English). R. HAENNINEN. Finnish Centre for Radiation and Nuclear Safety, Helsinki (Finland). Dept. of Nuclear Safety. Report No. STUK-B-TYO-52, Sep 1988, 23 pp.

This general review of the operation of the Finnish nuclear power plants concentrates on such events and discoveries related to nuclear and radiation safety as the regulatory body, the Finnish Centre for Radiation and Nuclear Safety, regards as noteworthy. The report also includes a summary of the radiation safety of the personnel and the environment, as well as tabulated data on the production and load factors of the plants. In the report period, no event essentially degraded plant safety nor posed a radiation hzard to the personnel or the environment.

1220. OPERATION OF FINNISH NUCLEAR POWER PLANTS. QUARTERLY REPORT, 4. QUARTER, 1987 AND THE ANNUAL SUMMARY (in English). H. HEIMBURGER. Finnish Centre for Radiation and Nuclear Safety, Helsinki (Finland). Dept. of Nuclear Safety. Report No. STUK-B-YTO--51, Aug 1988, 25 pp.

This general review of the operation of the Finnish nuclear power plants concentrates on such events and discoveries related to nuclear and radiation safety as the regulatory body, the Finnish Centre for Radiation and Nuclear Safety, regards as noteworthy. The report also includes a summary of the radiation safety of the personnel and the environment, as well as tabulated data on the production and load factors of the plants. In the report period, no event essentially degraded plant safety nor posed a radiation hazard to the personnel or the environment.

1221. RADIOLOGICAL CHARACTERIZATION OF NUCLEAR PLANTS UNDER DECOMMISSIONING. PROBLEMS AND EXPERIENCES(in Italian). M. MINCARINI. ENEA, Rome (Italy). Report No. ENEA-RT-PAS--89-2, 1989, 54 pp.

In the present work a description of major problems encountered in qualitative and quantitative radiological characterization of nuclear plants for decommissioning and decontamination purpose is presented. ..erring to several nuclear plant classes activation and contamination processes, direct and indirect radiological analysis and some italian significant experience are descripted.

1222. NOREM WEAR-RESISTANT, IRON-BASED HARD-FACING ALLOYS (in English). Electric Power Research Inst., Palo Alto, CA (USA); AMAX Research and Development Center, Golden, CO (USA). Report No. EPRI-NP--6466-M, Jul 1989, 21 pp.

Wear-resistance cobalt-free hardfacing alloys are needed to replace the cobalt-base alloys used to hardface nuclear valves in order to reduce the exposure of maintenance personnel. Some thirty heats of cast iron-base alloys were prepared and characterized. Selected heats were prepared and applied as hardfacing overlays on austenitic steel substrates using both GTA and PTA welding processes. Some of the iron-base alloys exhibited galling wear resistance as high as that of cobalt-base standards both in the cast condition and in the PTA overlays. Hardness, mechanical properties, and galling wear resistance were determined on weld overlays and on cast alloys. Dilution and thermal expansivity were determined for weld overlays. X-ray diffraction and scanning electron microscopy were used to determine the alloys' microstructures. Other commercially available alloys were tested for galling wear resistance and compared to iron-base alloys.

1223. DECONTAMINATION TECHNIQUES (in English). M. SANDERS and R. BOND. Atom (London) (UK), Mar 1989, pp. 9-12.

Recent decontamination work at the UKAEA's Winfrith site has concentrated on the scientific evaluation and subsequent development of existing physical, chemical and electrochemical processes. Some of the work, such as the decontamination of SGHWR using chemical methods and high pressure water jetting, supports the laboratory's own work programmes. Other techniques such as electrochemical decontamination, vibratory cleaning and wet grit blasting have been developed mainly on behalf of customers in the UK nuclear industry.

1224. SUMMARY, RETROSPECT, AND EVOLUTION OF MOBILE REMOTE SYSTEMS. J. OSBORN, L. HAMPENY, C. ROMME, and W.L. HITTAKER. Transactions of the American Nuclear Society (USA), Vol. 57, Oct 30 to Nov 4 1988, p. 506.

Many applications for mobile remote systems exist in the nuclear industry, particularly where such a work force can reduce human exposures or enable task performance where the capabilities of human workers are inadequate. Despite these opportunities, current remote technology is insufficient. The nuclear industry has pioneered equipment for remote manipulation and some specialized forms of mobility (legged, articulated tracks and hybrid locomotion), but few systems have been developed that integrate suitable manipulation, locomotion, environmental hardening, and other necessary features. A new class of equipment, remote work systems, specialized to nuclear applications, will ensue from this foundation and will be contingent on an understanding of the essential remote work system features and the ability to incorporate them in a capable system. Tasks that are candidates for use of mobile remote equipment exist throughout the life cycle of nuclear industry facilities. These include surveillance and inspections, maintenance of plant equipment, decontamination, waste handling, and decommissioning. Although work site conditions vary greatly and tasks can span such diverse objectives as passive inspection to active demolition, all applications require remote equipment threliable, capable, operable, decontaminable, maintainable, extensible, and compatible with the facility.

1225. OPERATIONS AND ACHIEVEMENTS OF REMOTE EQUIPMENT AT TMI-2 [THREE MILE ISLAND UNIT 2]. M.D. PAVELEK, W. UNDERHILL, J. BOUDREAUX, and F.L. BOZORGI. Transactions of the American Nuclear Society (USA), Vol. 57, Oct 30 to Nov 4 1988, p. 502.

The Three Mile Island Unit 2 (TMI-2) project team evaluated teleoperators and robotic devices to participate in recovery work. The team carefully evaluated available options with client personnel. The goals of an as low as reasonably achievable (ALARA) radiation protection program and safe, efficient cleanup of the facility were the primary program objectives. Teleoperators that met our requirements were not commercially available in 1980. The project team worked closely with Carnegie-Mellon University, which produced with remote reconnaissance vehicles (RRVs) that conducted the major recovery operations in the reactor building basement. The RRVs provided the capability to deliver multitudes of remotely controlled and robotic devices for extended time periods to the reactor building basement at TMI-2, as well as other hostile environments. The teleoperator experience gained in this program is comprehensive and diverse.

1226. PERSONNEL CONTAMINATION PROTECTION TECHNIQUES APPLIED. J.E. HILDEBRAND. Transactions of the American Nuclear Society (USA), Vol. 57, Oct 30 to Nov 4 1988, pp. 481-482.

The severe damage to the Three Mile Island Unit 2 (TMI-2) core and the subsequent discharge of reactor coolant to the reactor and auxiliary buildings resulted in extremely hostile radiological environments in the TMI-2 plant. High fission product surface contamination and radiation levels necessitated the implementation of innovative techniques and methods in performing cleanup operations while assuring effective as low as reasonably achievable (ALARA) practices. The approach utilized by GPU Nuclear throughout the cleanup in applying protective clothing requirements was to consider the overall health risk to the worker including factors such as cardiopulmonary stress, visual and hearing acuity, and heat stress. In applying protective clothing requirements, trade-off considerations had to be made between preventing skin contaminations and possibly overprotecting the worker, thus impacting his ability to perform his intended task at maximum efficiency and in accordance with ALARA principles. The paper discusses the following topics: protective clothing-general use, beta protection, skin contamination, training, personnel access facility, and heat stress.

1227. RADIATION PROTECTION TRAINING THEN AND NOW. T. MULLEAVY. Transactions of the American Nuclear Society (USA), Vol. 57, Oct 30 to Nov 4 1988, pp. 478-479.

Prior to the accident at Three Mile Island, the radiological control training was provided by the health physics and the chemistry departments. The programs that required the input from the health physics department were the Operations Training and the General Employee Training Programs, which were formally organized and presented radiological controls practices as part of the course objectives.

The operations training section periodically requested the health physics department management to provide a radiological portion to their operator qualification or operations requalification class. After the accident in 1979, the General Employee Training Program administration was taken over by the operation training department where it underwent changes with a dedicated effort. The Operations Training Program also underwent a change as well as the Radiological Control Technician Training Program. Each group now has a specialty program designed for their specific needs. These programs have been designed using the Institute of Nuclear Power Operations and regulator guidelines.

1228. COMPUTER SYSTEM DEVELOPMENT TO SUPPORT TMI-2 [THREE MILE ISLAND UNIT 2] RADIOLOGICAL CONTROLS, OPERATIONS, AND RECORDS MANAGEMENT ACTIVITIES. R.D. SCHAUSS. Transactions of the American Nuclear Society (USA), Vol. 57, Oct 30 to Nov 4 1988, pp. 470-471.

The regulations pertaining to the reporting and keeping of records pertaining to occupational radiation exposure of workers employed at US Nuclear Regulatory Commission (MRC) licensed facilities (licensees) are contained in 10CFR19 and 10CFR20, respectively. These regulations provide specific guidance concerning the types of occupational radiation exposure data that must be generated, tracked, reported, and preserved for individuals working at US nuclear power facilities. Certain forms, such as Form NRC-4 Occupational Radiation Exposure History and Form NRC-5 Current Occupational Radiation Exposure, are required to be maintained by the licensee, and specific guidance is provided as to the format and content of data to be documented on these forms. At the time of the accident, Metropolitan Edison Company (the operator of TMI-2), had in place a fairly comprehensive computerized radiation exposure control and record keeping system. The radiation exposure management (REM) system was maintained on the corporate mainframe computer and employed industry-accepted EDP standards for data entry and retrieval. It soon became apparent that the best approach would be to completely redesign the REM system to best meet the special requirements imposed by the accident situatiwas made to design and develop a totally new REM computer system employing on-line transaction processing and an integrated data base management system.

1229. THE TMI-2 [THREE MILE ISLAND UNIT 2] REACTOR BUILDING GROSS DECONTAMINATION EXPERIMENT: EFFECTS ON LOOSE-SURFACE CONTAMINATION LEVELS. E.N. LAZO. Transactions of the American Nuclear Society (USA), Vol. 57, Oct 30 to Nov 4 1988, pp. 455-456.

In March 1982, the Gross Decontamination Experiment was conducted in the Three Mile Island Unit 2 (TMI-2) reactor building. The intent of the experiment was twofold: (a) to determine which of several commonly used decontamination techniques would be the most efficient at reducing contamination levels on vertical and horizontal surfaces and (b) to actually reduce radiation and surface contamination levels in the accessible areas of the reactor building in order to reduce person-rem expenditures for future entries. Accessible areas included the entire reactor building except inside the D-rings, inside the enclosed stairwell, and elevation 282 ft. The experiment was broken into six separate tasks, implemented by nine different work packages, and accomplished during 15 reactor building entries over a 30-day period. Approximately 40 person-rem were expended in completing the experiment. While the results of the experiment did show which decontamination techniques were the most effective, loose-surface contamination and radiation levels in the reactor building were not substantially reduced. The decontamination techniques tested and the key test parameters are shown.

1230. RADIOLOGICAL CONDITIONS AND EXPERIENCES IN THE TMI-2 [THREE MILE ISLAND UNIT 2] AUXILIARY BUILDING. P.E. RUHTER and W.G. URLIENE. Transactions of the American Nuclear Society (USA), Vol. 57, Oct 30 to Nov 4 1988, 449 pp.

The accident at Three Mile Island Unit 2 (TMI-2) created circumstances that have seldom been encountered in the nuclear industry. These circumstances involved high radiation and contamination conditions in areas to which emergency access was required to maintain reactor control. The radiological conditions in the auxiliary building immediately following the accident, how those conditions varied from one location to another, and how they varied with time during the first 6 to 8 months required to return to some degree of normalcy are reviewed. The controls and methods used to maintain radiological exposures as low as reasonably achievable (ALARA) while the necessary work was completed, as well as the circumstances that led to non-ALARA exposures, are also described.

1231. MOBILE ROBOT FOR POWER PLANT INSPECTION AND MAINTENANCE. J.R. WHITE, K.A. ARNSTROM, H.W. ARVEY, R.G. PTON, and K.L. ALKER. Transactions of the American Nuclear Society (USA), Vol. 57, Oct 30 to Nov 4 1988, pp. 329-330.

An all-terrain, mobile robot (called SURBOT-T) has been developed to perform remote visual, sound, and radiation surveillance within contaminated areas of nuclear power plants. The robot can be equipped with a two-armed, telerobotic manipulator system to perform remote maintenance work. The SURBOT-T vehicle has a double-articulating track base that is capable of climbing 45-deg slopes and stairs and over 16-in.-high obstacles. The overall size of SURBOT-T is 28 in. wide by 38 in. long with the front and rear tracks raised and 52 in. high with the camera lowered. With the tracks in a level position, the base provides a sturdy work platform and can ascend/descend stairs without fear of tipping over. The track can be pivoted straight down to elevate the base 14 in. and pass through water up to 24 in. deep. All motors, amplifiers, computer boards, and other electronic components are contained within a sealed housing. The color television camera, spotlight, and directional microphone are mounted on a pan/tilt, which is attached to an elevating mechanism that has 8 ft of vertical travel. An air sampler, radiation detector, and temperature/humidity probe are mounted on the vehicle. The slave manipulator arms on the vehicle can be teleoperated using master artable stand near the control console. They can also be taught to perform motions or tasks by computer control much like robot arms in the automated manufacturing industry.

1232. PRELIMINARY ANALYSIS OF SHIELDING CHARACTERISTICS OF POSSIBLE MATERIALS FOR THE RB REACTOR SHIELDING DESIGN (in English). P. MARINKOVIC, S. VDIC, and M. ESIC Deutsches Atomforum e.V., Bonn (Germany, F.R.) ; Kerntechnische Gesellschaft e.V., Bonn (Germany, F.R.). Proceedings of a Symposium - Annual meeting on nuclear technology (JK '89), CONF-890540--, May 9-11 1989, pp. 607-610.

Published in summary form only.

1233. DESIGN EXPERIENCE IN MINIMIZING RADIATION EXPOSURE TO PERSONNEL DURING MAINTENANCE OF BWR POWER PLANTS (in English). J.R. PUNCHES. Journal of Engineering for Gas Turbines and Power (USA), Vol. 110, Oct 1988, pp. 664-665.

As nuclear power plants become older, in-vessel maintenance to high radiation components will become more common. This paper describes General Electric's experience in making such repairs on Boiling Water Reactors and how the use of engineering and project management enhances the productivity of special maintenance projects. Design techniques employed to minimize personnel radiation exposure are discussed. Specialized remote, automatic and semi-automatic equipment designs are presented. Included are sample descriptions of special tools developed, problems encountered, and lessons learned.

1234. AIRBORNE CARBON-14 ACTIVITIES IN THE WEST VAULT OF THE UNIT 1 REACTOR AT PICKERING NGS (in English). K.E. CURTIS. Ontario Hydro, Toronto, ON (Canada). Research Center. Report No. OH--85-248-K, Oct 1985, 21 pp.

A large sampling and analysis program has been carried out to measure the airborne carbon-14 activities in the west vault during shock heating of four fuel channels in Pickering NGS Unit 1. Particulate carbon-14 activities varied from 0.02 to 2.9 $\mu Ci/m^3$, depending on the sampling location and the fuel channel undergoing shock heating. By contrast, the gaseous carbon-14 activities were relatively constant for all samples, ranging from 1.6 to 5.2 $\mu Ci/m^3$. Greater than 98% of this activity was found to inorganic, probably from $(CO_2)^{14}$. Tritium was also found in the gaseous samples at an average concentration about seven times higher than the carbon-14 activity.

1235. STRACTURAL MATERIAL FOR NUCLEAR REACTOR PRIMARY COOLANT (in Japanese). HIROSHI URUMA and TAKAO BABA (Nippon Atomic Industry Group Co. Ltd., Tokyo (Japan)). Apr 26 1989, 5 pp.

The inner surface of structural materials in contact with coolants containing radioactive impurities in nuclear reactor primary coolant systems is applied with thin austenite stainless steel sheet, for example, by lining thereby forming one or plurality layers of releasable metal membranes. After the plant has been operated for a certain period of time and when corrosion layers are formed at the surface of the metal membranes and the radioactivity contained in the membranes exceeds an allowable level, the pipeway portion is dismantled and the metal membranes are separated and recovered. Since the radioactivity is contained in the metal membranes, it can be removed nearly up to 100 %. Since the released metal membranes can be compressed into an extremely compact structure, the processing is facilitated. Further, if noble metal such as platinum is used as the metal membranes, since the membranes are not attacked, exchange work for the metal membranes is no more necessary.

1236. FEATURES OF IMPROVED TECHNOLOGY IN TOMARI NUCLEAR POWER STATION, HOKKAIDO ELECTRIC POWER CO. (in Japanese). YOSHIAKI YASUI and TETSUO. KITAMURA (Hokkaido Electric Power Co., Inc., Sapporo (Japan)). Nippon Genshiryoku Gakkaishi (Japan), Vol. 31, May 1989, pp. 541-548.

In Tomari Nuclear Power Station, as the first nuclear power station in Hokkaido Electric Power Co., No.1 and No.2 plants began simultaneously construction in August, 1984, and at present, No.1 plant is in the final adjustment test just before the start of commercial operation in June, 1989, and the construction of No.2 plant is in progress smoothly, aiming at the start of commercial operation in June, 1991. This power station is the plants, in which the heightening of safety, the improvement of reliability and capacity factor, the reduction of exposure and so on were advanced by adopting the improvement based on the experience of the design, construction and operation of preceding PWRs and the introduction of advanced technologies. In this paper,the advanced technologies in the design and construction of the plants are reported. No.1 and No.2 plants are 17th and 18th PWRs in Japan, and the power output is 1650 MWt and 579 MWe. These are the first PWRs constructed in 50 Hz district, and the electric output was increased as compared with preceding 2-loop plants. It was confirmed by the trial operation that the results of improvement were sufficiently satisfactory.

1237. COLLECTION OF LAWS AND ORDINANCES CONCERNING REGULATION OF ATOMIC ENERGY, 1989 EDITION. (in Japanese). Nuclear Safety Bureau. Taisei (Tokyo, Japan), 1989, 1690 pp.

The collection of the laws and ordinances concerning the regulation of atomic energy, 1989 edition, was published by the Nuclear Safety Bureau, Science and Technology Agency. First, the abbreviated expressions of 56 laws and ordinances are shown. The contents are divided into Part 1: Fundamental laws and ordinances, Part 2: Regulation of nuclear source materials, nuclear fuel materials and nuclear reactors, Part 3: Prevention of radiation injuries due to radioactive isotopes and others, and Part 4: Related laws and ordinances. In Part 1, Atomic Energy Fundamental Act, Act of Institution of Atomic Energy Commission and Nuclear Safety Commission of Japan, Law Concerning the Technical Standard for Prevention of Radiation Injuries and 9 others are included. In Part 2, Law Concerning Regulation of Nuclear Source Materials, Nuclear Fuel Materials and Nuclear Reactors and 45 others are included. In Part 3, Law Concerning Prevention of Radiation Injuries Due to Radioisotopes and Others and 25 others are included. In Part 4, Electricity Enterprises Act, Road Transport and Vehicles Act, Ships' Safety Law, Labor Safety and Hygiene Law, Japan Atomic Energy Research Institute Law and 29 others are included. The contents are those as of November 30, 1988.

1238. PERSONNEL PROTECTION DURING A REACTOR ACCIDENT (in English). V.S. KOSHCHEEV, A.S. KOROSTIN, and S.P. RAJKHMAN. Proceedings of a Symposium - All-Union conference on medical ...cts. Kiev (Ukrainian SSR). International Atomic Energy Agency, Vienna (Austria) ; Ministerstvo Zdravookhraneniya SSSR, Moscow (USSR) ; All-Union. Report No. IAEA-TECDOC--516, Jul 1988, pp. 145-150.

In organizing individual protection and the provision of protective clothing and equipment for accident teams, particular attention must be paid to protection of the respiratory organs. A system of disciplinary barriers, personnel airlocks and gates and transport processing points - with obligatory monitoring of all movements of staff and equipment - is described.

1239. DEMONSTRATION OF RELIABILITY-CENTERED MAINTENANCE (in English). J.G. ANDERSON, M.J. FARRELL, A.J. HORN, E.A. HUGHES, R.E. LEVLINE, M.E. RODIN, and L.C. SOUTHWORTH (Erin Engineering and Research, Inc., Walnut Creek, CA). Report No. EPRI-NP--6152-Vol.2, Sep 1989, 120 pp.

On March 9, 1988 the Southern California Edison Company was selected by the Electric Power Research Institute as one of two utilities to perform a large scale Reliability-Centered Maintenance demonstration project to be co-funded by EPRI and SCE. The overall objective of this project is to demonstrate that RCM can be effectively performed in a plant environment leading to improved overall unit performance. The objective of SCE is to optimize the existing Preventive Maintenance program at San Onofre Nuclear Generating Station Unit 2.

The successful achievement of this goal requires the active involvement and the cooperative efforts of the operations and Maintenance Support, Maintenance, Station Technical, and Operations Divisions. This report describes progress to date in the RCM core elements such as system selection, applications of the RCM methodology to the systems selected, lessons learned, course corrections, implementation of the RCM recommendations, results achieved, and development of a living program. The overall project involves analysis of approximately twelve to sixteen systems. To date, six have been completed and two are underway. The results to date indicate a significant payback in maintenance cost reduction, even with the expansion of some PM activities for identified critical failure modes.

1240. STANDARD FORMAT AND CONTENT FOR DECOMMISSIONING PLANS FOR LICENSEES UNDER 10 CFR PARTS 30, 40, AND 70 (in English). Nuclear Regulatory Commission, Washington, DC (USA). Office of Nuclear Regulatory Research. Report No. REG/G--3.65, Aug 1989, 13 pp.

This regulatory guide, developed in conjunction with the amendments to the regulations concerning decommissioning, would be applicable to certain licensees (discussed above in Section 1) when they decide to permanently discontinue all licensed activities involving nuclear materials. The purpose of the guide is to identify the information needed by the NRC staff for evaluations involving decommissioning. The guide also provides a format for submitting this information. Conformance with this guide is not required,but its use will facilitate preparation of a decommissioning plan by licensees and timely, uniform review by the NRC staff. A different format will be acceptable to the staff if it provides an adequate basis for approval of a decommissioning plan. The guidance is appropriate for use in license amendments to partially clean up a nuclear facility and release that part for unrestricted use at a time other than at the decommissioning of the facility as a whole, e.g., cleanup of separate buildings. The amended sections of the decommissioning rule (30.36(b), 40.42(b), 70.38(b)) require each licensee to notify the Commission promptly, in writing, and request termination of license when the licensee decides to terminate all activities involving material authorized under the license.

1241. CALCULATIONAL STUDY OF RADIATION-THERMAL SHIELD WITH FORCED COOLING (in Russian). V.P. PANCHENKO, A.E. FEDOSEEV, and A.E. SAVELLO . Report No. INIS-SU--117, 1989, pp. 19-27.

Results of calculational study of NPP radiation shield cooled by the Field pipes and coils of different orientation, carried out to select the best way of NPP radiation shield cooling with and without regard for stratification of structural composition from cooling devices are presented. The results have shown that to ensure the required temperature regime of the apparatus casing and biological shield tank (BST) the cooling by the Field horizontal pipes proved to be the most acceptable. Moreover, thermal conductivity factor of structural composition 1-5 W/mxK for BST filling does not affect significantly the temperature regime of NPP radiation shield.

1242. A FAST-SORTING MEASUREMENT TECHNIQUE TO DETERMINE DECONTAMINATION PRIORITY. C.H. DISTENFELD, B. BROSEY, and H. IGARASHI. Transactions of the American Nuclear Society (USA), Vol. 57, Oct 30 to Nov 4 1988, 458 pp.

Recovery of large contaminated buildings, such as the Three Mile Island Unit 2 (TMI-2) reactor building, are complicated by ceilings that can be 12 to 13 m high. Much of the overhead space is filled with conduits, pipes, cable trays, ventilation ducts, and steel structures. The total complex surface can greatly exceed the total surface of walls and floors. Concrete pedestals, heavy steel stands, embedded steel rails, refueling mechanisms, and other similar structures complicate normally accessible areas and impede exposure reduction efforts. Initial recovery of contaminated spaces tends to involve treatment of hot spots and accessible spaces such as floor and wall surfaces. Subsequent decontamination may be less efficient since untreated surfaces, such as in overhead spaces, may be beyond the reach of ordinary decontamination tools. To conserve radiation exposure of recovery personnel, it is important to prioritize the effort so that early work provides maximum exposure reduction. Subsequent exposure reduction can then be carried out with less total exposure to recovery personnel. This favorable scenario depends on identification of key surfaces that most affect the exposure rate. The quick-sort method that was developed is based on the Eberline HP 220A directional survey system.

1243. AIRBORNE PARTICLES IN THE VENTILATION SYSTEM OF A BWR. L. STROEM. , Boston, MA (USA), Aug 22-25 1988. Nuclear Regulatory Commission, Washington, DC (USA). Office of Nuclear Regulatory Research ; Harvard Univ., Boston, MA. NUREG/CP--0098-Vol.2, May 1989, pp. 814-823.

This investigation was undertaken in order to obtain a better description of the aerosol to be monitored by the stack sampler of a nuclear power plant. Between June - August 1986 radioactivity, size and adhesion to solid surfaces of airborne particles were measured at four different places in the ventilation plant of the nuclear power station Barsebaeck 1. About 95% of the particulate radioactivity came from the reactor building, the rest from the turbine building and the waste handling building. Particle median diameter with respect to radioactivity was about 10 μm, with large variations. Nuclides with long half-lives were mainly associated with larger particles. Particle size might be a function of coarse particle loss through sedimentation within the plant. There are also indications of an aerosol formed through resuspension of deposits on the ventilation channel walls. Particle adhesion to a dry, smooth steel surface was low, only 1 - 10% of impacting particles remained on the surface. The risk of overloading the sampling filter with particles under accident conditions should be considered.

1244. MEASURES ON OPERATION RELIABILITY INCREASE AND COLLECTIVE RADIATION DOSE DECREASE AT NUCLEAR POWER PLANTS WITH BWR TYPE REACTORS (in Russian). O.I. MARTYNOVA. Teploenergetika (Moscow) (USSR), Dec 1988, pp. 68-71.

Methods of fight against intercrystalline stress corrosion cracking of BWR reactor pipeline system are described. High efficiency of hydrogen dosing in the NPP operation regime is demonstrated. Measures directed to the reduction of collective radiation doses at American and Japanese NPPs are enumerated.

1245. BWR HYDROGEN WATER CHEMISTRY GUIDELINES: 1987 REVISION (in English). Electric Power Research Inst., Palo Alto, CA (USA). Report No. EPRI-NP--4947-SR, Dec 1988, 124 pp .

Boiling water reactors (BWRs) have experienced stress corrosion cracking in the reactor cooling system piping resulting in adverse impacts on plant availability and personnel radiation exposure. The BWR Owners Group for IGSCC Research and EPRI have sponsored a major research and development program to provide remedies for this stress corrosion cracking problem. Results from this program show that the likelihood of cracking depends on reactor water chemistry (particularly on the concentrations of ionic impurities and oxidizing radiolysis products such as oxygen) as well as on material condition and stress level. Tests have demonstrated that the concentration of oxidizing radiolysis products in the recirculating reactor water of a BWR can be reduced substantially by injecting hydrogen into the feedwater. This report presents suggested generic hydrogen water chemistry specifications, discusses the proposed water chemistry limits, suggests responses to out-of-specification water chemistry, discusses available chemical analysis methods as well as data management and surveillance schemes, and details the management philosophy required to successfully establish and implement a hydrogen water chemistry control program. In addition to pipe cracking, fuel performance and effects on shutdown and operation radiation fields were considered in developing the feedwater and reactor water chemistry specifications presented in the body of the report. An appendix contains recommendations for water quality of auxiliary systems.

1246. RADIOLOGICAL PROTECTION PRINCIPLES FOR THE DECOMMISSIONING AND DISMANTLING OF NUCLEAR POWER PLANT. INTERNATIONAL ACTIVITIES (in German). K.H. SCHALLER Bundesministerium fuer Umwelt, Naturschutz und Reaktorsicherheit, Bonn (Germany, F.R.), Vol 11. Proceedings of a Symposium - Closed meeting of the Strahlenschutzkommission, Gundremmingen (Germany, F.R.), Nov 6-7 1986, pp. 203-223.

The IAEA, NEA of OECD, and EURATOM are international organisations concerned with and cooperating on problems of waste management and radiological safety in relation to the decommissioning of nuclear installations. International activities currently are concerned with the methodology for evaluating radiological consequences of the management and treatment of very low-level solid waste from decommissioning, criteria for setting up a time schedule for decommissioning, dumping sites for disposal, and the institutions responsible for planning and surveillance of disposal of solid wastes from decommissioning in EC member states.

1247. ENRICHED BORIC ACID FOR PWR APPLICATION: COST EVALUATION STUDY FOR A TWIN-UNIT PWR (in English). J.A. BATTAGLIA, R.M. WATERS, J.M. VON HOLLEN, L.A. LAMATIA, C.A. BERGMANN, and S.M. DITOMMASO. Report No. EPRI-NP--6458, Sep 1989, 103 pp.

In the nuclear industry boric acid dissolved in the reactor coolant is used as a soluble reactivity control agent. Reactivity control in nuclear plants is also provided by neutron absorbing control rods. This neutron absorbing duty is distributed between the control rods and soluble boric acid in such a way as to provide the most economical split. Typically, the control rods take care of rapid reactivity changes and the boric acid handles the slower long term control of reactivity by varying the boric acid concentrations within the reactor coolant. In PWR reactor plants the dissolved boric acid is referred to as a soluble poison or chemical shim due to the high capacity for thermal neutron capture exhibited by the boron-10 isotope contained in the boric acid molecule. This slow reactivity change or chemical shim control would otherwise have to be performed using control rods, a much more expensive proposition. Reactivity changes are controlled by the B-10 isotope by virtue of its very high cross section (3837 barns) for thermal neutron absorption. However, natural boron contains only 20 atom percent of the B-10 isotope and essentially all the remaining 80 percent is the B-11 isotope. The B-11 isotope of cross section .005 barns is essentially of no use as a neutron absce B-11 makes up the bulk of the total boron present and contributes little to the nuclear operation it would seem logical to eliminate this isotope of boron from the boric acid molecule. In so doing boric acid concentration in operating PWR plants need only be a fraction of that existing to accomplish identical nuclear operations. However, to achieve the elimination of B-11 from NBA (Natural Boric Acid) an isotope separation must be performed.

1248. CORROSION-PRODUCT RELEASE IN LIGHT WATER REACTORS (in English). D. LISTER and R.D. DAVIDSON. Report No. EPRI-NP--6512, Sep 1989, 107 pp.

This is the final report of a research program aimed at measuring and studying the release of corrosion products from typical PWR and BWR materials to reactor coolant. The program has provided measurements of release from stainless steel steam generator alloys and hard-facing material (Stellite) to PWR coolant under several chemistry conditions. Kinetic expressions for cumulative release as a function of time have been developed. Corrosion measurements in- and out-reactor have indicated little effect of reactor radiation on corrosion of these materials. Detailed surface analysis has characterized the formation of oxide films in PWR coolant, and has led to suggestions of mechanisms of release. The mechanisms have been made the basis of a system model which has been used to evaluate the effects of various system parameters on the concentration of dissolved cobalt in the coolant--i.e., on the source term for activity transport. The understanding of film formation and release have led to a proposed method of preconditioning PWRs to reduce substantially radiation fields during subsequent operation. Correlations for elemental release from stainless steel and Stellite under BWR conditions have also been derived. They indicate that cobalt-based alloys in BWR reactor circuits arejor source of corrosion-released cobalt. The effects of zinc on the growth of oxide films on carbon steel, Inconel-600 and Stellite-6 are also described.

1249. ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS. W.J. SHACK, T.F. KASSNER, P.S. MAIYA, J.Y. PARK, and E. RUTHER. . Nuclear Regulatory Commission, Washington, DC (USA). Div. of Engineering. NUREG--0975-Vol.7, May 1989, pp. 179-200.

Piping in light-water-reactor (LWR) power systems has been affected by several types of environmental degradation. Intergranular stress corrosion cracking (IGSCC) of austenitic stainless steel (SS) piping in boiling-water reactors (BWRs) has required research, inspection, and mitigation programs that will ultimately cost several billion dollars. As extended lifetimes are envisaged, other potential environmental degradation problems such as corrosion fatigue must be considered. The objective of this program is to develop an independent capability for the assessment of environmentally assisted degradation in light water reactor (LWR) systems.

1250. REDUCTION OF COBALT 60 INVENTORY IN LIGHT WATER REACTORS (in English). B.C. FRIEDRICH. Kerntechnik (F.R. Germany), Vol. 54, Aug 1989, pp. 109-113.

The contamination of primary water carrying systems be cobalt 60 has been identified as a main contributor to the radiation dose accumulated by maintenance personnel. Analytical models determined Co-based hardfacings as the main sources of cobalt 60. Research and development projects qualified alternative low-cobalt hardfacings. These replacement alloys were applied in recent power plants. Measurements have proved a dose rate reduction of about 50% for the first replacement step. A reduction by a total factor of 3 is expected for the next step. Additional dose rate reduction can be achieved by optimization of water chemistry.

1251. NUCLEAR SAFETY (in English). E.G. SILVER (ed.). Oak Ridge National Lab., TN (USA). Report No. TPR-NS--30-3, 1989, 159 pp.

This document is a review journal that covers significant developments in the field of nuclear safety. Its scope includes the analysis and control of hazards associated with nuclear energy, operations involving fissionable materials, and the products of nuclear fission and their effects on the environment. Primary emphasis is on safety in reactor design, construction, and operation; however, the safety aspects of the entire fuel cycle, including fuel fabrication, spent-fuel processing, nuclear waste disposal, handling of radioisotopes, and environmental effects of these operations, are also treated.

1252. SUMMARY OF WORK ON CHARACTERIZATION OF THE RADIOACTIVE DEPOSITS ON PWR PRIMARY CIRCUIT SURFACES (in English). M.E. PICK. Decontamination and decommissioning of nuclear facilities. Final report of three research co-ordination meetings held between 1984 and 1987. International Atomic Energy Agency, Vienna (Austria), Jun 1989, pp. 79-92.

In examining decommissioning strategies for LWR's and the possible role of decontamination, characterization of the radioactive deposits on circuit surfaces is required to provide information on the radioactive inventory and the type of oxide on the surface. Knowledge of the latter will determine which is the most appropriate decontamination process to use and its potential efficiency. Results from examinations performed on a number of Inconel 600 steam generator and stainless steel PWR specimens and also a limited number of BWR and CANDU specimens are summarized. A variety of techniques have been utilized including: gamma spectrometry, alpha spectrometry, scanning electron microscopy and wet chemical analysis. In addition, preliminary studies using secondary ion mass spectrometry (SIMS) have been performed. The sources of the major radionuclides present on circuit surfaces are also considered.

1253. DATA BASE FOR MAN-REM MANAGEMENT AT MADRAS ATOMIC POWER STATION (in English). K.R. VISWAMBHARAN, B.S.K. NAIR, B. RAMAMIRTHAM, R.S. VARADHAN, and K. CHUDALAYANDI. Bulletin of Radiation Protedtion (India), Vol. 11, Mar 7-9 1988, pp. 53-55.

In view of the observed steady increase in the collective dose to the radiation workers at MAPP, a sound data base is essential to ensure an effective exposure control programme. A new system of data collection and analysis of the exposure to the station workers is discussed in the paper.

1254. SOURCEBOOK FOR CHEMICAL DECONTAMINATION OF NUCLEAR POWER PLANTS (in English). C.J. WOOD and C.N. SPALARIS. Electric Power Research Inst., Palo Alto, CA (USA). Nuclear Power Div.. Report No. EPRI-NP--6433, Aug 1989, 118 pp.

This sourcebook provides information on the chemical decontamination of nuclear power plants. An overview of the current status of the technology is given, including a brief description of commercially-available processes. BWR recirculation piping and PWR steam generator decontaminations are described, with a comparison of the two types of operation. Corrosion data, methods of reducing recontamination rates, and wastes issues are discussed. Cost benefit methodologies, planning for decontamination and utility lessons-learned are outlined. Future developments, including full-system decontaminations, are reviewed.

1255. IMPLEMENTATION OF REMOTE EQUIPMENT AT TMI-2 [THREE MILE ISLAND UNIT 2]. D. GIEFER and A.B. JEFFRIES. Transactions of the American Nuclear Society (USA), Vol. 57, Oct 30 to Nov 4 1988, pp. 501-502.

Each of the remote vehicles in use, or planned for use, at Three Mile Island Unit 2 (TMI-2) during the period from 1984 to the present had certain distinct common features. These were proven to be desirable for remote application in the TMI-2 environment. Proper implementation requires consideration of the following: control systems, rigging systems, power supplies, operator/support interface, maintenance concerns, viewing systems, contamination control, and communications. Design and component fabrication of these features allowed deployment of each of the remote devices. This paper discusses these systems and their impact for the use of remote mobile equipment at TMI-2. In most cases, the means of implementation dictated the design features of the devices.

1256. CRITERIA DEVELOPMENT OF REMOTELY CONTROLLED MOBILE DEVICES FOR TMI-2 [THREE MILE ISLAND UNIT 2]. R. FILLNOW, P. BENCEL, and D. GIEFER. Transactions of the American Nuclear Society (USA), Vol. 57, Oct 30 to Nov 4 1988, 500 pp.

Since 1982, GPU Nuclear Corporation has used a series of remote mobile devices for data collection and cleanup of highly contaminated areas in the Three Mile Island Unit 2 (TMI-2) nuclear facilities. This paper describes these devices and the general criteria established for their design. Until 1984, the remote equipment used at TMI was obtained from industry sources. This included devices called SISI, FRED, and later LOUIE-1. Following 1984, the direction was to obtain custom-made devices to assure a design that would be more appropriate for the TMI-2 environment. Along with this approach came more detailed criteria and a need for a thorough understanding of the task to be accomplished by the devices. The following families of equipment resulted: (1) remote reconnaissance vehicles (RRVs), (2) the LOUIE family, and (3) remote working vehicle (RWV) family.

1257. DOSE REDUCTION: TRIM PERSONNEL TRANSIT EXPOSURE TO THE RB DEFUELING PLATFORM AND REDUCE DEFUELING EXPOSURE. W.J. COOPER and P.P. VELEZ. Transactions of the American Nuclear Society (USA), Vol. 57, Oct 30 to Nov 4 1988, 485 pp.

The October 1984 supplement dealing with occupational radiation doses for the cleanup of Three Mile Island Unit 2 (TMI-2) estimated total doses of 13,000 to 46,000 person-rem for the cleanup program. It now appears likely that the actual total dose incurred for the cleanup through the proposed monitored storage conditions will be on the order of 5,000 person-rem. A major contributor to this dramatically lower dose is the effort expended on dose reduction for the defueling tasks. GPU Nuclear Corporation radiological engineering began the process by performing a detailed review of the known condition of the reactor building itself, determining where radiation sources external to the defueling system were that could affect the dose rate to the defueling workers and potential sources within the system itself. It became apparent that the 9-mrem dose each worker received while transiting to the defueling work area as a result of major sources in other areas of the building would require dose reduction efforts. As part of the dose reduction activities, the major identified sources were decontaminated, shielded, or removed. Three sources of worker exposure in the vessel were dominant in system design: the dissolved gamma-emitting isotopes in the water, the fuel, and the contaminants. To reduce exposure from these sources, the working platform is 6 in. of steel for shielding. Tools were designed to prevent removing a tool from the water accidentally, and the loaded canisters are under several feet of water under the shielded platform.

1258. TMI-2 [THREE MILE ISLAND UNIT 2] REACTOR BUILDING DOSE REDUCTION TASK FORCE. R.S. DANIELS. Transactions of the American Nuclear Society (USA), Vol. 57, Oct 30 to Nov 4 1988, pp. 484-485.

In late October 1982, the director of Three Mile Island Unit 2 (TMI-2) created the dose reduction task force with the objective of identifying the principal radiological sources in the reactor building and recommending actions to minimize the dose to workers on labor-intensive projects. Members of the task force were drawn form various groups at TMI. Findings and recommendations were presented to the US Nuclear Regulatory Commission in a briefing on November 18, 1982. The task force developed a three-step approach toward dose reduction. Step 1 identified the radiological sources. Step 2 modeled the source and estimated its contribution to the general area dose rates. Step 3 recommended actions to achieve dose reductions consistent with general exposure rate goals.

1259. RESPIRATORY PROTECTION-LESSONS LEARNED AT THREE MILE ISLAND. E. F. GEE. Transactions of the American Nuclear Society (USA), Vol. 57, Oct 30 to Nov 4 1988, pp. 480-481.

At the time of the Three Mile Island Unit 2 (TMI-2) accident, GPN Nuclear was ill prepared for the respiratory protection demands were about to be endured. Although a recognized respiratory protection program was in place that permitted application of protection factors, the depth and detail needed to attack an accident and recovery process of this magnitude was lacking. Airborne radioactivity concentrations following the accident reached 1000 times the maximum permissible concentration. Surface contamination levels ranged upward of 1 rad/smear. Oxygen concentrations in the reactor building were % prior to the first purge. A multitude of short- and long-term challenges necessary to expand the scope of the program was faced. Some of those most significant are discussed. Immediate problems included respiratory equipment inventory, self-contained breathing apparatus charging capabilities and qualified respirator wearers. Long-term problems included the following: training; selection, issue, and use of respirators; equipment cleaning, maintenance, and inspection; breathing air supplies; and emergency preparedness.

1260. AN UPGRADED PERSONNEL DOSIMETRY SYSTEM FOR TMI-2 [THREE MILE ISLAND UNIT 2]. J.W. SCHMIDT and HARWORTH J.M.. Transactions of the American Nuclear Society (USA), Vol. 57, Oct 30 to Nov 4 1988, pp. 479-480.

Following the Three Mile Island Unit 2 (TMI-2) accident,it was identified that due to the unusual radiological conditions created, an improved thermoluminescent dosimetry (TLD) system was needed to support the cleanup and recovery. The deficiencies of the existing system were identified as an unsuitable dosimeter design and limited system automation available to support the ~6000 dosimeters being processed monthly for record dose. As a result, a Panasonic-based TLD personnel dosimetry system was developed and installed by GPU Nuclear at the TMI facility. The components of this dosimetry system include a dosimeter design and associated interpretation algorithm, an extensive quality assurance program, and a computer-based dosimeter processing system. This dosimeter/algorithm design provides for the use of a changing beta correction factor (BCF), which is derived from beta spectral data collected by the dosimeter. The system computer-based processing equipment is driven using software developed to be user friendly, totally menu driven, and geared toward the implementation of an extensive quality assurance program for a production dosimetry system. In total, this software consists of over 95 programs that specifically support written dosimetry procedures.

1261. DEVELOPMENT OF ALL METAL FILTERS FOR USE IN THE NUCLEAR INDUSTRY. H. RANDHAHN and B. GOTLINSKY. Proceedings of the 20th DOE/NRC nuclear air cleaning conference. Sessions 6--15, Boston, MA (USA). Nuclear Regulatory Commission, Washington, DC (USA). Office of Nuclear Regulatory Research ; Harvard Univ., Boston, MA. NUREG/CP--0098-Vol.2, May 1989, pp. 726-748.

This paper summarizes the experiments performed by Pall Corporation Laboratories in Germany for qualification of a filtration system for venting of P.W.R. The basis for the parameters examined were different model calculations for a possible venting in P.W.R. The calculations gave process specifications for a filter venting system. Aside from principal demands on mechanical and thermal strength of the system, efficiency examinations under differing process conditions were necessary. Here the simulation of these possible process conditions by the test parameters used were essential. From the model calculations for examination of filter performance the following fluid conditions were chosen - dry to wet gas at temperatures up to 160 °C and absolute pressures of 5.5 bar. Aerosols that are not affected in size, shape or other physical properties by those fluid parameters had to be used and under these conditions aerosol generation and measuring systems had to be evaluated. The important size range for this was below 1 micrometer, since these have been shown to be the hardest to remove.

1262. MILLIREMS, MICROSIEVERTS AND RADIATION DOSE MANAGEMENT (in English). F.P. YOUELL. Proceedings of a Symposium - Conference on nuclear power station project management, Blackpool (UK), Jul 20-22 1988, pp. 51-56.

Whilst the radiological impact of nuclear power stations on members of the public and station staff is under constant scrutiny, the generally low level of impact under normal operating conditions is not sufficiently appreciated. This paper sets down the radiological criteria against which the AGR stations have been designed and describes the measures taken to reduce the radiation exposure of members of the public and station staff.

1263. METHOD OF REDUCING RADIOACTIVITY IN NUCLEAR POWER PLANT (in Japanese). YOSHITAKA NISHINO, TOSHIO SAWA, KATSUMI OSUMI, and HISAO ITO. Mar 14 1989, 11 pp.

In primary coolant circuits of a nuclear power plant, since the conversion ratio to nickel ferrite due to reaction between iron ingredient (iron hydroxide, oxide and ion) and nickel is low, the yield of iron cruds is increased and no sufficient reduction can be attained for the radioactivity. Accordingly, the primary coolants,are recycled under heating at the presence of beryllium to form membranes mainly composed of nickel ferrite and cobalt ferrite to the surfaces of pipeways and equipments in the primary coolant circuits. That is, the iron cruds, nickel and cobalt are converted into insoluble nickel ferrite and cobalt ferrite at high conversion rate and rapidly under the catalytic effect of beryllium. Accordingly, radioactivity concentration of cobalt-58 and cobalt-60 in the reactor water is reduced, by which the surface dose of pipeways and equipments are reduced to remarkably decrease the operators' exposure dose.

1264. EFFORTS ON EXPOSURE REDUCTION AT SHIMANE POWER PLANT (in Japanese). MASAYOSHI SHIRAISHI and HIROAKI DANDA. Karyoku Genshiryoku Hatsuden (Japan), Vol. 40, Apr 1989, pp. 407-412.

In nuclear power stations, the exposure dose of workers has been kept below the standard values determined by the relevant laws, and the efforts for restricting exposure dose have been repeated in conformity with the principle of ALARA. As the result, the total exposure dose in Japanese nuclear power stations in each fiscal year showed the tendency of gradual decrease in spite of the increase of the number of workers accompanying the increase of the number of plants. This is largely due to up-to-date plants in which the improvement and standardization of LWRs were adopted, and exposure dose is restricted to very low level. Particularly in BWR plants, the plants which started operation in 1981 or earlier are responsible for most of exposure dose. Therefore, in order to promote exposure reduction further, the countermeasures to those plants are required. In Shimane Nuclear Power Station, Exposure Reduction Investigation Committee was organized at the end of fiscal year 1986, and the lowering of radiation level and the shortening of working hours have been studied. In Shimane Nuclear Power Station, two BWR plants of 460 MWe and 820 MWe output are in operation. 81 items of the countermeasures were found, and 42 items were materialized already. The remarkable reduction of exposure was obtained.

1265. EVALUATION METHOD FOR EFFECTIVE DOSE EQUIVALENT IN INTERNAL EXPOSURE (in Japanese). NOBUYUKI TAKEUCHI, TAKEHIKO EMOTO, and KUNIHIRO MURAMATSU. Fuji Jiho (Japan), Vol. 62, May 1989, pp. 339-344.

According to the recommendation of the ICRP Publ. 30, Fuji Electric has developed a method of evaluating internal exposure of radiation-area workers as an effective dose equivalent in a specific period for the purpose of radiation management in nuclear power plants and others. This method is to evaluate a derived approximation with a personal computer. By numerical processing with a large-capacity computer, we evaluated the applicability of this approximation method with regard to some nuclides. The result showed that sufficient accuracy for practical use could be obtained with appropriate evaluation periods, for example, about three days for the class D such as ^{137}Cs and about three months for the class Y such as ^{60}Co.

1266. OUTLINE OF DESIGN, CONSTRUCTION AND START-UP TEST OF HAMAOKA NUCLEAR POWER STATION UNIT NO.3 (in Japanese). AKIHIKO ITOH. Karyoku Genshiryoku Hatsuden (Japan), Vol. 40, May 1989, pp. 531-536.

August 1987, Hamaoka Nuclear Power Station, Unit No.3, 1,100-MWe BWR plant, with the improved Mark-I containment vessel started its commercial operation. This plant reflects Japanese Plant Design Improvement and Standardization Program and adopts many design features. This paper briefly outlines design, construction and start-up operation of Hamaoka Unit No.3.

1267. FINANCIAL IMPACT OF IMPLEMENTING DRAFT ANSI STANDARD N13.30, PERFORMANCE CRITERIA FOR RADIOBIOASSAY (in English). R.J. TRAUB and J.A. MACLELLAN. Nuclear Regulatory Commission, Washington, DC (USA). Div. of Regulatory Applications ; Pacific Northwest Lab., Richland, WA. NUREG/CR--5396, Jul 1989, 66 pp.

In order to establish standards of bioassay performance upon which a uniform national program of performance testing might be based, the Health Physics Society Standards Committee (HPSSC) formed Working Group 2.5 to prepare the draft American National Standards Institute, Inc. (ANSI) Standard N13.30, Performance Criteria for Radiobioassay. Because the US Nuclear Regulatory Commission (NRC) is considering whether to require that all bioassay service laboratories meet the criteria of the draft Standard, the NRC staff requested that the Pacific Northwest Laboratory (PNL) estimate the costs that may be incurred in implementing the draft Standard. Two types of laboratories were involved in the cost study: service laboratories and a performance testing laboratory. Cost estimates based on responses to questionnaires sent to seven facilities performing radiobioassays varied in relation to the extent of each facility's radiobioassay program and to their perception of their readiness for accreditation. Implementation of accreditation was estimated to cost each facility between a few thousand dollars to over one-quarter million dollars depending on the facility type. Likewise,annual costs ranged from a few hundred dollars to $170,000. For all but one facility, annual costo be less than $25,000. In addition to these costs, startup cost for the [illegible]ing laboratory were estimated to be approximately $322,000.

[illegible] EXPOSURE OF THE US POPULATION FR[illegible] OCCUPATIONAL RADIATION (in E[illegible]n). National Council on Radiation Protection [illegible] Measurements, Bethesda, MD (USA). Report [illegible]. NCRP--101, Jun 1 1989, 106 pp .

The National Council on Radiation Protection and Measurements (NCRP) has reviewed the literature on the subject of occupational exposure to ionizing radiation to assemble, in a single document, the effective dose equivalents and the collective effective dose equivalents for the work force. Although achieving this goal involved the difficulties, the current Report provides evidence that occupational exposures are responsible for only a small fraction of the total collective effective dose equivalent for the entire US population. The current Report is intended primarily for the information of the radiation protection community, but it should also be of interest to the layperson for the perspective it provides on the contribution of occupational exposures to the total population exposure.

1269. OPERATIONAL POWER REACTOR HEALTH PHYSICS (in English). B.A. WATSON. Proceedings of a Symposium - 21. mid-year topical meeting of the Health Physics Society, Bal Harbour, FL (USA), Dec 13-17 1987, pp. 1-12.

Operational Health Physics can be comprised of a multitude of organizations, both corporate and at the plant sites. The following discussion centers around Baltimore Gas and Electric's (BG and E) Calvert Cliffs Nuclear Power Plant, located in Lusby, Maryland. Calvert Cliffs is a twin Combustion Engineering 825 MWe pressurized water reactor site with Unit I having a General electric turbine-generator and Unit II having a Westinghouse turbine-generator. Having just completed each Unit's ten-year Inservice Inspection and Refueling Outge, a total of 20 reactor years operating health physics experience have been accumulated at Calvert Cliffs. Because BG and E has only one nuclear site most health physics functions are performed at the plant site. This is also true for the other BG and E nuclear related organizations, such as Engineering and Quality Assurance. Utilities with multiple plant sites have corporate health physics entity usually providing oversight to the various plant programs.

1270. HEALTH PHYSICS PROBLEMS RESULTING FROM POWER REACTOR OPERATION WITH FAILED FUEL (in English). R.V. WARNOCK, W.F. RIGBY, and E.M. GOLDIN. Proceedings of the 21. midyear topical meeting of the Health Physics Society., Dec 13-17 1987, pp. 113-120.

This paper discusses the source, detection, characteristics, hazards, and control of microscopic, irradiated reactor fuel fragments (fleas). An approach to predicting the presence of fleas in reactor systems is suggested. Information and described hot particle controls are based on experience from two fuel cycles and two refuelings at San Onofre, a site with two 1100 MWe Combustion Engineering PWRs.

1271. THE IMPACT OF FUEL CLADDING FAILURE EVENTS ON OCCUPATIONAL RADIATION EXPOSURES AT NUCLEAR POWER PLANTS (in English). M.P. MOELLER, G.F. MARTIN, J.L. KENOYER, G.A. STOETZEL, and H.J. VANDERMOLEN. Proceedings of a Symposium - 21. mid-year topical meeting of the Health Physics Society, Bal Harbour, FL (USA), Dec 13-17 1987, pp. 107-112.

This paper summarizes two case studies that evaluated the impact of fuel cladding failures on occupational radiation exposures at pressurized water reactors (PWRs). For the case studies, radiation measurements were made both during routine operations and during the subsequent maintenance and refueling outage ata PWR with more than 0.2% failed fuel. Gamma spectroscopy measurements, radiation exposure rate determinations, thermoluminescent dosimeter assessments, and air sample analyses were made in the plant's radwaste, pipe penetration, and containment buildings. Small highly radioactive fuel particles, which contaminated the plant's fuel handling building, were also analyzed. Based on the data collected, fuel cladding failures increased radiation exposure rates an estimated 540% in some areas of the plant during routine operations. Furthermore, the fraction of the total exposure rates due to fission products remained relatively constant over the duration of the outage.

1272. DESIGN AND PERFORMANCE OF THE CONTROL ROD DRIVE FLUSH TANK AT BROWNS FERRY NUCLEAR PLANT (in English). F.S. TSAKERES, H.W. DEASON, A.W. SORRELL, R.E. KNIGHT, and S.R. HOWARD. Proceedings of a Symposium - 21. mid-year topical meeting of the Health Physics Society, Bal Harbour, FL (USA), Dec 13-17 1987, pp. 269-278.

An in-house Control Rod Drive (CRD) flush tank was designed, built and used to reduce dose rates and control radioactive contamination during CRD maintenance at Browns Ferry Nuclear Plant (BFN). This tank was designed to reduce personnel exposure routinely experienced during refurbishment of the reactor CRD mechanism as well as minimize airborne and surface contamination levels. Initial results using the flush tank indicated that administrative, engineering, and protective device controls could be reduced. In addition,special mechanical devices were developed to allow effective remote decontamination of the CRD and other associated components thereby further reducing personnel radiation exposure.

1273. METHOD OF CONTROLLING IRON CONCENTRATION IN FEEDWATER OF NUCLEAR POWER PLANT (in Japanese). KOJI. KUBO. Toshiba Corp., Kawasaki, Kanagawa (Japan). May 11 1989, 4 pp.

A pipeway for entering heater drain water of a feedwater heater into feedwater and a pipeway of entering heater drain water to the upstream of the condensater and/or condensate clean-up device are disposed and the ratio of the flow rates in these pipeways are controlled depending on the iron concentration and the nickel concentration in the feedwater, as well as the nickel concentration and the cobalt concentration in the reactor water. Thus, the amount of the heater drain water flowing into the feedwater is adjusted to control the iron concentration in the feedwater. Accordingly, it is possible to control the iron concentration in the feedwater rapidly and accurately, thereby decreasing the radioactivity in a nuclear power plant and reducing the operators exposure dose.

1274. GENERIC COST ESTIMATES FOR RETROFIT ACTIVITIES AT NUCLEAR POWER PLANTS. F.W. SCIACCA, G.P. SIMION, and S. FELD. Transactions of the American Nuclear Society (USA), Vol. 57, Oct 30 to Nov 4 1988, pp. 207-208.

Producing realistic and consistent cost estimates of construction-related activities at nuclear power plants is essential to determine the potential economic impact on the utility, as well as to assist in evaluating economically sound alternatives. Several studies completed recently under the sponsorship of the US Nuclear Regulatory Commission deal specifically with physical modification costs at nuclear plants. The approach developed utilizes median baseline costs for labor, equipment, and materials associated with new plant construction as a starting point. These costs are then adjusted to reflect actual conditions existing at operating or nearly completed nuclear plants. Successive sets of factors are used to estimate the total resource requirement, as well as aspects such as worker radiation exposure. This methodology allows analysts to generate reasonable cost estimates quickly, it requires only a modest amount of user input, and it helps identify and quantify the cost elements involved in retrofit activities. Major cost categories associated with physical modifications at nuclear power plants are illustrated. The generic cost-estimating approach requires the analyst to consider some or all cost aspects depending on the complexity of the retrofit, the size of aother constraints. For comparisons of estimated versus actual costs, ~48 cost data points were obtained. This generic methodology produced cost estimates that, on the average, were within 15% of the actual cost incurred by the utilities.

1275. UPGRADING RADIOLOGICAL WORK PRACTICES THROUGH EMPLOYEE PARTICIPATION. W.L. BECKMAN. Transactions of the American Nuclear Society (USA), Vol. 57, Oct 30 to Nov 4 1988, pp. 62-63.

Following the shutdown of the Midland Plant in 1984, Consumers Power Company found itself with a need to reorganize its nuclear operations department. The reorganization took place in November 1984. At that time the plant was just completing an intermittent outage that had begun in September 1983. Over the previous 2 yr,the plant had expended 1,500 person-rem and generated 1330 m^3 of radioactive waste. In addition, an overexposure and a shipping violation in early 1984 had contributed to a deteriorating regulatory picture for radiological services. In an attempt to understand the problem confronting the plant, the new organization set up a group of meetings for each department to identify their barriers to becoming a high-performance organization. These meetings, which became known as barrier meetings, were used to identify barriers to performance, such as overly restrictive requirements, excessive paperwork, facility limitations, and improper job assignments. Following the barrier meetings, corrective actions and individuals responsible for completing these actions were identified. Most recent efforts have been in upgrading radiological work practices.

1276. TRAMP URANIUM. E.S. HENDRIXSON and T.G. WILLIAMSON. Transactions of the American Nuclear Society (USA), Vol. 57, Oct 30 to Nov 4 1988, pp. 229-230.

Many utilities have implemented a no leaker philosophy for fuel performance and actively pursue removing leaking fuel assemblies from their reactor cores whenever a leaking fuel assembly is detected. Therefore, the only source for fission product activity in the RCS when there are no leaking fuel assemblies is tramp uranium. A technique has been developed that strips uranium impurities from $ZrCl_4$. Unless efforts are made to remove natural uranium impurities from reactor materials, the utilities will not be able to reduce the RCS specific ^{131}I activity in PWRs to below the lower limit of $\sim 1.0 \times 10^{-4}\ \mu Ci/g$.

1277. ROTATING MACHINERY SURVEILLANCE SYSTEM REDUCES PLANT DOWNTIME AND RADIATION EXPOSURE. J.S. BOHANICK and J.C. ALLEN, J.W. ROBINSON. Transactions of the American Nuclear Society (USA), Vol. 57, Oct 30 to Nov 4 1988, pp. 197-198.

A rotating machinery surveillance system (RMSS) was permanently installed at Grand Gulf nuclear station (GGNS) as part of a program sponsored by the US Department of Energy whose goal was to reduce radiation exposure to power plant personnel resulting from the inspection, maintenance, and repair of rotating machinery. The RMSS was installed at GGNS in 1983 to continuously monitor 173 analog vibration signals from proximity probes mounted on 26 machine trains and {approximately}450 process data points via a computer data link. Vibration frequency spectra, i.e., the vibration amplitude versus frequency of vibration, and various characterizations of these spectra are the fundamental data collected by the RMSS for performing machinery diagnostics. The RMSS collects vibration frequency spectra on a daily basis for all the monitored rotating equipment and automatically stores the collected spectra for review by the vibration engineer. Vibration spectra automatically stored by the RMSS fall into categories that include the last normal, alarm, minimum and maximum, past three-day data set, baseline, current, and user-saved spectra. During first and second fuel-cycle operation at GGNS, several significant vibration problems were detected by the RMSS. Two of ter: recirculation pumps and turbine-generator bearing degradation. The total reduction in personnel radiation exposure at GGNS from 1985 to 1987 due to the presence of the RMSS was estimated to be in the range from 49 to 54 person-rem.

1278. DESIGN MEASURES TO MEET SAFETY REQUIREMENTS (in English). B.V. GEORGE and J.R. APPLEBY. Proceedings of the Institution of Mechanical Engineers, Vol. 203, Dec 3 1987, pp. 123-128.

Nuclear safety is concerned with acceptably limiting the exposure of the general public and the station's operating staff to the ionizing radiation produced as a byproduct of the nuclear fission process, while preserving the economic and environmental benefits of nuclear power. Exposure to radiation can arise as a result of normal operation of the plant, e.g. due to operating staff having to work in areas where radiation levels are relatively high, and to the public, due to planned radioactive discharges from the station. It can also arise as a result of accidents, causing radioactive material to be released from the plant. The achievement of safety of nuclear plant is a continuous process throughout the life of the plant. It is based on the application of certain fundamental principles during the design, construction, commissioning and operation of the plant, and does not end until the plant is de-commissioned and dismantled. This paper concentrates on what is done to achieve nuclear safety during the first part of this process, the design stage, and is illustrated by reference to the proposed design of the Sizewell B reactor.

1279. THE COST OF OPERATING WITH FAILED FUEL AT VIRGINIA POWER. C.A. FORD. Transactions of the American Nuclear Society (USA). CONF-881011--, Vol. 57, Jan 12-16 1988, pp. 95-97.

Virginia Power has completed a study of the costs incurred due to fuel failures in its pressurized water reactors. This study was prompted by histories of high primary coolant activity and subsequent fuel inspections at the North Anna and Surry power stations. The study included an evaluation of the total costs of fuel failures as well as an evaluation of the economics of postirradiation fuel inspections. The major costs of fuel failures included personnel radiation exposure, permanently discharged failed fuel, radwaste generation, increased labor requirements, containment entry delays due to airborne radioactivity, and ramp rate restrictions. Although fuel failures affect a utility in several other areas, the items evaluated in the study were thought to be the most significant of the costs. The study indicated that performing a postirradiation failed fuel examination can be economically justified at tramp-corrected ^{131}I levels of 0.015 μCi/g. The savings to the utility can be on the order of several million dollars. Additionally, the cost penalty of performing a fuel inspection at lower iodine levels is generally in the range of $200,000. This economic penalty is expected to be outweighed by the intangible benefits of operating with a defect-free core.

1280. HUMAN ENGINEERING IN MOBILE RADWASTE SYSTEMS. DD. JONES, J. MCMAHON, and G. MOTL. Transactions of the American Nuclear Society (USA). CONF-881011--, Vol. 56, Jun 12-16 1988, 83 pp.

To a large degree, mobile radwaste systems are replacing installed plant systems at US nuclear plants due to regulatory obsolescence, high capital and maintenance costs, and increased radiation exposure. Well over half the power plants in the United States now use some sort of mobile system similar to those offered by LN Technologies Corporation. Human engineering is reflected in mobile radwaste system design due to concerns about safety, efficiency, and cost. The radwaste services business is so competitive that vendors must reflect human engineering in several areas of equipment design in order to compete. The paper discusses radiation exposure control, contamination control, compact components, maintainability, operation, and transportability.

1281. DEVELOPMENT OF ADRA (in English). J.S. BRTIS. Proceedings of a Symposium - Topical meeting on artificial intelligence, Snowbird, UT, Aug 31 - Sep 2 1987, pp. 599-606.

All nuclear power plant modifications must be designed in a way that keeps the radiation dose to radiation workers and the public As Low As is Reasonably Achievable (ALARA). This is not only a good design practice, it is required by the NRC in accord with Title 10 of the Code of Federal Regulations). As a result, millions of dollars are spent each year developing, analyzing, and justifying the ALARA design of plant modifications. Providing the necessary design reviews is a major challenge to the industry. In response, Sargent and Lundy has developed ADRA (the ALARA Design Review Assistant), a microcomputer-based system that aids in developing, reviewing, and documenting good ALARA designs. ADRA is now in an early operational stage, undergoing testing at Sargent and Lundy. This paper reviews the findings and results of the ADRA development effort.

1282. CONSIDERATION OF RADIATION PROTECTION OF WORKERS IN THE DESIGN AND OPERATION OF NUCLEAR PLANTS. PT. 2. OPERATION (VERSION 6/89). AS OF JULY 12, 1989 (in German). Bundesanzeiger (F.R. Germany), Vol. 41, Jun 12-16 1988, pp. 5-11.

The standard has to be applied for fixing protective measures for the people working in the nuclear power station with regard to exposure to radiation. It concerns the measures which are necessary for activities during operation as authorized and the planning of measures with regard to the incidents and accidents defined according to appendix I StrlSchV. With regard to the protection of the people working in the nuclear power station against exposure to radiation it has the objective to ensure that the radiation protection rules are transformed into appropriate organizing and technical protective measures according to paragraph 28 section 1 StrlSchV.

1283. VOICE COMMUNICATION SYSTEMS COMPATIBLE WITH RESPIRATORY PROTECTION (in English). M.D. DONOVAN, J.H. BAILEY, and R.G. AYERS. Arinc Research Corp., Annapolis, MD (USA); Electric Power Research Inst., Palo Alto, CA (USA). Report No. EPRI-NP--6559, Nov 1989, 88 pp.

The needs and benefits of improved voice communication systems for nuclear plant maintenance workers in respiratory protection devices are discussed. The results of a literature survey are presented as are results of on-site interviews to determine voice communication requirements during maintenance task performance and environmental conditions in which communication equipment is used. General technical requirements for a voice communication system compatible with respiratory protection devices are defined. From the results of a survey of communication equipment vendors, it is concluded that radio systems that meet most of these requirements are commercially available but are not in widespread use in the nuclear industry. The experience of current users of these communication systems, as determined during interviews at several nuclear plants, is discussed. Guidelines for the selection, use, and maintenance of voice communication systems are presented.

1284. SHIELDING DEVICE (in English). L.D. BRAHM, F.C. BROWN, D.F. CIARLONE, M.R. DEDRICH, J.B. HASSALL, and F.M. SALVA. US Patent Number: US4865801, Sep 12 1989.

This patent describes in a boiling water nuclear reactor having an under-vessel area with a plurality of downwardly extending control rod drives below the vessel,each control rod drive having a flange assembly that is a source of radiation and comprising a control rod drive housing flange having a base and a mating control rod drive flange having a base, with the flanges in each assembly being jointed together at their bases by connecting means, and usually having position indicating apparatus extending through the flange assemblies, with the areas beneath the flange bases each defining an underflange zone proximate the source of radiation, a shielding device capable of being mounted and demounted relative to a zone. The shielding device comprising: a structure for disposition around the flange bases; the structure being configured to substantially surround the zone in its mounted condition; attachment means associated with the structure, for mounting and demounting the structure relative to the flange assembly;an amount of shielding material sufficiently effective to substantially reduce the measured radiation dose rate emanating from the zone; and with the shielding material comprising at least a portion of the structure.

1285.DOSE MANAGEMENT; WEIGHING UP THE COSTS AND BENEFITS OF REMOTE MAINTENANCE (in English). C. LEFAURE, J. LOCHARD, and A. BLAIN. Nuclear Engineering International (UK), Vol. 34, Jun 1989, pp. 38-40.

Reducing occupational exposure during maintenance, while at the same time preserving the economic viability of nuclear power plants, will be one of the major challenges to be faced by the industry over the next decade. Increased use of robotics is a possibility. The costs and benefits are examined in this paper.

1286. CONTROLLING OCCUPATIONAL DOSE AT HUNGARY'S PAKS ORMAI, P.; ROSA, G.; HORVATH, E.; RONAKY, J.; SZABO, I.C.; GERMAN, E. (in English). PAKSI ATOMEROEMUE VALLALAT (HUGARY). Nuclear Engineering International (UK), Vol. 34, Jun 1989, pp. 40-41.

Occupational dose at the Hungarian Paks plant is one of the lowest in the world, with an average collective dose of 1.3Sv/GWy generated. The dose limitation system is based on the principles laid down by ICRP publication No 26 and on the Hungarian national regulations. Workers are regularly monitored for external radiation doses by checking film badges and TL dosemeters. The internal radiation burden is also routinely controlled by examinations of the whole body (the thyroid in particular) and by urine analysis. About 90 per cent of the collective dose comes from refuelling and maintenance. Dose distribution among the personnel is quite even with no extremely high individual doses. With the exception of a few cases, there was no evidence of any individual receiving more than 15mSv/y in the five year operational period.

1287. WATER CHEMISTRY PORBLEMS IN NUCLEAR POWER INDUSTRY (in English). B OMBAY BHABHA ATOMIC RESEARCH CENTRE (India). Proceedings of the Indian National Science Academy, Vol. 54, 1988, pp. 729-736.

The paper discusses the mechanism of corrosion in reactor vessel and piping in a pressurised water reactor. The main cause is the altered behaviour of water due to changes in its physical properties at high temperature and high pressure. The ^{60}Co produced from small impurity of cobalt in stainless steel is responsible for the radio-activity in out-of-core piping. The crud consists of corrosion products having particle size greater than 0.4 μm.

1288. SAFETY ASPECTS OF THE US ADVANCED LMR DESIGN (in English). D.R. PEDERSEN, G.L. GYOREY, J.F. MARCHATERRE, and S. ROSEN. Argonne National Lab., IL (USA). CONF-890841--3, 1989, 15 pp.

The cornerstones of the United States Advanced Liquid Metal Cooled Reactor (ALMR) program sponsored by the Department of Energy are: the plant design program at General Electric based on the PRISM (Power Reactor Innovative Small Module) concept, and the Integral Fast Reactor program (IFR) at Argonne National Laboratory (ANL). The goal of the US program is to produce a standard, commercial ALMR, including the associated fuel cycle. This paper discusses the US regulatory framework for design of an ALMR, safety aspects of the IFR program at ANL, the IFR fuel cycle and actinide recycle, and the ALMR plant design program at GE.

1289. DEPOSITION OF COBALT ON SURFACE-TREATED STAINLESS STEEL UNDER PWR CONDITIONS (in English). D.H. LISTER, P.G. ANDERSON, B.J. BARRY, and R.G. LAVOIE. Electric Power Research Inst., Palo Alto, CA; Atomic Energy of Canada Ltd., Chalk River, ON . Report No. EPRI-NP-6528, Oct 1989, 60 pp.

As part of an on-going program aimed at reducing radiation exposures in light water reactors, the modification of surfaces to minimize their propensity to pick up radioactivity under reactor conditions has been studied. This report describes how stainless steel specimens, surface-treated with a variety of processes, picked up Co-60 from high-temperature water under PWR conditions in a high-pressure loop. The build-up of activity was monitored on-line with a movable gamma spectrometer. Off-line counting at the end of the experiment established the absolute activity levels, and selective examinations with SEM and metallography characterized the surface condition of the exceptional specimens. The effectiveness of the surface treatments was gauged by fitting simple parabolae to the activity build-up data and comparing the coefficients with those obtained from untreated control specimens.

1290. RADIATION-RELATED IMPACTS FOR NUCLEAR PLANT PHYSICAL MODIFICATIONS (in English). F. SCIACCA, R. KNUDSON, G. SIMION, G. BACA, H. BEHLING, K. BEHLING, W. BRITZ, and S. COHEN. Nuclear Regulatory Commission, Washington, DC; Science and Engineering Associates, Inc., Albuquerque, NM (USA). NUREG/CR--5236, Oct 1989, 55 pp.

The radiation fields in nuclear power plants present significant obstacles to accomplishing repairs and modifications to many systems and components in these plants. The NRC's generic cost estimating methodology attempts to account for radiation-related impacts by assigning values to the radiation labor productivity factor. This radiation labor productivity factor is then used as a multiplier on the greenfield or new nuclear plant construction labor to adjust for the actual operating plant conditions. The value assigned to the productivity factor is based on the work-site radiation levels. The relationship among ALARA practices, work-place radiation levels, and radiation-related cost impacts previously had not been adequately characterized or verified. The assumptions made concerning the use and application of radiation-reduction measures such as system decontamination and/or the use of temporary shielding can significantly impact estimates of both labor requirements and radiation exposure associated with a particular activity. Overall guidance was needed for analysts as to typical ALARA practices at nuclear power plants and the effects of these practices in reducing work-site dose rates and overall labor requirements. This effort was undertaken to better characterize that modification cost and radiological exposure impacts related to the radiation environment of the work place. More specifically, this work sought to define and clarify the quantitative relationships between or among: radiation levels and ALARA practices, such as the use of temporary shielding, decontamination efforts, or the use of robots and remote tools; radiation levels and labor productivity factors; radiation levels, in-field labor hours, and worker radiation exposure; radiation levels and health physics services costs; and radiation levels, labor hours, and anti-contamination clothing and equipment.

1291. HEALTH PHYSICS PRACTICES AND EXPERIENCE AT DUKE POWER COMPANY, II (in English). L LEWIS. Proceedings of a Symposium - ANS executive conference on dollars and sense, San Diego, CA, Dec 9-12 1984, pp. 1-2.

This paper briefly describes the Health Physics and ALARA programs, practices and experience at Duke Power Company, particularly at the Oconee Nuclear Station. It consists of three PWR units and has been operating for more than 10 years now. In the beginning, the Corporate System Health Physics staff established the Health Physics Program and later the ALARA Program for the station and an independent station Health Physics organization has conducted the program. The corporate Health Physics organization also provides technical guidance and direction to the station in conducting the program and reviews its effectiveness, modifying it as necessary based on experience and new regulations.

1292. WESTINGHOUSE EXPERIENCE (in English). A.H. FERO. Proceedings of a Symposium - ANS executive conference on dollars and sense, San Diego, CA, Dec 9-12 1984, pp. 1-2.

The experience at Westinghouse has been that radiation exposure management does, in fact, make sense both philosophically and economically. However, it is economics which truly drives the innovation process. Almost every significant change which has the effect of reducing occupational radiation exposure does so in such a way as to also reduce the cost of operatig and maintaining a nuclear power plant. This paper demonstrates this idea through a discussion of four examples: the recent replacement of the steam generators at the Point Beach Nuclear Power Plant; control of reactor coolant system chemistry; elimination of the RTD bypass loop; and the development of the Remotely Operated Service Arm.

1293. CONDENSATE POLISHING IN BWR COOLING SYSTEM WITH HOLLOW FIBER FILTER (in English). KAZUYA YAMADA, TAKAMORI SHIRAI, FUMIO TAJIMA, H. EHLING, K. BEHLING, W. BRITZ, and S. COHEN. NAIG Annual Review (Japan), 1989, pp. 73-75.

Published in summary form only.

1294. NPP OPERATION AND MAINTENANCE WITH FRENCH-BUILT NPPS. J.J. MIRA and S. CHARBONNEAU. Transactions of the American Nuclear Society(USA), Vol. 56, Sep 7-11 1987, pp. 866-872.

In France, 80% of the electricity production will be nuclear in 1990. More than to-day, PWR units will be operated by Electricite de France on a load-follow basis. Every effort is made to reduce planned and forced outages. Maintenance is shared between EdF and Framatome,the latter being in charge of high technology operations. All these actions are eased by the standardization of units, within each power class, the resulting build-up of experience being available to all PWR operators in the world, more particularly those of Framatome-built units.

1295. NUCLEAR SAFETY ENDEAVOR IN KOREA. S.H. LEE. Transactions of the American Nuclear Society (USA), Vol. 56, Sep 7-11 1987, pp. 808-815.

Korea's nuclear pow plant program is growing. As it grows, nuclear safety becomes an important issue. This article traces the development of Korean nuclear power program, the structure of the nuclear industries, the Nuclear Safety Center and its roles in the regulation and licensing of nuclear power plant, and also identifies some of the activities carried out to enhance the safety of nuclear power plants.

1296. COMPARISON OF NUCLEAR PLANT EMERGENCY PLANS OF PBNCC MEMBERS. W .Y. KATO and J.H. HOPWOOD. Transactions of the American Nuclear Society (USA), Vol. 56, Sep 7-11 1987, pp. 696-702.

The Nuclear Safety Working Group (NSWG) of the Pacific Basin Nuclear Corporation Committee initiated cooperation among Pacific Basin areas based primarily around emergency planning. The NSWG conducted a review of the emergency response plans of members. This paper briefly reviews and makes a comparison of the emergency response plans, with particular attention on the response organization, the planning zone, and the protective action guidelines for emergencies. Although all areas have adopted the same basic elements of emergency planning and are similar, there are also variances due to different governmental structures, population densities, and available resources. It is found that the most significant difference is in the size of the emergency planning zone. The paper concludes with a discussion on possible future cooperative activities of the working group.

1297. PROGRAMS TO IMPROVE PLANT PERFORMANCE. N.L. FELMUS. Transactions of the American Nuclear Society (USA), Vol. 56, Sep 7-11 1987, pp. 679-689.

Looking toward the 1990's, the authors see a period in which the industry will face the challenge of improving the performance of the nuclear plants which are built and operating. The skills and technology are at hand to make good plant performance a reality, and the time has come to use them to achieve that end. As reserve margins decline, utilities and their regulators will increasingly seek to tap the unexploited capacity tied up in plants operating below their optimum availability. This paper describes a number of the programs, plant improvements and operations improvements which can yield a significant increase in nuclear plant availability and capacity factor now and into the 1990's.

1298. IMPROVEMENT OF NUCLEAR POWER PLANT OPERATION AND MAINTENANCE IN JAPAN. K. HAMAZAKI. Transactions of the American Nuclear Society (USA), Vol. 56, Sep 7-11 1987, pp. 610-617.

Following the inauguration of commercial nuclear power generation in Japan in 1966, LWR capacity factors were held in the relatively low level until around 1975 due to initial-period troubles. With subsequent improvement, however, capacity factors have climbed steadily and recently been sustaining more than 70%. To obtain this successful result, a various kind of improvement have been made not only for the operation management area but also for the maintenance management area in conjunction with the successive effort to reflect the operating experiences to the early stage design. Nowadays nuclear generation has assumed increasing importance for Japan's electrical power needs, and is making a great contribution to stabilizing power supply costs.

1299. NUCLEAR SAFETY REGULATION IN THE PEOPLES REPUBLIC OF CHINA. S. GUANG-CHANG. Transactions of the American Nuclear Society (USA), Vol. 56, Sep 7-11 1987, pp. 336-344.

The present report gives a general view of how the problem of nuclear safety is dealt with in China, with particular reference to the nuclear power plants. The most relevant nuclear legal regulations and procedures are reported. Organization of the National Nuclear Safety Administration (NNSA) of China and its working activities are presented. The report gives also the principle and practice with regard to licensing process and regulatory inspection of nuclear power plant in China. A general outline of research and development programs and activities after Chernobyl accident is also discussed.

1300. METHOD OF MONITORING NUCLEAR REACTOR PIPEWAYS (in Japanese). KUNIO ENOMOTO, KUNIO HASEGAWA, TASUKU SHIMIZU, MAKOTO HAYASHI, MASAHIRO OTAKA, SATOSHI SUGANO, TAKASHI SAITO, and HIDEYO SAITO. Hitachi Ltd., Tokyo (Japan). Dec 23 1988, 4 pp.

Light emitting layers coated with scintillator films is disposed at the outer surface of a pipeway through which radioactive material-containing fluids flow. Then, since the leaking radioactive rays at the surface of the tube are different depending on the abnormality and normality at the inside of the tube, the abnormality at the inner surface can be inspected intuitively by the light emitted from the scintillator film disposed to the outer surface of the tube. The scintillator film is appended to the pipeway by way of adhesives, heat resistant tape and sensitizing foil. A protection film is disposed to the other surface of the scintillator film. As the scintillator film, heat resistant paint of water glass-incorporated sodium aluminate mixed with fluorescent material such sodium iodide, cesium iodide or zinc sulfide is most suitable.

1301. HYDRAULIC PRESSURE DEVICE FOR CONTROL ROD DRIVE (in Japanese). AKIHIRO INOUE. Toshiba Corp., Kawasaki, Kanagawa (Japan), Dec 2 1988, 6 pp.

A hydraulic pressure control unit for controlling the pressure of driving water to be supplied is disposed to control rod drives provided in the lower portion of a pressure vessel, and a required amount of purging water is supplied to the hydraulic pressure control unit upon inserting operation of a control rod. In addition to the supply of a predetermined amount of purging water to the entire control rod drives, an additional supplementary purging water is supplied to the mechanism to be driven to thereby prevent the backward flow of the reactor water to the driving mechanism. In other operation modes,a required amount of purging water is supplid to the entire control rod drives. In this way, it is possible to maintain the control rod drives clean, reduce the radiation dose thereby decrease the operator's exposure dose upon periodical inspection and remarkably improve the operationability in a lower dry well. Further, the plant heat loss can be decreased without impairing the scram performance.

1302 DEVELOPMENT AND PERFORMANCE OF HEAVY WATER SYSTEM AND HELIUM SYSTEM OF ATR FUGEN (in Japanese). NAOKI KITAYAMA, SATOSHI MORITA, SEIICHI KOSHII, YUJI NAKASHIMA, SETSUO IIJIMA, and TOSHIO KAWAHARA. Donen Giho (Japan), Dec 1988, pp. 19-44.

Heavy water and helium system of the Fugen, a 165 MWe prototype of heavy water moderated, boiling light water cooled, pressure tube type reactor, has demonstrated its excellent performance and reliability through over ten years' operation. Heavy water chemistry control method was also established after a number of R and D works against degradation of purification resins with deuteriumperoxide generated from heavy water rediolysis. An overview of the ten years' operational experience is described concerning maintenance works, operational procedures, heavy water chemistry control, and radiation protection methods against tritium internal exposure.

1303. REMOTE TECHNIQUES FOR INSPECTION AND REFURBISHMENT OF NUCLEAR PLANT (in English). British Nuclear Energy Society (London, UK); Proceedings of a Symposuim - International conference on remote techniques for inspection and refurbishment of nuclear plant, Stratford-upon-Avon, UK. CONF-8811215--, Nov 28 - Dec 1 1988, 226 pp.

These proceedings draw on the experience of nuclear engineers from around the world to form a substantial reference work on remote techniques for the inspection and refurbishment of nuclear plant. Remote techniques are being developed to cope with more complex plant, and to enable a wider variety of activities to be undertaken within the reactor core while, at the same time, reducing the exposure of operatives to ionizing radiation. Thus many components of reactors may now be inspected in situ, for example, checking coolant pipes for corrosion and cracking, or locating failed fuel rods. The required repairs can then also be made using remote procedures. The proceedings include over 40 papers covering manipulators and robotics systems, the technology of remote refurbishment, and techniques for visual inspection and non-destructive inspection in Magnox reactors, AGRs. Candu, PWRs, BWRs and FBRs. All are indexed separately.

1304. CLEAN UP ACTS CONDUCTED BY MOBILE ROBOTS IN NUCLEAR FACILITIES (in English). H.B. MEIERAN British Nuclear Energy Society, London (UK) ; Proceedings of a Symposium - International conference on remote techniques for inspection and refurbishment of nuclear plant, Stratford-upon-Avon, UK. CONF-8811215--, Nov 28 - Dec 1 1988, pp. 7-13.

This paper will review some of the roles and methodologies that mobile robots are now assuming in the clean-up and decontamination of nuclear power plants and other nuclear facilities, as well in response to radiological accidents and emergencies. These robots can conduct many tasks and missions that are currently assigned to plant personnel or to emergency response team personnel in the case of accidents. The relative degrees of success and problems experienced by these robots will be identified along with additional missions that the devices could have assumed had the time and the opportunity been available.

1305. SIGHT AND SOUND: DEVELOPMENTS IN REMOTE SENSING (in English). P.K.J. SMITH British Nuclear Energy Society, London (UK) ; Proceedings of a Symposium - International conference on remote techniques for inspection and refurbishment of nuclear plant, Stratford-upon-Avon, UK, Nov 28 - Dec 1 1988, pp. 217-220.

Particular reference is made to fibrescope/endoscopic viewing in conjunction with closed circuit television, in a number of diverse nuclear applications ranging from weld inspection in a Fast Reactor, viewing of irradiated fuel underwater, to special endoscopic viewing in conjunction with remote work packages deployed by manipulators. Coverage is also given to ultrasonic sensing systems for remotely obtaining non destructive examination (NDE) data. A range of fast standpipe scanners has been developed. Other applications include the remote ultrasonic examination of the thickness of corroded pipework in a nuclear plant, and post-weld ultrasonic inspection.

1306. REMOTELY OPERATED INSPECTION EQUIPMENT FOR THE CANDU FUEL CHANNELS (in English). K.S. MAHIL, G.N. JARVIS, and D.W. DONNELLY British Nuclear Energy Society, London (UK) ; Proceedings of a Symposium - International conference on remote techniques for inspection and refurbishment of nuclear plant, Stratford-upon-Avon, UK, Nov 28 - Dec 1 1988, pp. 127-132.

Equipment is described which has been successfully used for the nondestructive inspection of fuel channel components within Ontario Hydro's CANDU nuclear reactors. By the use of automated systems, significant savings in personnel radiation exposure and unit outage duration have been realized, with improved quality and quantity of nondestructive examination information.

1307. ENGLISH TRANSLATION OF THREE DOCUMENT RELATING TO THE SFR-1. OPERATING PERMISSION. RADIATION PROTECTION INSTRUCTIONS. CHAPTER 4 OF THE ASSESSMENT MEMORANDUM (in English). National Inst. of Radiation Protection, Stockholm (Sweden). Report No. SSI--88-21, Sep 26 1988, 61 pp.

After approval from the National Institute of Radiation Protection, (the SSI) on April 26th, 1988 the Swedish Nuclear Fuel and Waste Management Company, the SKB, put the Final Repository for Radioactive Waste, the SFR-1 (Forsmark), into operation. This report contains English translations of the Operating Permission issued by SSI and the associated radiation protection instructions. Also included is a translation of chapter 4, the viewpoints and evaluations, of the Assessment Memorandum which was the background material for the Board of the SSI when deciding on the operational permission.

1308. ESTABLISHING REQUIREMENTS FOR THE NEXT GENERATION OF PRESSURIZED WATERREACTORS - REDUCING THE UNCERTAINTY. W.P. CHERNOCK, W.R. CORCORAN, W.H. RASIN, and K.E. STAHLKOPF. Transactions of the American Nuclear Society (USA). CONF-870905--, Vol. 56, Sep 7-11 1987, pp. 120-126.

The Electric Power Research Institute is managing a major effort to establish requirements for the next generation of US light water reactors. This effort is the vital first step in preserving the viability of the nuclear option to contribute to meeting US national electric power capacity needs in the next century. A major thrust of the program is to reduce the uncertainties which would be faced by the utility executives in choosing the nuclear option. The uncertainties to be reduced include those related to safety, economic, operational, and regulatory aspects of advanced light water reactors. This paper overviews the Requirements Document program as it relates to the US Advanced Light Water Reactor (ALWR) effort in reducing these uncertainties and reports the status of efforts to establish requirements for the for the next generation of pressurized water reactors. It concentrates on progress made in reducing the uncertainties which would deter selection of the nuclear option for contributing to US national electric power capacity needs in the next century and updates previous reports in the same area.

1309. OUTLINE OF ADVANCED BOILING WATER REACTOR. Y. MATSUO. Transactions of the American Nuclear Society (USA). CONF-870905--, Vol. 56, Sep 7-11 1987, pp. 112-118.

The ABWR design is based on construction and operational experience in Japan, USA, and Europe. The major objectives in developing the ABWR are: (1) enhanced plant operability, maneuverability and daily load-following capability; (2) increased plant safety and operating margins; (3) improved plant availability and capacity factor; (4) reduced occupational radiation exposure; (5) reduced rad-waste volume; and (6) reduced plant capital and operating costs. The ABWR Phase II preliminary design was completed in 1983. Phase III, which followed and was completed in late 1985, achieved new goals by optimization of many ABWR design features, systems, and construction practices. This confirmed the technical superiority cost the ABWR design and achieved significant cost reduction while retaining the overall safety and performance advantages.

1310. OUTLINE OF THE DEVELOPMENT OF THE ADVANCED PRESSURIZED WATER REACTOR (APWR). T. TANAKA. Transactions of the American Nuclear Society (USA). CONF-870905--, Vol. 56, Sep 7-11 1987, pp. 54-61.

The APWR development program was initiated as a part of the third phase of Improvement and Standardization Program for LWRs to improve the availability and economy for the future nuclear power plants in Japan. The major design objectives of the development program are as follows: (1) improved availability, (2) improved nuclear power economics, (3) improved operational performance, (4) improved plant safety, (5) reduction of radioactive waste and occupational radiation exposure, (6) efficient use of plant sites, and (7) conservation of resources. The major features of the APWR described emphasizes improvements in the nuclear steam supply system, fuel assembly, steam generator, fluid systems, and instrumentation and control systems, all of which have been modified significantly when compared with these of conventional PWRs.

1311. ADVANCES IN COMMERCIAL HEAVY WATER REACTOR POWER STATIONS. G.L. BROOKS. Transactions of the American Nuclear Society (USA). CONF-870905--, Vol. 56, Sep 7-11 1987, pp. 41-46.

Generating stations employing heavy water reactors have now firmly established an enviable record for reliable, economic electricity generation. Their designers recognize, however, that further improvements are both possible and necessary to ensure that this reactor type remains attractively competitive with alternative nuclear power systems and with fossil-fueled generation plants. This paper outlines planned development thrusts in a number of important areas, viz., capital cost reduction, advanced fuel cycles, safety, capacity factor,life extension, load following, operator aids, and personnel radiation exposure.

1312. EXPERIENCE IN PERSONNEL BETA DOSIMETRY IN AN ARGENTINE CANDU REACTOR (in English). C.A. SALAS International Radiation Protection Association, Washington, DC (USA) ; Australian Radiation Protection Society, Sydney. Proceedings of a Symposium - 7. international congress of the IRPA, Sydney, Australia, CONF-880404--, Apr 10-17 1988, pp. 440-443.

This paper describes the present difficulties existing through out the world for the execution of a correct Beta personnel dosimetry together with the method implemented at a Nuclear Power Station - (600 MWe CANDU REACTOR).

1313. EXPERIENCE OF DOSE REDUCTION PROCEDURE USING TARGET DOSE MANAGEMENT (in English). T. HASHIMOTO, M. NISHIKAWA, and Y. MITARAI International Radiation Protection Association, Washington, DC (USA) ; Australian Radiation Protection Society, Sydney. Proceedings of a Symposium - 7. international congress of the IRPA, Sydney, Australia, CONF-880404--, Apr 10-17 1988, pp. 1431-1434.

The exposure dose of plant workers in Japan showed a tendency toward increase year by year along with the increase in the number of operating units. However, the tendency has shown a slight decrease of late. The annual collective dose per reactor in 1986 was approximately 360 man rem at BWR and approximately 230 man rem at PWR. The dose could be limited to such a low level as a result of various dose reduction measures implemented in order at the existing plants as well as newly constructed plants. The efforts of concerned parties toward further reducing the dose and, at the same time, the establishment of a dose management and control system are recommended.

1314. CONTROL OF EMISSIONS FROM NUCLEAR POWER REACTORS IN CANADA (in English). D.J. GORMAN, B.C.J. NEIL, and R.M. CHATTERJEE International Radiation Protection Association, Washington, DC (USA) ; Australian Radiation Protection Society, Sydney. Proceedings of a Symposium - 7. international congress of the IRPA, Sydney, Australia, CONF-880404--, Apr 10-17 1988, pp. 1358-1361.

Nuclear power reactors in Canada are of the CANDU pressurised heavy water design. These are located in the provinces of Ontario, Quebec, and New Brunswick. Most of the nuclear generating capacity is in the province of Ontario which has 16 commissioned reactors with a total capacity of 11,500 MWe. There are four reactors under construction with an additional capacity of 3400 MWe. Nuclear power currently accounts for approximately 50% of the electrical power generation of Ontario. Regulation of the reactors is a Federal Government responsibility administered by the Atomic Energy Control Board (AECB) which licenses the reactors and sets occupational and public dose limits.

1315. QUALITATIVE AND QUANTITATIVE DECISION AIDING TECHNIQUES APPLICABLE IN RADIATION PROTECTION (in English). J.P. BERTHET. International Radiation Protection Association, Washington, DC (USA) ; Australian Radiation Protection Society, Sydney. Proceedings of a Symposium - 7. international congress of the IRPA, Sydney, Australia, CONF-880404--, Apr 10-17 1988, pp. 1294-1298.

EDF's exposure reduction program comprises a three pronged attack on the radiation field buildup process, based on modification of PWR's design features, and operating conditions.

1316. ALARA PRACTICES DURING NEUTRON SPECTRAL MEASUREMENTS INSIDE REACTOR CONTAINMENT (in English). K.L. SOLDAT and G.W.R. ENDRES International Radiation Protection Association, Washington, DC (USA) ; Australian Radiation Protection Society, Sydney. Proceedings of a Symposium - 7. international congress of the IRPA, Sydney, Australia, CONF-880404--, Apr 10-17 1988, pp. 64-67.

The accurate assessment of radiation dose to personnel who enter reactor containment is a difficult problem compounded by the presence of mixed radiation fields of betas, neutrons, and high-and low-energy photons. Many present dosimeters do not adequately assess the true dose from neutrons, betas, or high-energy photons. In 1980, the National Council on Radiation Protection and Measurements (NCRP) announced that it is considering lowering the maximum permissible dose for neutrons, perhaps by a factor of 3 to 10 less than existing limits. These changes could have serious consequences for the operation of present commercial nuclear power plants and the design of new plants. Present personnel dosimeters will not be adequate if these proposed changes are adopted; in fact, many dosimeters are not sufficiently accurate to be adequate with existing limits.

1317. REASONS WHY TVO HAS ONE OF THE LOWEST COLLECTIVE DOSES AMONG THE WORLD'S NUCLEAR POWER PLANTS (in English). R.O. SUNDELL International Radiation Protection Association, Washington, DC (USA) ; Australian Radiation Protection Society, Sydney. Proceedings of a Symposium - 7. international congress of the IRPA, Sydney, Australia, CONF-880404--, Apr 10-17 1988, pp. 585-588.

Industrial Power Company Ltd. (TVO) (.s and operates two boiling water reactor (BWR) units of Asea-Atom design in Olkiluoto, Finland. The installed net electric power of each unit is 710 MW. The full power operation of TVO I and TVO II began in 1979 and 1980, respectively. This paper discusses the main reasons why the annual personal doses have stayed on a very low level at the TVO power plants. The variation in annual collective doses falls between 0.3 and 0.8 Sv, and the highest annual effective dose equivalent was 16.75 mSv. The main reasons are listed below: 1. Plant design: the highly radioactive systems are separated from the low ones. The capacity of the reactor-water cleanup system is relatively high. The main components are made of materials with low cobalt concentration. 2. The radiological work permit system is computer based. The most important information about different components, systems and rooms, like dose rates and contamination levels, is stored in the memory. Radiological work permits, which eve been carried out earlier, are also stored in the memory. Good prior knowledge of characteristics in the work objects makes careful work planning possible. 3. Work dosimetry system: to be able to limit doses, a good knowledge of the work activities that contribute todoses is needed. TVO has a very up-to-date microprocessor-based work dosimetry system equipped with seven readers and 350 dosimeters. The only way to perform satisfactory as-low-as-reasonably-achievable calculations is to know the exact work doses. 4. Outage planning: outages cause 80% of all annual doses, and that is why special emphasis must be placed on advance planning. With careful planning, the annual outage time has been shortened to 2 to 3 weeks.

1318. DOSE REDUCTION AND CONTROL AT THE WINFRITH REACTOR (in English). B.G. CHAPMAN and T.E. BLACKMAN. International Radiation Protection Association, Washington, DC (USA) ; Australian Radiation Protection Society, Sydney. Proceedings of a Symposium - 7. international congress of the IRPA, Sydney, Australia, CONF-880404--, Apr 10-17 1988, pp. 573-576.

The Winfrith Reactor is a 100 MW(e) heavy water moderated, light water cooled, pressure tube reactor. It was designed and built in the 1960s as a prototype, using the best technology available at the time. Coolant chemistry problems initially caused fuel failures and due to material specification, significant quantities of activated corrosion products and minor quantities of fission products are transported into working areas. For several years the reactor has been the focus of a major dose reduction programme with the emphasis on reducing individual doses. The options available in a dose reduction programme are straightforward. Remove or reduce the source and then apply the usual distance, shielding and time. Source reduction depends upon the replacement of dose contributing alloys (eg Cobalt-rich stellite) where practicable, the refinement of chemical control of the coolant to minimise pick-up, transport and plate-out of the active species together with chemical cleaning of the circuit before major maintenance work starts in the primary containment. Gains can also be made by arranging rapid removal of active waste.

1319. RADIOLOGICAL PROTECTION IN THE CEGB - THE CHANGING SCENE (in English). R. B. PEPPER International Radiation Protection Association, Washington, DC (USA) ; Australian Radiation Protection Society, Sydney. Proceedings of a Symposium - 7. international congress of the IRPA, Sydney, Australia, CONF-880404--, Apr 10-17 1988, pp. 935-938.

This paper briefly outlines changes which are being consolidated or introduced into radiological protection in the CEGB nuclear establishments.

1320. DECOMMISSIONING NUCLEAR POWER STATIONS (in English). P.B. WOOLLAM. CEGB (Central Electricity Generating Board) Research (UK), Sep 1988, pp. 32-41.

The main objective of research at Berkeley Nuclear Laboratories into the decommissioning of nuclear power stations is to assess the safety implications of the various strategies which are being considered worldwide for the eventual dismantling of these stations. Although the BNL studies have primarily addressed the oldest of the CEGB's plant, the steel pressure vessel Magnox reactors, assessments of PWR decommissioning have also been undertaken to allow design changes to be introduced which will reduce eventual dismantling problems and minimise the production of radioactive waste.

1321. USING ICE TO BLAST OFF CRUD [DECONTAMINATION] (in English). TAKAAKI OGUCHI. Nuclear Engineering International (UK), Vol. 34, Jan 1989, pp. 49-50.

In Japan ice-blasting is being used to decontaminate irradiated material. Blasting with ice particles was developed in order to eliminate the problems associated with conventional mechanical methods. It has been applied to the decontamination of pump casings, valves, and primary circuit piping of BWR type reactors.

1322. FACTORS RELEVANT TO THE RECYCLING OR REUSE OF COMPONENTS ARISING FROM THE DECOMMISSIONING AND REFURBISHMENT OF NUCLEAR FACILITIES (in English). International Atomic Energy Agency, Vienna (Austria), 1988, 75 pp.

The decommissioning and decontamination of nuclear facilities is a topic of great interest to many Member States of the International Atomic Energy Agency (IAEA) because of the large number of older nuclear facilities which are or soon will be retired from service. To assist in the development of the required decommissioning expertise, the IAEA is developing reports and recommendations which will eventually form an integrated information base covering in a systematic way the wide range of topics associated with decommissioning. This information is required so that Member States can decommission their nuclear facilities in a safe, timely and cost effective manner and the IAEA can effectively respond to requests for assistance One area which warrants more detailed analyses is an assessment of the factors important to the recycling or reuse of components arising from the refurbishment or decommissioning of nuclear plants, the topic of the present report. The document provides an up to date review of the engineering, social, scientific and administrative factors relevant to the safe recycling or reuse of components arising from decommissioning or refurbishment of nuclear facilities. This report should be of interest to owners, operators, policy makers and regulators involvuclear facilities, especially those in developing countries.

1323. EXCHANGE OF DOSE DATA WITHIN NUCLEAR ACTIVITIES IN FINLAND AND SWEDEN (in English). O. VILKAMO and L. MALMQVIST. International Radiation Protection Association, Washington, DC (USA) ; Australian Radiation Protection Society, Sydney. Proceedings of a Symposium - 7. international congress of the IRPA, Sydney, Australia, CONF-880404--, Apr 10-17 1988, pp. 1427-1430.

In the Nordic countries, Denmark, Finland, Iceland, Norway and Sweden, only Sweden and Finland have introduced nuclear power into energy production. The first still operating nuclear power plant was commissioned in Sweden in 1972 and in Finland in 1977. It was soon noticed that there was a growing tendency that small groups of workers used to move at short notice between Finland and Sweden to work in the nuclear power plants in both countries during maintenance periods. In 1983, the regulatory authorities for radiation protection, National Institute of Radiation Protection in Sweden and Finnish Centre for Radiation and Nuclear Safety in Finland, surveyed the radiation exposure to those workers. The authorities have brought about an arrangement by means of which the central dose data bases in the other country since 1984 have been able to record without delay the radiation doses received by her own citizens in the nuclear power plants of the neighbouring country. In addition, the authorities have confirmed the procedures of controlling dose data on workers from the neighbouring country, before those workers start working in a nuclear power plant regulated by the national authorities in question. The paper describes the starting point of the activity, the established practice and the experience achieved. Until now, the practical experiences are positive. The total radiation exposure to the workers in the Swedish and Finnish nuclear power plants has been relatively low at each plant site. Thus, the main objective in the exchange of dose data, is to achieve a good radiation protection control.

1324. OCCUPATIONAL EXPOSURES AND PRACTICES IN NUCLEAR POWER PLANTS (in English). J.W. BAUM. Brookhaven National Lab., Upton, NY (USA). Proceedings of the 25th annual meeting of the National Council on Radiation Protection and Measurements. Radiation Protection Today - The NCRP at Sixty Years, Washington, D C., Apr 5-6 1989, pp. 101-123.

Exposures in U.S. nuclear power plants and U.S. naval ships and shipyards are reviewed. This is followed by a review of the evalution of the ALARA concept and its application in nuclear power plants. It is concluded that additional developments and implementation of the quantitative aspects of the

ALARA process are needed, and that plant and national goals can have importan: impacts or collective doses.

1325. IMPLEMENTING OPTIMIZATION PRINCIPLE IN THE DESIGN OF NUCLEAR POWER PLANTS: REGULATORY TRENDS IN ITALY (in English). S. BENASSAI. 4. European congress and 13., Salzburg, Austria. Oesterreichischer Verband fuer Strahlenschutz (OeVS), Vienna (Austria). CONF-860969--, Report No. OEVS-Mitteilung--1988, Nov 1988, pp. 28-32.

As requested by the 1981 National Energy Plan, ENEA/DISP, the Italian regulatory authority, has been deeply involved in setting up of general design criteria for the future nuclear power plants (NPP). The purpose of the criteria is to guarantee the absence of undue risk for the workers and general public arising from exposure to ionizing radiation, and in this framework the application of the optimization principle plays a relevant role.

1326. DOSE ASSESSMENT AND MEDICAL CONSEQUENCES OF RADIATION EXPOSURE OF EMPLOYEES OF ELECTRICITE DE FRANCE (in English). M. BERTIN, J. LALLEMAND, and J.P. BERTHET. 4. European congress and 13. regional congress of the International Radiation Protection Association. Oesterreichischer Verband fuer Strahlenschutz (OeVS), Vienna (Austria). CONF-860969--, Report No. OEVS-Mitteilung--1988, Nov 1988, pp. 440-443.

It is pointed out that radiation protection is both for physicians and engineers, and that information gained from personnel dosimetry should be used to modify the design of equipment, to eliminate or reduce causes of radiation exposure. The field of experience in question are nuclear power plants operated by Electricite de France.

1327. 4. EUROPEAN CONGRESS AND 13. REGIONAL CONGRESS OF IRPA. 20 YEARS EXPERIENCE IN RADIATION PROTECTION - A REVIEW AND OUTLOOK (in German, English). E. TSCHIRF and A. HEFNER. Oesterreichischer Verband fuer Strahlenschutz (OeVS), Vienna (Austria). 4. European congress and 13. regional congress of the International radiation Protection Association. CONF-860969--,Report No. OEVS-Mitteilung--1988, Nov [illegible], 909 pp.

153 papers on various aspects of radiation protection (including historical) were presented, 146 of them are in INIS scope. The content is indicated in the session headings: radiation exposure of the public; radioecology,radioactive waste; occupational radiation exposure; regulatory, legal and social aspects of radiation protection; risk assessment and radiation effects; instrumentation; external and internal dosimetry; non-ionizing radiation; Chernobyl session.

1328. RECURRENT ANALYSES OF RADIATION PROTECTION CONDITIONS AT THE SWEDISH NUCLEAR POWER PLANTS (in English). S. HENNIGOR. 4. European congress and 13. regional congress of IRPA. 20 years experience in radiation protection - a review and outlook. Oesterreichischer Verband fuer Strahlenschutz (OeVS), Vienna (Austria). CONF-860969--,Report No. OEVS-Mitteilung--1988, Nov 1988, pp. 396-398.

A complete safety analysis is performed every 10th year for each nuclear power reactor in Sweden. NIRP review the radiation protection conditions. Analyses of different areas gives the status of the radiation protection activities at the reactor unit and points out which fields will require most attention in the nearest future.

1329. COBALT-60 CONTROL IN ONTARIO HYDRO REACTORS (in English). C.S. LACY. Proceedings of a Symposium - 3. international symposium on environmental degradation of materials in nuclear power systems: water reactors, Traverse City, MI, US. CONF-870839--, Aug 30 - Sep 3 1987, pp. 749-754.

This paper discusses the impact of specifying reduced Cobalt-59 in the primary heat transport circuit materials of construction on the radiation fields developed around the primary circuit. An eight-fold reduction in steam generator radiation fields due to Cobalt-60 has been observed for two identical sets of reactors, one with and one without Cobalt-59 control. The comparison is between eight reactors at the Pickering Nuclear Generating Station (PNGS). Units 5 to 8 (PNGS-B) are identical to Units 1 to 4 (PNGS-A) except that PNGS-B has reduced impurity Cobalt-59 in the alloys of construction and a reduced use of stellite. The effects of chemistry control are also discussed.

1330. BERKELEY NUCLEAR POWER STATION: THE FINDINGS OF NII'S ASSESSMENT OF THE CEGB'S LONG TERM SAFETY REVIEW (in English). Health and Safety Executive, London (UK). Nuclear Installations Inspectorate. Her Majesty's Stationery Office (London, UK), 1988, pp. 16.

A report is presented on the assessment of the safety of Berkeley Nuclear Power Station, carried out by HM Nuclear Installations Inspectorate (NII). The assessment was undertaken because the licencees of Berkeley Nuclear Power Station had indicated their will to operate the station beyond its 20 year expected lifetime. The report contents contains a summary of NII's findings on the following topics:-reactor pressure circuit integrity, effects of ageing and in-service wear, application of modern criteria, role of the operator, radiation doses, and natural and other hazards.

1331. REGULATORY PROCEDURES FOR THE DECOMMISSIONING OF NUCLEAR INSTALLATIONS (in English). P.B. WOODS and P.K. BASU IBC Technical Services Ltd., London (UK). Proceedings of a Symposium - International seminar on the decommissioning of nuclear facilities, London, GB, CONF-8807165--, Jul 6-7 1988.

The basic safety legislation under which operational safety at nuclear installations is regulated does not change when the plant is decommissioned. In the United Kingdom the relevant nuclear safety legislation is embodied in several Acts of Parliament or international conventions. These are listed and described. The potential risk in decommissioning is from radiation exposure of the workers and to a lesser extent of the public and environment. The regulations try to ensure this risk is reduced to acceptable levels. This objective can be achieved if the project is adequately planned, there is reliable information about the plant, the risks are identified and assessed, the quality assurance is good and personnel are trained, and the radioactive wastes produced are managed and disposed of suitably.

1332. ACTIVITIES WITHIN THE EUROPEAN COMMUNITY RELATED TO SAFETY GOALS FOR NUCLEAR POWER PLANTS (in English). S.A. HARBISON, G.N. KELLY, E.V. GILBY, F. LANGE, P.A. GOTTSCHALK, and B.J. TOLLEY. International Atomic Energy Agency, Vienna (Austria). Proceedings of a Symposium - International conference on nuclear power performance and safety, Vienna, AT,. CONF-8709263--, Sep 28 - Oct 2 1987, v. 4.

In February 1982, after consultation with its Safety of Light Water Reactors Working Group No. 1 "Methodology, Criteria, Codes and Standards", the Commission of the European Communities appointed a special Task Force to review existing safety objectives for nuclear power plants and to report on the degree of coherence between the various approaches. The Group considered all relevant safety objectives which existed within Community Member countries as well as the Safety Goals proposed by the United States Nuclear Regulatory Commission and presented its general conclusions and perspectives in its first Status Report in May 1983. The paper gives a brief overview of the more important observations in the first Status Report and reports on progress made by the Task Force since then. In particular it summarizes the preliminary results of a benchmark exercise organised by the Group to elucidate the degree of coherence between the dose-frequency targets used in the different Member States. Eight organizations from seven countries participated in the exercise, which was organized in a form that was as unambiguous as possible to facilitate the investigation of differences in the dose predictions of the various participants. The exercise was also structured in such a way that comparisonscould readily be made at intermediate stages in the calculation. The comparison of the various results is currently at a provisional stage but a number of significant differences have already emerged for particular releases in particular release conditions. These differences mainly result from the different assumptions used by the various participants in relation to the conditions (e.g. distance, meteorological conditions, etc.) under which the dose is calculated. The paper also discusses the role of individual dose-frequency targets in relation to other targets.

1333. PROBABILISTIC SAFETY CRITERIA FOR NUCLEAR POWER PLANTS (in English). B. EDMONDSON and F. NIEHAUS International Atomic Energy Agency, Vienna (Austria). Proceedings of a Symposium - International conference on nuclear power performance and safety, Vienna, AT. CONF-8709263--, Sep 28 - Oct 2 1987, v. 4.

With the increasing use of probabilistic safety assessment many countries have developed probabilistic criteria for nuclear safety to complement deterministic criteria. However their status and use vary. One group of countries is already using them (or is close to a final decision to do so) for safety decisions including design, licensing and operation. Another group of countries is at present debating the basic concepts and their implications. The third group of countries is monitoring the developments for possible use. Based on qualitative objectives a number of quantitative criteria have been proposed. There is a wide range of attitudes. For example, there is a contrast between the reactor safety practitioners' views and those expressed by the International Commission on Radiological Protection (ICRP) Publication 46 and practitioners in the radwaste field. The latter impose individual risk limits, limits in the sense that they must not be exceeded. This position was arrived at via the established risk levels for individual continuous exposure at the ICRP limits. They also take the view that other objectives should be subsumed in an optimization process which takes account of all relevant factors, including costs, societal risk, amenity detriment, etc. Reactor practitionernot accept the concept of individual risk limits. Rather, they prefer to see probabilistic safety criteria as design guidelines, objectives, assessment principles. Within the community of reactor safety practitioners itself, there is a variety of attitudes to such criteria depending on local circumstances. In some countries strenuous efforts are being made to develop the concept to the point where it can ease the burden of rule making requirements. In other countries it is thought that the value of the concept is limited by the likelihood that the burden to demonstrate compliance will be very severe except at the safety function/system level.

1334. NRC SAFETY RESEARCH IN SUPPORT OF REGULATION, 1988. Nuclear Regulatory Commission, Washington, DC (USA). Office of Nuclear Regulatory Research. NUREG--1266-Vol.3, May 1989.

This report, the fourth in a series of annual reports, was prepared in response to Congressional inquiries concerning how nuclear regulatory research is used. It summarizes the accomplishments of the Office of Nuclear Regulatory Research during 1988. The goal of this office is to ensure that safety-related research provides the technical bases for rulemaking and for related decisions in support of NRC licensing and inspection activities. This research is necessary to make certain that the regulations that are imposed on licensees provide an adequate margin of safety so as to protect the health and safety of the public. This report describes both the direct contributions to scientific and technical knowledge with regard to nuclear safety and their regulatory applications.

1335. IMPROVED CRUD IRON REMOVAL EFFICIENCY FOR POWDER RESIN TYPE CONDENSATE FILTERS (in Japanese). HIROSHI NAGAI and TAKAO INO. Ehara Jiho (Japan), Jan 1989, pp. 58-62.

In 1984, a precoat type condensate filtration system was delivered to The Tokyo Electric Power Co., Inc. by Ebara and stable operation of the system is reported ever since. Originally, condensate filtration systems are used to remove crud iron in condensate water. However, it has become desirable to freely control the crud iron in the outlet flow of such filtration system. The main source of radioactivity in a BWR plant, is Cobalt 60, and it is necessary to optimally control the amount of crud iron released into the reactor to match the nickel and cobalt amounts in the reactor feed water for achieving an overall reduction of the concentration of radioactivity within the BWR plant. The method of such control, developed by the authors, is outlined in the following. By this method, the radioactive level within the overall plant is significantly decreased. Consequently, the risk of radioactive exposure of personnel at time of periodical checkup is greatly reduced.

1336. ANALYTICAL TECHNIQUES FOR THE ASSESSMENT OF ENVIRONMENTAL EXPOSURE RESULTING FROM HYDROGEN INJECTION CHEMISTRY. M.R. BIERMAN, J.E. CLINE, and G.C. RE. Spectrum '86: Proceedings: Volume 2. American Nuclear Society (USA). Fuel Cycle and Waste Management Div. ; American Nuclear Society (USA). Niagara-Finger Lakes Section. CONF-860905--, Vol.2, Jul 1987, pp. 2005-2016.

It has been observed in boiling water reactors (BWR) that intergranular stress corrosion cracking (IGSCC) of primary steam system pipes and components can occur under certain environmental parameters. One parameter, the oxidizing environment that exists, can be reduced by lowering oxygen concentrations via hydrogen injection. This decreases the potential of IGSCC and the production of corrosion products that activate and result in higher levels of radwaste. During normal BWR operations ^{16}N formed from ^{16}O combines in water-soluble nitrates and nitrites.

However, the reduction of oxidizing potential created by hydrogen injection increases the amount ^{16}N combined into ammonia (NH_3). The increase is transported through the steam system, resulting in increased radiation levels in and around the plant site. Field measurements taken during a hydrogen water chemistry (HWC) mini-test at James A. Fitzpatrick Nuclear Power Station permitted the assessment of environmental exposure rate resulting from the increased hydrogen concentration in the feedwater. Adequate data existed for the development of a methodology for estimating exposure rates at and around the plate site based on the hydrogen concentration and main steam-line

1337. COMPREHENSIVE RADIOLOGICAL PROTECTION PROGRAM - AN ESSENTIAL ELEMENT OF SUCCESSFUL DECOMMISSIONING. R.P. NAIR, M.A. MAAN, and S. SCHAFER. Spectrum '86: Proceedings: Volume 2. American Nuclear Society (USA). Fuel Cycle and Waste Management Div.; American Nuclear Society (USA), Niagara-Finger Lakes Section. CONF-860905--Vol.2, Jul 1987, pp. 1378-1389.

The decommissioning of the 250 MWe Gentilly-1 nuclear plant over a two-year period called for a clearly defined radiation protection program. The decommissioning activities were carried out within the framework of radiation protection guidelines and techniques so structured that all the elements of the program could be applied for the future handling of more radioactive facilities.

1338. COMMISSION OF THE EUROPEAN COMMUNITIES' PROGRAM ON DECOMMISSIONING OF NUCLEAR FACILITIES. K.H. SCHALLER and B. HUBER. Spectrum '86: Proceedings: Volume 2. American Nuclear Society (USA). Fuel Cycle and Waste Management Div.; American Nuclear Society (USA). Niagara-Finger Lakes Section. CONF-860905--Vol.2, Jul 1987, pp. 1174-1183.

The Commission's cost-sharing research program on decommissioning of nuclear facilities has supported development of a number of techniques and processes applicable in decommissioning activities, aimed at providing efficient procedures taking into account radiation protection, minimization of waste arisings and cost-effectiveness. After a short description of the program, the main results of the already performed contractual work are summarized. Special attention is given to the remote handling aspect and to management of large quantities of very low-level activity waste. In the currently running program, a large part of the effort is devoted to the testing of new techniques under real conditions in the frame of large-scale decommissioning operations.

1339. REVIEW OF THE UNITED KINGDOM PWR PRIMARY-CIRCUIT-CHEMISTRY PROGRAM: PROGRESS REPORT NO. 1. T. SWAN. Electric Power Research Inst., Palo Alto, CA (USA) ; Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs. Report No. EPRI-NP--6368, May 1989, 44 pp.

The United Kingdom is building its first PWR station at Sizewell 'B' which is due to start generation of power during 1994. This is planned to be the first of a number of PWRs. The construction program is being supported by an R and D program which, inter alia, addresses the accumulation of radioactive deposits in the primary circuit. The present report reviews recent developments in the United Kingdom man-rem control and chemistry program which is funded by the Central Electricity Generating Board and the UK Department of Energy. The aim is to highlight the principal findings from recent work and to set them in the context of world wide developments in this area. The object is to make EPRI members aware of the studies being conducted in the United Kingdom and also to give a view of the forward program for 1989.

1340. QUALIFICATION OF COBALT FREE HARDFACING ALLOYS FOR PWR VALVE APPLICATIONS. G.P. AIREY. Proceedings of a Symposium - 3. international symposium on environmental degradation of materials in nuclear power systems, Traverse City, MI, US. CONF-870839--, Aug 30 1987, pp. 755-762.

The presence of cobalt in the primary circuit of pressurized water reactors (PWR) is a major contributor to the primary circuit activity level. There are two principal cobalt sources in primary circuit materials; as an impurity in components such as steam generator tubing (Inconel 690, Inconel 600, Incoloy 800), pipework (austenitic stainless steel) or reactor internals and as the principal alloying element in hardfacing materials (e.g. Stellite). There have been many assessments of the relative contributions of these sources to the cobalt level in the primary circuit. The lack of agreement in these assessments has justified attempts to reduce the cobalt input into the primary circuit from all sources. For new plants such as

Sizewell--B in the UK lower cobalt contents have been specified for the Inconel 690 steam generator tubing and stainless steels exposed to the primary circuit. The two major uses of Stellite in the primary circuit are in control rod drive mechanisms and in valves. In the UK, efforts to reduce the usage of Stellite have focussed on valve applications, and two approaches have been considered. For certain applications, in particular flow control valves, hardfacings may not be necessary, and wrought stainless steels can be specified. For other applications cobalt-free alternate alloys need to be qualified. It is recognized that it is unlikely that cobalt-free alternate alloys will be qualified in time for use in Sizewell--B but would be for use in any future UK PWR.

1341. MATERIALS QUALIFICATION TESTING IN SUPPORT OF CHEMICAL DECONTAMINATION AT OYSTER CREEK. D.W. COVILL, F.S. GIACOBBE, and G.E. VONNIEDA. Proceedings of a Symposium - 3. international symposium on environmental degradation of materials in nuclear power systems: water reactors, Traverse City, MI, US. CONF-870839--, Aug 30 1987, pp. 763-770.

Chemical decontamination is an effective means of reducing personnel exposure to radiation at a nuclear power plant. But a utility must be assured that the processes used do not jeopardize the integrity of the system materials. GPU Nuclear planned to perform a chemical decontamination of the Oyster Creek Recirculation system during a major refueling, maintenance, and inspection outage. To qualify a commercially available decontamination process, the authors performed an evaluation to determine the effects of three processes on the materials' resistance to general corrosion, intergranular attack, pitting, and intergranular stress corrosion cracking. The authors performed an extensive literature search; then established a test program to supplement the existing data. Testing was performed on samples removed from the Oyster Creek piping system. The corrosion layer was analyzed using X-ray diffraction to identify any harmful chemical species such as chlorides, fluorides, and sulfur compounds. Corrosion coupons and CERT specimens were then exposed to the decontamination processes and examined. CERT specimens were tested to fracture in a simulated BWR water environment.

1342. INFLUENCE OF RADIOLYSIS PRODUCTS AND IMPURITIES ON THE CRITICAL POTENTIAL FOR IGSCC OF TYPE 304 STAINLESS STEEL IN WATER AT 250 °C. A. MOLANDER and B. ROSBORG. Proceedings of a Symposium - 3. international symposium on environmental degradation of materials in nuclear power systems: water reactors, Traverse City, MI, US. CONF-870839--, Aug 30 1987, pp. 333-340.

This paper applies to intergranular stress corrosion cracking (IGSCC) of Type 304 stainless steel piping in boiling water reactors and the so-called hydrogen water chemistry as a remedy to avoid IGSCC. IGSCC of sensitized austenitic stainless steel in oxygenated high-purity water can be avoided by keeping the corrosion potential below a certain electrode potential, that is a critical potential for IGSCC. The authors previously presented results of measurements of the critical potential for a heavily sensitized Type 304 stainless steel in a high purity water - oxygen environment. Data on the influence of other radiolysis products and a few detrimental impurities on the critical potential for IGSCC of the very same steel is presented in this paper.

1343. MORTALITY AND CAREER RADIATION DOSES FOR WORKERS AT A COMMERCIAL NUCLEAR POWER PLANT: FEASIBILITY STUDY. R. GOLDSMITH, J.D. JR. BOICE, Z. HRUBEC, P.E. HURWITZ, T.E. GOFF, and J. WILSON. Health Physics (USA), Vol. 56, Feb 1989, pp. 139-150.

Career radiation doses for 8,961 male workers at the Calvert Cliffs Nuclear Power Plant (CCNPP) were determined for both utility (n = 4,960) and contractor (n = 4,001) employees. Workers were followed from the time of first employment at CCNPP (including plant construction) to the end of 1984 (mean follow-up = 5.4 y). Plant operation began in 1975. The mean duration of employment was 1.9 y at CCNPP and 3.1 y in the nuclear industry. Career radiation doses were determined from dosimetry records kept by the utility company and the U.S. Nuclear Regulatory Commission (NRC). For all exposed workers, the average career dose was 21 mSv and was higher for contractor (30 mSv) than utility (13 mSv) workers. Career doses were also higher among those employed in the nuclear industry for greater than or equal to 15 y (111 mSv) and among workers classified as health physicists (56 mSv). Cumulative doses of greater than or equal to 50 mSv were received by 12% of the workers; the maximum career dose reported was 470 mSv. The availability of social security numbers for practically all employees facilitated record-linkage methods to determine mortality; 161 deaths were identified. On average the workers experienced mortality from all causes that was 15% less than that of the general population othe U.S., probably due to healthier members of the population being selected for employment. Our investigation demonstrates that historical information is available from which career doses could be constructed and that, in principle, it is feasible to conduct epidemiologic studies of nuclear power plant workers in the U.S. Although difficult, the approach taken could prove useful until such time as a comprehensive registry of U.S. radiation workers is established.

1344. REASONS WHY TVO HAS ONE OF THE LOWEST COLLECTIVE DOSES AMONG THE WORLD'S NUCLEAR POWER PLANTS. R. SUNDELL. Transactions of the American Nuclear Society (USA). CONF-870837--, Vol. 54, Aug 30 1987, p. 95.

Industrial Power Company Ltd. (TVO) owns and operates two boiling water reactor (BWR) units of Asea-Atom design in Olkiluoto, Finland. The installed net electric power of each unit is 710 MW. The full power operation of TVO I and TVO II began in 1979 and 1980, respectively. This paper discusses the main reasons why the annual personal doses have stayed on a very low level at the TVO power plants. The variation in annual collective doses falls between 0.3 and 0.8 Sv, and the highest annual effective dose equivalent was 16.75 mSv. The main reasons are a compartmentalized plant design, a computer based radiological work permit system and work desimitry system, and careful outage planning.

1345. DEVELOPING POSITIVE WORKER ATTITUDES TOWARD RADIATION PROTECTION. N.L. MILLIS. Transactions of the American Nuclear Society (USA). CONF-870837--, Vol. 54, Aug 30 1987, p. 95.

Teamwork, productivity, and reducing exposure are admirable goals presented to the workers in a nuclear power plant. A common thread to achievement in these areas resides in worker attitudes toward the tasks presented. A positive, alert, and cooperative attitude is an element in a worker's mind that must be created and maintained by good leadership and management practices. At the Calvert Cliffs Nuclear Power Plant, management has used certain strategies to foster good positive worker attitudes toward radiation protection and quality workmanship in all tasks. Strategies differ from management by objectives in that they have no deadlines or timetables in and of themselves. Rather, strategies are preplanned methods that can be called upon when the opportunity arises to improve worker attitudes. A series of five strategies for positive attitude development are described in the full paper. The strategies are identified with buzz words to allow the user a recall mechanism (as with the acronyms abounding in the nuclear industry). They cover the range of management techniques from example setting to reward/recognition. Although not unique to radiation exposure management, nor all inclusive, the strategies provide some though stimulation in creating productive worker attitudes.

1346. A SELF-ASSESSMENT PROGRAM FOR COMMON WEALTH EDISON COMPANY. D. GARNER, F. KROWZAK, and R. WISHAU. Transactions of the American Nuclear Society (USA), Vol. 54, Aug 30 1987, p. 102.

This paper explains the assessment program developed by Commonwealth Edison Company (CECo) to evaluate the effectiveness of technical activities at its nuclear generating stations. The technical activities evaluated are in the areas of chemistry, emergency planning, radiation protection, and radioactive waste operations. The stations have technical programs for each of these areas and are set up to comply with applicable governmental laws and regulations, industry guidelines, and corporate policies and procedures. Reviews and audits need to be performed to ensure that technical program objectives are being met and that the programs are progressing as intended. Many organizations, including the Institute of Nuclear Power Operations, U.S. Nuclear Regulatory Commission, American Nuclear Insurers, and the CECo Quality Assurance Department perform such reviews and audits, which generate a wide variety of information on the stations' activities.

1347. DOSE REDUCTION THROUGH AN ALARA PROGRAM AT ALMARAZ NPP (in Almaraz). A. LEAL, D. SUSTACHA, and J.M. ANEIROS. Transactions of the American Nuclear Society (USA). CONF-870837--, Vol. 54, Aug 30 1987, p. 96.

Radiation exposure is in keeping with the rate at which nuclear power production is increased. Therefore, it becomes more and more important that nuclear power producing plants develop an effective dose optimization and minimization [as-low-as-rasonably-achievable (ALARA)] program. Although radiation exposure suffered by the workers is carefully kept below administrative limits, there is a moral obligation to keep these exposures as low as possible. This requirement becomes apparent in the ALARA principle, supported and accepted by all countries with nuclear power plants in operation. Empresarios Agrupados (a Spanish architect engineer company) collaborates with nuclear power producing plants in an effort to maintain the collective ALARA doses through the efforts of a group of engineers specializing in dose minimization and optimization techniques. This group is organized as a radiation protection and maintenance team (ALARA team).

1348. CANDU HUMAN PERFORMANCE ANALYSIS (in CANDU). I. WALKER. Transactions of the American Nuclear Society (USA). CONF-870837--, Vol. 54, Aug 30 1987, p. 178.

An evaluation of human performance is presented in this paper in the context of the operational safety management system. To focus on problems, an experience review program has been developed to establish trends, demonstrate the degree of compliance with standards, and determine the causes of poor performance. The primary method by which the experience review takes place is significant event reporting (SER). A significant event is an incident that causes an undesirable effect on safety, product quality, environmental protection, or product cost. In spite of advanced technology and the degree of automation of the Canada deuterium uranium (CANDU) design, mistakes and malfunctions to occur. Considerable effort has been made to prevent or reduce the incidence of error. The Institute of Nuclear Power Operations developed a system to analyze human error, called the Human Performance Evaluation System (HPES). To encourage an open exchange of information, the system is anonymous and nonpunitive. All data gathered during HPES evaluations are kept confidential.

1349. LESSONS LEARNED IN PLANNING ALARA/HEALTH PHYSICS SUPPORT FOR MAJOR NUCLEAR POWER PLANT OUTAGES. T.R. GILMAN and M.L. LESINSKI. Transactions of the American Nuclear Society (USA). CONF-870837--, Vol. 54, Aug 30 1987, p. 95.

Although as low as reasonably achievable (ALARA)/health physics is viewed as necessary support for nuclear power plant outage work, it can be the last area to which attention is given in preparing for a large-scope outage. Inadequate lead times cause last-minute preparations resulting in delays in planned work. The Dresden Unit 3 Recirculation Piping Replacement Project is examined from a planning viewpoint. The attention that was given the various areas of a comprehensive ALARA/health physics program is examined, and approximate recommended lead times are discussed. The discussion will follow a chronological path from project inception to the beginning stages of outage work. Initially, the scope of work needs to be assessed by individuals familiar with similar projects of equivalent magnitude. Those individuals need to be health physics professionals who understand the particular utility and/or the site's way of doing business. They should also possess a good understanding of preferred industry practices.

1350. TRAINING OF THE PWR STAFF AT EDF. M. JUSSELIN. Transactions of the American Nuclear Society (USA). CONF-870837--, Vol. 54, Aug 30 1987, p. 143.

Electricite de France (EdF) is now operating thirty-two 900-MW and nine 1300-MW pressurized water reactor (PWR) units and a 1500-MW sur generator; two 900-MW and ten 1300-MW PWR units are under construction. The number of persons employed to run these plants is at present 24000. A very important training program provided initial knowledge, specific training, and retraining in order to ensure nuclear safety. The following list includes those features required in the most viable and efficient training program: first contacts for familiarization with company and safety training (including radiation protection in nuclear plants); industrial adaptation, depending on the specialty; training for the function performed; skill maintenance and refresher courses in the function; and preparatory or promotion courses when performing a new function.

1351. PUTTING MICRODOSIMETRY TO WORK: THE MEASUREMENT CAMPAIGNS OF 1987 AT KKG, KKL AND PSI. C. BARTH and C. WERNLI. Radiation Protection Dosimetry (UK), CONF-871020--, Vol. 23, Oct 12 1987, pp. 257-260.

Measurements have been made at the Goesgen PWR (KKG) and the Leibstadt BWR (KKL) nuclear power stations, as well as at the PSI 600 MeV proton accelerator using a tissue-equivalent low pressure proportional counter based on the KFA-Juelich design. Modifications to the Uelich design consisted on an outer shell of 0.8 mm Al and 8 mm Plexiglass instead of the original 20 mm PE, as well as the omission of the internal α source. The dose equivalent fraction arising from neutrons was found to be below

20% at both KKG and KKL under normal operating conditions. The neutron spectra were also measured by Birch and Delafield, as well as by Vylet and Valley, using proton recoil counters and Bonner spheres. The spectra measured at PSI differ fundamentally from reactor spectra mostly due to very penetrating neutrons in the 'window' (around 100 MeV) of concrete shielding, with a very low γ component which is presumably of neutron capture origin.

1352. STUDY OF HARD MATERIALS FOR COATING PWR COMPONENTS (in French). M. AUBERT, A. BOUGAULT, D. BRENET, M. GUTTMANN, and C. BENHAMOU. CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Dept. de Technologie. Report No. CEA-CONF--9703, CONF-8811180--, Nov 1988, 9 pp.

Cobalt free alloys colmonoys, ceniums, nitronics and alloy Ni50Mo35Cr13 are compared to stellite 6 for vane parts in nuclear power plants. No material is as good as stellite, nitronic 60 is fairly good but hardness should be increased possibly by carbide dispersion.

1353. DECOMMISSIONING IN BRITISH NUCLEAR FUELS PLC. A. COLQUHOUN IBC Technical Services Ltd., London (UK). Proceedings of a Symposium - International seminar on the decommissioning of nuclear facilities, London, GB, CONF-8807165--, July 6, 1988.

Decommissioning projects at the BNFL Sellafield site have been selected taking the following into account; the need to gain experience in preparation for the decommissioning of the Magnox reactors and for the post Magnox stage; the need to develop larger scale projects; the need to be cost effective and to foster long term safety. The balance between prompt or delayed decommissioning has to consider operator dose uptake and radioactive waste management. The ten year plan for decommissioning at Sellafield is described briefly. Currently decommissioning is of the fuel pond and decanning plant, the Windscale Pile Chimneys, the coprecipitation plant and the uranium recovery plant.

1354. NUCLEAR ENERGY AND RADIATION HYGIENE (in Dutch). J.C. ABRAHAMSE. NVS-Nieuws (Netherlands). Dutch Society for Radiation Dosimetry scientific conference, CONF-8803206--, Vol. 13, Mar 4 1988, pp. 28-36.

The radiation protection of reactor operators of the Borssele reactor is described. The author concludes that working in this reactor is good from the viewpoint of radiation hygiene. Also the amount of radioactive effluents and the produced radioactive waste are mentioned.

1355. SUPERVISION ON INTERNAL CONTAMINATION IN DUTCH NUCLEAR POWER PLANTS (in Dutch). J.A.M.M. KOPS. Measurement of internal contamination. Nederlandse Vereniging voor Stralingshygiene, Eindhoven (Netherlands). CONF-8610457--, Report No. NVS--9, 1987, 12 pp.

The system by which in the Dutch nuclear power stations internal contaminations are controlled and measured is described. Aerosol measurements show that the aerosols are mainly respirable, the concentrations are low and ^{60}Co is by far the most important nuclide. Measurements of the ^{60}Co retention in two people indicate a lower retention than assumed in ICRP 30. The evaluations of effective dose equivalent commitments from measured internal contaminations, by means of the 'conservative' ICRP 30 figures, clearly show that radiation doses resulting from internal contaminations are very low compared to those from external radiation.

1356. CECIL. R.A.S. LEE, S. RUGGIERI, S.A. TROVATO, and C.L. WILLIAMS. Proceedings of a Symposium - Joint ASME-ANS nuclear power conference, Myrtle Beach, SC, US. CONF-8804145--, Apr 17 1988, pp. 75-78.

Conventional methods for inspecting the condition of steam generators and for removing sludge deposits are inadequate. A robotic device, CECIL (Consolidated Edison Combined Inspection and Lancing system), was developed to increase the quality of PWR steam generator tube bundle inspection while at the same time improving the capability for removal of sludge. Recent field testing of CECIL at Consolidated Edison Company's Indian Point Unit 2 has demonstrated the feasibility of fully robotic inspection, sampling, and cleaning of steam generators while reducing radiological exposures normally associated with such activities.

1357. A STORAGE CASK FOR CONSOLIDATED FUEL. A.H. WELLS, T.C. THOMPSON, and T.H. LESSER. Proceedings of a Symposium - Joint ASME-ANS nuclear power conference, Myrtle Beach, SC, US. CONF-8804145--, Apr 17 1988, pp. 133-136.

Nuclear Assurance Corporation has developed a 28 consolidated fuel canister capacity cask (56 PWR assemblies) which will be used at the Surry reactor site. This cask will be US NRC licensed and the consolidated fuel criticality safety must be demonstrated. The worst case for criticality is a container with the fewest allowed number of rods in the canister because this allows more water moderator within the relatively dry fuel lattice. A minimum consolidation factor of 1.73 was selected (352 rods in a canister with a 408 rod capacity) for analysis purposes, which conservatively bounds the factor of 1.9 expected. Criticality control is ensured by using a boron-poisoned fuel basket, which must maintain the fuel/poison geometry in any off-normal condition. A quasi-static structural analysis with a dynamic load factor of 1.5 was performed to demonstrate that the aluminum basket would not yield under end drop or tip over conditions.

1358. CONSTAR DRY STORAGE FOR SPENT FUEL. D.B. YOUNG. Proceedings of a Symposium - Joint ASME-ANS nuclear power conference, Myrtle Beach, SC, US. CONF-[illegible]45--, Apr 17 1988, pp. 137-142.

The Babcock and Wilcox Company (B and W) has developed a low cost, flexible design alternative to metal dry cask storage of spent nuclear fuel. It's called the CONSTAR (Concrete Storage At Reactor/Repository) dry-storage cask system. It not only satisfies storage requirements for all reactor sites, but also addresses the needs of larger government-owned receiving and handling facilities. The cornerstone of the CONSTAR System is a series of concrete-shielded casks that offers two size options. For utilities, it offers the CONSTAR-16/36, which stores 16 PWR or 36 BWR intact fuel assemblies or an equal number of consolidated canisters. An optional basket design to accommodate round canisters that would be compatible with DOE repository or MRS planning is also available. For larger receiving and handling facilities, the expanded version CONSTAR-28/63 is available. It interfaces well with larger fuel storage and handling requirements where crane capacities may not be as restrictive as those at utility sites. Dry casks offer one of the most attractive fuel storage alternatives because capacity additions may be purchased incrementally, and they represent temporary site structures. Concrete shielding offers the additional advantage of lower cost and quicker availability than metal casks.

1359. SPENT FUEL METAL STORAGE CASK PERFORMANCE TESTING AND FUTURE SPENT FUEL CONCRETE MODULE PERFORMANCE TESTING. M.A. MCKINNON and J.M. CREER. Proceedings of a Symposium - Joint ASME-ANS nuclear power conference, Myrtle Beach, SC, US. CONF-8804145--, Apr 17 1988, pp. 157-164.

REA-2023 (currently marketed by Mitsubishi Heavy Metals of Japan as an MSF IV), Gesellshaft fur Nuklear Service (GNS) CASTOR-V/21, Transnuclear TN-24P, and Westinghouse MC-10 metal storage casks, have been performance tested under the guidance of the Pacific Northwest Laboratory to determine their thermal and shielding performance. The REA-2023 cask was tested under Department of Energy (DOE) sponsorship at General Electric's facilities in Morris, Illinois, using BWR spent fuel from the Cooper Reactor. The other three casks were tested under a cooperative agreement between Virginia Power Company and DOE at the Idaho National Engineering Laboratory (INEL) by EG and G Idaho, Inc. using intact spent PWR fuel from the Surry reactors. The Electric Power Research Institute (EPRI) made contributions to both programs. A summary of the various cask designs and the results of the performance tests is presented.

1360. PASSIVE SAFETY IN THE SAFR DESIGN. R.T. LANCET, J.E. BRUNINGS, J.C. MILLS, and R.D. RUTHERFORD. Proceedings of a Symposium - Joint ASME-ANS nuclear power conference, Myrtle Beach, SC, US. CONF-8804145--, Apr 17 1988, pp. 201-206.

The Sodium Advanced Fast Reactor (SAFR) has been designed to provide an advanced level of safety assurance. A defense-in-depth design approach to the accommodation of transients is utilized to protect both plant investment and public safety. A hierarchy of highly reliable engineered systems is used for the initial lines of defense. The ultimate safety protection mode relies on the inherent response of the SAFR plant to ensure a safe response to all credible events in addition to postulated accidents without scram. This inherency is made economically possible by such distinct SAFR design characteristics as a pool-type configuration, a metal-fuel core, and natural convection decay heat removal systems. Transient analyses and probabilistic risk assessment results are presented to demonstrate the viability of the SAFR safety approach.

1361. AN INHERENTLY SAFE MODULAR HIGH TEMPERATURE GAS-COOLED REACTOR. G. JONES, D. MEHTA, H. CHI, and D. DILLING. Proceedings of a Symposium - Joint ASME-ANS nuclear power conference, Myrtle Beach, SC, US. CONF-8804145--, Apr 17 1988, pp. 207-210.

The Modular High Temperature Gas-Cooled Reactor design has been evolving over the last few years to the point where it is now under review by the U.S. Nuclear Regulatory Commission. Key features of this design are the use of fuel particle containment of fission products instead of major structural containment, and reduction in size of the reactor such that core heatup as a result of loss of forced circulation or loss of coolant will not result in fuel damage. These features result in a new approach to licensing and a capability to demonstrate the response to key design basis events in an operating plant. This paper gives the current status of the design and some of the new approaches being adopted.

1362 AN EXPERT SYSTEM FOR THE ANALYSIS OF JOBS WITH RADIATION. F.A. TRUJILLANO and M.M. GARCIA. Transactions of the American Nuclear Society (USA). CONF-870837--, Vol. 54, Aug 30 1987, p. 154.

The SPR expert system being developed for application in Spanish nuclear power plants with an instruction level corresponding to an advanced prototype seeks to accomplish the following: The analysis in real time of data on jobs with radiation that because of their volume,would overload a human specialist; The extraction of useful knowledge from each experience for storage and future use; The incorporation of the knowledge acquired into working methods and procedures for improved achievement of the as-low-as-reasonably-achievable concept; and the ability to analyze foreseeable situations in order to optimize working procedures in advance.

1363. THE DUTY HEALTH PHYSICIST PROGRAM AT BYRON NUCLEAR POWER STATION (in Byron). D.G. GOLDSMITH and T.R. CAREY. Transactions of the American Nuclear Society (USA). CONF-870837--, Vol. 54, Aug 30 1987, p. 185.

The Duty Health Physicist Program at Byron Station was established to deal with routine health physics tasks and provide an interface between frontline and upper radiation-chemistry management. The program consists of a weekly rotation of selected members of the health physics staff into the duty health physicist position to handle the assigned duty tasks. The tasks include, but are not limited to, daily isotopic and air sample review,effluent release package review, maximum permissible concentration calculations, dose approvals, as-low-as-reasonably-achievable action review of pending jobs, and general availability to answer questions and address problems in health-physics-related areas of plant operation. The daily attendance of the duty health physicist at the radiation-chemistry and station plan-of-the-day meetings has increased the overall presence and visibility of the health physics program to upper station management and other station departments. Since its inception in July of 1985, the Duty Health Physics Program has been a major contributor to the observed 50% reduction in reportable personnel errors in the radiation-chemistry department (based on personnel-error-related deviation reports and license event reports generated on the radiaon Station). Although difficulty to quantify, other important benefits of this program are also discussed in this paper.

1364. CORROSION PREVENTION IN WATER-STEAM CIRCUITS OF POWER PLANTS - PRINCIPLES OF SELECTING THE WATER-CHEMICAL CONDITIONS (in German). J. BOSHOLM and V. ENDER. Kernenergie (German Democratic Republic), Vol. 32, Apr 1989, pp. 154-158.

Based on thermodynamical considerations, the water-chemical conditions appropriate for forming and maintaining protective oxide films on different materials are derived. In salt-free waters the oxidation-reduction potential and the pH determine stability of the oxides whereby, due to the minimal solubility, the pH should correspond to the isoelectric point of the oxide concerned as far as possible. In the presence of salts the alkalinity compensating the formation of metal salts has primary significance. In order to simplify the water-chemical conditions, a single-metal design of water-steam circuits is desirable.

1365. DECOMMISSIONING OF THE SHIPPINGPORT ATOMIC POWER STATION. T.S. LAGUARDIA IBC Technical Services Ltd., London (UK). Proceedings of a Symposium - High temperature superconductors: progress and prospects, London, GB. CONF-880746--, July 6, 1988.

The Shippingport reactor was originally designed as a pressurized water reactor and operated for approximately 10 years in that mode. Later, in 1967 it was converted to a light water breeder reactor and continued its operation until 1985, when the reactor was shut down. However, the decommissioning planning for Shippingport was begun in 1979. Detailed engineering and planning was undertaken to look at alternatives for disposal of the reactor vessel, the overall detailed estimated costs, the exposure to the workers and the waste volume generated and to prepare activity specifications for performance of the work. The program scope and component removal are detailed. The scarification of contaminated concrete, building demolition, special tools and equipment needed and work performance data are described. The successful removal of the primary system components and piping has been completed.

1366. COMPARING EXPRESSED AND REVEALED PREFERENCES FOR RISK REDUCTION: DIFFERENT HAZARDS AND QUESTION FRAMES. T.L. MCDANIELS. Risk Analysis (USA), Vol. 8, Dec 1988, pp. 593-604.

Studies often note the wide differences that exist in costs per death avoided across US federal programs and regulatory contexts. This paper explores two new, related explanations for these differences. First, it argues that the patterns of revealed preferences (public allocations) may be related to public values, which are measured here through subjects' expressed preference responses to a contingent valuation survey regarding risk reduction. Subjects' expressed values are compared to actual (and proposed) costs of safety regulations for a similar set of hazards. The authors discover strong congruence in the ranking of expressed values and actual values. Second, the paper presents the results of a subsequent survey that investigates why the patterns observed in the first survey might occur. It suggests that one reason for the observed similarities between revealed and expressed preferences may be in how choices are framed. The paper hypothesizes that both subjects and decision makers may frame valuation decisions in the same way: as percentage changes from the reference point provided by the base rate of deaths for that hazard.

1367. US NUCLEAR REGULATORY COMMISSION SAFETY GOAL POLICY: A CRITICAL REVIEW. V.M. BIER. Risk Analysis (USA), Vol. 8, Dec 1988, pp. 563-568.

The most recent US Nuclear Regulatory Commission (NRC) statement of safety goal policy is a significant advance over previous versions. However, some areas of the policy are still in need of refinement, and the resolution of several key questions was deferred pending further review. To clarify some of these issues, this paper presents a critical review of the NRC safety goal policy to date.

1368. DECOMMISSIONING OF THE WINDSCALE ADVANCED GAS COOLED REACTOR. J. JONES. International Atomic Energy Agency, Vienna (Austria). Proceedings of a Symposium - International conference on nuclear power performance and safety, Vienna, AT. CONF-8709263--, Sep 28 1987, v. 5.

The Windscale Advanced Gas Cooled Reactor (WAGR), built as a prototype for the commercial AGRs, was commissioned in the early 1960s and ran very successfully until 1981 when on completion of the experimental programme it was shut down. Prior to this final shutdown, the United Kingdom Atomic Energy Authority (UKAEA), increasingly aware of the need of the nuclear industry to demonstrate that a reactor could be decommissioned both safely and economically, decided to authorize the decommissioning of WAGR to Stage 3 (that is to a green field site). The reactor rated at 33 MW(e) comprises a graphite core within a steel pressure vessel housed within a reinforced concrete bioshield which is all housed within the spherical reinforced concrete containment buildings. The similarities of this reactor to the much larger Magnox reactors are sufficient to be able to provide valuable data and experience for the eventual decommissioning of these reactors. An account of the progress to date in decommissioning WAGR together with information on the decommissioning development work that is being undertaken is presented. It is possible that the estimates produced for the overall project cost of dismantling WAGR from Stage 1 to Stage 3 can establish indicative decommissioning costs for the other reactors in use in the United Kingdom.

1369. PIPING ISSUES IN NUCLEAR SYSTEMS IN THE USA. A. TABOADA, L.C. SHAO, and R.H. VOLLMER. International Journal of Pressure Vessels and Piping (UK). CONF-8708329--, Vol. 34, Aug 24 1987, p. 34.

Issues dealing with major problems of piping for nuclear reactor systems are discussed. Plans for implementation of the USNRC Piping Review Committee recommendations on resolving the intergranular stress corrosion cracking problem in boiling water reactors are detailed, including alternative acceptable mitigative actions and guidelines for crack evaluation and repairs. The problem of pipe wall thinning due to erosion-corrosion, highlighted by a major pipe break in the Surry Power Station, is described.

1370. MEASUREMENT OF THE NEUTRON SPECTRUM INSIDE THE CONTAINMENT BUILDING OF A PWR. R. BIRCH, H.J. DELAFIELD, and C.A. PERKS. Radiation Protection Dosimetry (UK). CONF-871020--, Vol. 23, Oct 12 1987, pp. 281-284.

Neutron spectra inside the containment building of a PWR at Gosgen, Switzerland have been measured using a high resolution, transportable, neutron spectrometry system based on proportional counters. Small spherical proportional counters, filled with hydrogen to pressures of 100, 300 and 1000 kPa, were used to cover the energy range from 50 keV to 1.5 MeV. A larger cylindrical counter, filled with 600 kPa of ^{4}He plus 400 kPa Ar was used to extend the measurement range up to 16 MeV. Compact electronics provided simultaneous data acquisition for up to four counters allowing spectrum measurements to be performed in dose equivalent rates down to 50 μSv center dot h^{-1}. Supporting measurements were made with a multisphere spectrometer to determine the fraction of neutrons below 50 keV. The neutron spectra measured at two positions show a steady rise in the neutron fluence as the neutron energy decreases, with very few neutrons having energies above 700 keV.

1371. IMPLICATIONS OF THE SURRY PIPING FAILURE FOR OTHER NUCLEAR AND FOSSIL UNITS. V.K. CHEXAL and R.L. JONES. International Journal of Pressure Vessels and Piping (UK), Vol. 34, August 24, 1987, p. 34.

A description is given of the key elements of an inspection plan to address utility concerns of flow assisted corrosion due to single-phase flow in carbon steel piping. The key elements of the plan are (a) where to look, (b) how to look, (c) when to look, and (d) how to respond. The plan is designed to provide utilities with the ability to predict piping thickness as a function of plant life for a given component and to assess the costs and benefits of a variety of remedy-repair options. The thrust of the inspection plan is to (a) conduct appropriate analysis and a limited but thorough baseline inspection program, (b) determine the extent of thinning, if any, and repair/replace components as necessary, and (c) perform follow-up inspections to confirm or quantify thinning and take longer term corrective actions (i.e. adjust chemistry, operating parameters, or others) as appropriate.

1372. BELOW REGULATORY CONCERN OWNERS GROUP: EVALUATION OF DRY ACTIVE WASTE MONITORING INSTRUMENTS AND TECHNIQUES: FINAL REPORT. V.W. THOMAS, G.W.R. ENDRES, S.E. MERWIN, M.P. MOELLER, D.E. ROBERTSON, and J.A. YOUNG. Electric Power Research Inst., Palo Alto, CA (USA) ; Pacific Northwest Lab., Richland, WA (USA). Report No. EPRI-NP--5682, Mar 1989, 156 pp.

The disposal of low-level radioactive waste (LLRW) is costly, so most nuclear power stations have found or will find that it is cost-effective to dispose of dry active waste (DAW) with activity levels that are Below Regulatory Concern (BRC) at a sanitary landfill or incinerator. It appears that substantial volumes of DAW can be exempted from disposal as LLRW if the maximum exposure to an individual member of the public from BRC waste does not exceed a few mrem per year effective dose equivalent. The Electric Power Research Institute (EPRI) has requested that Battelle, Pacific Northwest Laboratories (BNW) evaluate instruments and methods that could be used to measure surface contamination (activity per unit area) and radioactivity concentrations (activity per unit mass or volume) in BRC waste. Instrumentation utilized in a DAW BRC monitoring program must be capable of satisfying performance objectives. This instrumentations must measure bulk concentrations of radioactivity in DAW to assure that annual inventory disposal limits are not exceeded at each disposal site; measure radionuclide concentrations in disposal containers (e.g., bags, boxes, etc.) to assure that maximum allowable concentration limits in the DAW are not exceeded; assure that discrete radioacti, if present in DAW, do not exceed maximum permissible activity limits; and possess detection capability to allow utilities to set operational limits between the detection limit and the disposal limit at their option. Our evaluations indicate that bag monitors and barrel counters have the necessary sensitivity to meet all of these objectives.

1373. DEVELOPMENT PROGRAMME FOR THE DECOMMISSIONING OF THE WINDSCALE ADVANCED GAS-COOLED REACTOR. E.H. PERROTT and J.R. WAKEFIELD. Institution of Mechanical Engineers, London (UK) ; British Nuclear Energy Society, London ; Institution of Nuclear Engineers, London UK). Proceedings of a Symposium - IMechE international conference on decommissioning of major radioactive facilities, London, GB. CONF-881008--, Oct 11 1988, pp. 201-207.

The major items in the UKAEA Windscale Advanced Gas-cooled Reactor (WAGR) decommissioning development programme are discussed. The areas covered include plant radioactivity assessment (both for operator safety and waste management), dismantling techniques, remote manipulation and viewing requirements, and filtration and decontamination techniques.

1374. DECONTAMINATION OF MAGNOX BOILERS. D. BRADBURY, C. KIRBY, M.G. SEGAL, and W.J. WILLIAMS. Institution of Mechanical Engineers, London (UK) ; British Nuclear Energy Society, London ; Institution of Nuclear Engineers. Proceedings of a Symposium - IMechE international conference on decommissioning of major radioactive facilities, London, GB. CONF-881008--, Oct 11 1988, pp. 131-138.

Decontamination may have a significant role to play in the decommissioning of boilers from the UK Magnox stations. There are two possible roles of decontamination, namely dose reduction for the process of cutting (if necessary) and removal, or cleaning material removed for free release as non-radioactive scrap (since the boilers are only lightly contaminated). If decontamination is worthwhile at all, present studies are suggesting that the most beneficial technique will probably be to decontaminate the whole boiler in-situ, using engineering techniques already well established for decontamination of water reactors. Secondary decontamination of components may also be beneficial which, when coupled with advanced monitoring techniques such as assay by melting, would offer the possibility of reducing radioactivity levels below the free release limit of 400 Bq/kg. Secondary waste is an important issue which is also being addressed in the present studies.

1375. EXAMPLE OF POWER REACTOR COOLING CIRCUIT DECONTAMINATION. M. MONTJOIE, J.R. COSTES, and F. JOSSO. Institution of Mechanical Engineers, London (UK) , British Nuclear Energy Society, London ; Institution of Nuclear Engine. Proceedings of a Symposium - I-MechE international conference on decommissioning of major radioactive facilities, London, GB. CONF-881008--, Oct 11 1988, pp. 127-130.

This paper describes the process developed to decontaminate the CO_2 cooling system of the G2 gas-cooled reactor with the objective of recycling the majority of the scrap material as conventional waste. The process is first implemented before dismantling to facilitate pipe cutup operations. If necessary, a second application is performed in a sealed cell to meet the specified objective.

1376. DECOMMISSIONING THE WINDSCALE ADVANCED GAS-COOLED REACTOR - A DEMONSTRATION PROJECT FOR UK REACTORS. P.J. THOMAS, T. BOORMAN, and A.R. GREGORY. Institution of Mechanical Engineers, London (UK) ; British Nuclear Energy Society, London ; Institution of Nuclear Engine. Proceedings of a Symposium - IMechE international conference on decommissionioning of major radioactive facilities, London, GB. CONF-881008--, Oct 11 1988, pp. 1-10.

A description is given of the project to decommission to Stage 3 the Windscale Advanced Gas-cooled Reactor, which is being carried out by the UKAEA with the support and participation of the UK Generating Boards. The engineering programme and progress to date are detailed.

1377. REMOTE CUTTING SYSTEMS FOR DISMANTLEMENT OF THE JAPAN POWER DEMONSTRATION REACTOR CONCRETE BIOLOGICAL SHIELD. H. NAKAMURA, T. NARAZAKI, H. YASOSHIMA, and T. KONNO. Institution of Mechanical Engineers, London (UK) ; British Nuclear Energy Society, London ; Institution of Nuclear Engine. Proceedings of a Symposium - I-MechE international conference on decommissioning of major radioactive facilities, London, GB. CONF-881008--, Oct 11 1988, pp. 223-228.

The decommissioning programme of the Japan Power Demonstration Reactor (JPDR) is under way. For the purpose of dismantling the biological shield of JPDR safely without increasing the radiation exposures to workers, two remote cutting systems, namely, a diamond sawing and coring and an abrasive-waterjet, have been developed. These systems are to be applied to the highly-activated region of the biological shield. The features of these systems and the dismantling procedures by these systems are described in this paper including the fundamental test results used for the system design.

1378. DECONTAMINATION OF SYSTEMS AND COMPONENTS FOR DECOMMISSIONING. H. WILLE and H.-O. BERTHOLDT. IMechE international conference on decommissioning of major radioactive facilities. CONF-881008--, Oct 11 1988, pp. 145-149.

For the decontamination of complete systems the CORD process is applied. This process is a "soft", low chemical concentration (2000 ppm/cycle), multicycle decontamination process. It can be performed with a mobile external system or, for complete primary loop decontamination, with the systems of the nuclear power plant itself. The amount of waste can be reduced considerably if the decontamination solution is evaporated. Actual decontamination tasks resulted in activity level reductions of 6 - 8 in the compartments of the systems, associated with local decontamination factors (DFs) of 20 or more. A severe problem is the decontamination for unrestricted release of dismantled components and material. A demonstration in decontamination was carried out with recirculation piping sections, which were replaced in 1985 in a BWR. About 15 tons of piping and valves were decontaminated from an initial contact dose rate of 30000 μSv/h to 1 - 2 μSv/h in a combined chemical and electrochemical treatment by CORD and ELPO decontamination processes. This enabled the requirements of ICRP for unrestricted release to be easily met.

1379. BELOW REGULATORY CONCERN OWNERS GROUP: RADIONUCLIDE CHARACTERIZATION OF POTENTIAL BRC WASTE TYPES FROM NUCLEAR POWER STATIONS: FINAL REPORT.

D.E. ROBERTSON, C.W. THOMAS, D.C. HETZER, N.L. WYNHOFF, P.J. RANEY, J.D. FORSYTHE, J.S. SCHMITT, R.L. BUSCHBOM, and K.T. HARA. Electric Power Research Inst., Palo Alto, CA (USA) ; Pacific Northwest Lab., Richland, WA (USA). Report No. EPRI-NP--5677, Mar 1989, 224 pp.

The objective of this study was to perform a detailed radiological characterization and statistical assessment of the measured radionuclide distributions for four candidate "below regulatory concern" (BRC) waste types from commercial nuclear power stations. These measurements and statistical evaluations will provide the bases for conducting detailed dose assessments associated with various disposal options for BRC wastes. The four waste types selected were dry active waste (DAW), oil, soil, and secondary side ion exchange resin. The measurement included gamma-spectrometric analyses of 558 total samples, including 102 DAW samples, 231 oil samples, 142 soil samples, and 83 resin samples. Radionuclides usually detected during the gamma spectrometry included ^{60}Co, ^{137}Cs, ^{134}Cs, ^{54}Mn, and ^{58}Co. Frequently, ^{95}Zr, ^{95}Nb, ^{106}Ru, and ^{125}Sb were detected, especially in wastes from plants which had experienced a relatively high rate of fuel cladding failures. Selected aliquots of the gamma-counted samples were radiochemically analyzed for ^{14}C, 55Fe, ^{63}Ni, ^{90}Sr, ^{129}I, ^{238}Pu, and ^{239}Pu,^{240}Pu. The gamma-emitting radionuclides in the BRC wastes, which are the primary contributors to the limitingto the general population for BRC waste disposal, were dominated by ^{60}Co and ^{137}Cs. The variability in the major gamma-emitting nuclides was assessed as a function of reactor type and waste stream. A similar evaluation was conducted for the difficult-to-measure radionuclides. From these assessments, it was concluded that the variability in radionuclide composition from all waste streams and all plants was sufficiently small to justify the development of a single, conservative radionuclide composition that would be representative of the BRC waste generated at commercial US nuclear power stations.

1380. 'RADIOLOGICAL PRINCIPLES TO BE APPLIED FOR HARMLESS RECYCLING AND REUSE OF LOW-LEVEL RADIOACTIVE STEEL AND IRON SCRAP FROM NUCLEAR POWER PLANT', A RECOMMENDATION OF THE SSK (in German).

R. NEIDER. Reuse of residual materials from repai. and dismantling of nuclear facilities. Kernforschungsanlage Juelich G.m.b.H. (Germany, F.R.) ; Bundesanstalt fuer Materialpruefung, Berlin (Germany, F.R.). CONF-8711296--, Report No. Juel-Conf--62, Jul 1988, pp. 33-50.

The recommendation presented here has been worked out over several years by the SSK (Strahlenschutzkommission) and an SSK sub-comittee on 'Radiological protection with regard to radioactive scrap and other residual materials'. The recommendation is concerned with: (a) Unconditional release for reuse; (b) radiation limits for scrap accepted for unrestricted reuse; (c) radiation limits for restricted reuse of scrap. The recommendation explains the criteria and limits defined.

1381. DRESDEN UNIT 2 HYDROGEN WATER CHEMISTRY: CHEMICAL SURVEILLANCE, OXIDE-FILM CHARACTERIZATION, AND RECONTAMINATION DURING CYCLE 10: FINAL REPORT. C.P. RUIZ, J.P. PETERSON, R.N. ROBINSON, and L.L. SUNDBERG. Electric Power Research Inst., Palo Alto, CA (USA) ; General Electric Co., San Jose, CA (USA). Report No. EPRI-NP--6278-M, Mar 1989, 25 pp.

This document provides an Executive Summary of work performed under Project RP1930-7, BWR Hydrogen Water Chemistry - Chemical Surveillance. It describes the work performed to monitor chemical and radiological performance at Commonwealth Edison's Dresden Nuclear Power Station Unit 2 during Cycle 10, its second full fuel cycle on Hydrogen Water Chemistry. It includes the results of water chemistry measurements, shutdown gamma scan/dose rate measurements, and the results of stainless steel oxide film characterization. This experience at Dresden-2 continues to demonstrate that a plant can operate on Hydrogen Water Chemistry with only minor impact on plant parameters, compared with the beneficial effect on intergranular stress corrosion cracking (IGSCC) mitigation of sensitized stainless steel components.

1382. IMPLICATIONS OF POSSIBLE REDUCTION IN RADIATION EXPOSURE LIMITS: FINAL REPORT. J.E. LE SURF and J.I. CEHN. Electric Power Research Inst., Palo Alto, CA (USA) ; Le Surf (J.E.), Niagara Falls, ON (Canada). Report No. EPRI-NP--6291, Mar 1989, 62 pp.

The International Commission on Radiological Protection (ICRP) is currently reviewing the recommended limits on radiological exposure of nuclear workers. The findings of the review are expected to be issued in 1990 or 1991. There are indications that the revised recommendations will be for reduced limits. Based on new estimates of the exposures received by Japanese atomic bomb victims, the new recommended limit may be in the range 1 to 2.5 rem/year. Many countries will accept these recommendations unilaterally and some have already issued interim guidance statements recommending an annual exposure limit of 1.5 rem. Although the USA is not bound by these recommendations, it is unlikely that the NRC would continue with a set of limits higher than the rest of the world adopted. This report is preliminary examination of how such a change in worker exposure limit could be implemented in the US nuclear industry. No attempt is made to quantify either the impact or the benefits of such a change. The likely effect on the nuclear industry of a major decrease in exposure limits is discussed and the approaches taken by other countries with major nuclear programs to minimize radiation exposures is presented. Changes to the philosophy of radiation protection in this country that would alloccommodation of lower limits are suggested. Improvements to dose tracking and dose monitoring techniques are discussed. Methods for reducing existing radiation fields and for preventing future radiation field increases are briefly reviewed.

1383. DEVELOPMENT OF HPLC TECHNIQUES FOR THE ANALYSIS OF TRACE METAL SPECIES IN THE PRIMARY COOLANT OF A PRESSURISED WATER REACTOR. K.R.P. BARRON. Council for National Academic Awards, London (UK), Mar 1988, 267 pp.

The need to monitor corrosion products in the primary circuit of a pressurised water reactor (PWR), at a concentration of 10pg ml^{-1} is discussed. A high performance liquid chromatography (HPLC), system was developed to determine trace metal species in simulated PWR primary coolant. An on-line preconcentration system was developed. Separations were performed on Aminex A9 and Benson BC-X10 analytical columns. Detection was by post column reaction with Eriochrome Black T and Calmagite Linear calibrations of 2.5-100ng of cobalt (the main species of interest), were achieved using up to 200ml samples. The detection limit for a 200ml sample was 10pg ml^{-1}. In order to achieve the desired aim of on-line collection of species at 300 °C, the use of inorganic ion-exchangers is essential. Titanium dioxide, zirconium dioxide, zirconium arsenophosphate and pore controlled glass beads, were used for the preconcentration of trace metal species. The performance of these exchangers, at ambient and 300 °C was assessed by their inclusion in the developed analytical system and by the use of radioisotopes. The particular emphasis during the development has been upon accuracy, reproducibility of recovery, stability of reagents and system contamination, studied by the usesotopes and response to post column reagents. A monitoring system that can follow changes in coolant chemistry, on deposition and release of m... al species in simulated PWR water loops has been developed. On-line detection of cobalt at 11pg ml^{-1} was recorded.

1384. INDUCED RADIOACTIVITY OF A CONCRETE USED FOR NUCLEAR FACILITY SHIELDS (in Russian). P.A. LAVDANSKIJ, V.M. NAZAROV, and M.V. FRONTAS'EVA. Atomnaya Energiya (USSR), Vol. 64, Jun 1988, pp. 419-422.

An attempt is made to determine experimentally the presence and concentration of ^{45}Sc, ^{59}Co, ^{133}Cs, ^{151}Eu, ^{153}Eu, ^{181}Ta nuclides in the most widespread concrete aggregates (limestone, granites, and serpentinites) and also to calculate the aggregate activity appearing in the process of NPP operation. Aggregate samples from different deposits used to produce NPP biological shields, are investigated. Nuclide concentrations are determined by the activation method after irradiation in the IBR-2 reactor channel. Thermal, resonance and fast neutron flux densities constitute $1.1x10^{12}$; $0.23x10^{12}$ and $1.4x10^{12} cm^{-2}xs^{-1}$, respectively. Analysis of the data obtained shows that long-lived activity of aggregates at cooling time 0.5 year is determined by europium, cobalt and cesium. Induced specific activity of serpentinites is the maximum one of all the aggregates investigated.

1385. MANTIS HELPS COLLECTION OF PIPING EROSION/CORROSION DATA. D.B. FAIRBROTHER. Nuclear Engineering International (UK), Vol. 33, Nov 1988, pp. 52- 54.

One of the challenges posed by erosion/corrosion inspection of piping in nuclear power plants is handling the large amounts of data generated. Babcock and Wilcox's MANTIS (Modular Automated Nondestructive Thickness Inspection System) uses a small battery-powered, waterproof, dustproof computer to log thickness readings. The readings are obtained from a flaw detector with an analog output signal that is proportional in thickness.

1386. RADIATION CONTROL PROGRAMME AND EXPOSURE TO WORKERS AND THE PUBLIC IN THE DECOMMISSIONING OF THE JPDR. H. MATSUI, Y. IKEZAWA, C. NAKAMURA, Y. ASHIKAGAYA, M. OSHINO, and T. NUMAKUNAI. International Atomic Energy Agency, Vienna (Austria). Proceedings of a Symposium - International conference on radiation protection principles in nuclear energy, Sydney, AU. CONF-880401--, Apr 18 1988, v. 2.

The decommissioning of the Japan Power Demonstration Reactor is intended for complete removal of all the installations and buildings, using various newly developed dismantling techniques. The dose rates are highest at the reactor internals and at the connection of the primary coolant pipes and the upper part of the reactor vessel - 900 R/h and 1000 mR/h, respectively. Remote high dose rate measuring instruments in air and under water, a respirable dust monitor, an ingress and egress control system, a contamination inspection monitor and a waste package contamination and dose rate monitor have been developed and prepared for radiation surveillance. The collective doses to workers over the whole period of decommissioning are estimated to be 1020 mancenter dotmSv of external dose and 1.5 mancenter dotmSv of internal dose. The dose to the public is estimated to be $2.3x10^{-10}$ mSv by inhalation, $7.1x10^{-11}$ mSv by radioactive cloud, $7.4x10^{-5}$ mSv by ingestion, and $2.8x10^{-5}$ mSv by skyshine and direct gamma radiation.

1387. MECHANICAL METHODS OF IMPROVING RESISTANCE TO STRESS CORROSION CRACKING IN BWR PIPING SYSTEMS. J.S. ABEL, J. TITRINGTON, R. JORDAN, J.S. POROWSKI, W.J. O'DONNELL, M.L. BADLANI, and E.J. HAMPTON. International Journal of Pressure Vessels and Piping (UK). CONF-8708329--, Vol. 34, Aug 24 1987, p. 34.

Pipelocks and the mechanical stress improvement process (MSIP) have been applied in BWR plants. Pipelocks restore the integrity of the weldments with identified cracks. MSIP removes residual tensile stresses from weldments, thus preventing initiation of cracks or retarding growth of pre-existing flaws in piping systems. Extensive qualification has been completed for MSIP under US Nuclear Regulatory Commission and Electric Power Research Institute sponsorship. The use of mechanical methods becomes especially adequate for reactor safe-ends including bi- or tri-metallic joints. The use of overlay technique or induction heat stress improvement is more difficult due to high thermally induced strains at the strong discontinuity interface between materials of different thermal expansion. Basic concepts and practical application of mechanical methods to inhibit stress corrosion attack are described.

1388. STRESS CORROSION CRACKING OF NUCLEAR REACTOR PRESSURE VESSEL AND PIPING STEELS. M.O. SPEIDEL and R.M. MAGDOWSKI. International Journal of Pressure Vessels and Piping (UK). CONF-8708329--, Vol. 34, Aug 24 1987, p. 34.

This paper presents an extensive investigation of stress corrosion cracking of nuclear reactor pressure vessel and piping steels exposed to hot water. Experimental fracture mechanics results are compared with data from the literature and other laboratories. Thus a comprehensive overview of the present knowledge concerning stress corrosion crack growth rates is provided. Several sets of data confirm that 'fast' stress corrosion cracks with growth rates between 10^{-8} and 10^{-7} m/s and threshold stress intensities around 20 MN $m^{-3/2}$ can occur under certain conditions. However, it appears possible that specific environmental, mechanical and metallurgical conditions which may prevail in reactors can result in significantly lower stress corrosion crack growth rates. The presently known stress corrosion crack growth rate versus stress intensity curves are discussed with emphasis on their usefulness in establishing safety margins against stress corrosion cracking of components in service. Further substantial research efforts would be helpful to provide a data base which permits well founded predictions as to how stress corrosion cracking in pressure vessels and piping can be reliably excluded or tolerated. It is emphasized, however, that the nucleation of stn cracks (as opposed to their growth) is difficult and may contribute substantially to the stress corrosion free service behaviour of the overwhelming majority of pressure vessels and pipes.

1389. RECENT TOPICS OF FLAW PREVENTION. S. FURUYA, N. MORISHIGE, Y. TANAKA, T. UMEMOTO, and M. AMANO. International Journal of Pressure Vessels and Piping (UK). CONF-8708329--, Vol. 34, Aug 24 1987, p. 34.

Flaws resulting from corrosion are still a significant issue in nuclear power plants. Prevention of these flaws is one of the biggest themes of plant maintenance and new countermeasures have been developed year by year. In this paper three newly proposed techniques are presented: (1) ceramics coatings on flange seal surface by surface hardening due to laser-alloying to prevent pitting corrosion; (2) Induction Heating Stress Improvement (IHSI) on nozzle safe-end with thermal sleeve by using forced cooling; and (3) TIG Torch Heating Stress Improvement (THSI) to reduce welding residual stress on small diameter pipes.

1390. LIFE EXTENSION AND WATER CHEMISTRY OF LWRS. I.E. LE SURF. Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP. CONF-8804174--, Apr 13 1988, v.2.

There is considerable economic incentive to extend the operating life-time of nuclear power plants, both boiling water reactors (BWRs) and pressurized water reactors (PWRs). Plant life may be limited by degradation of critical components, generally by corrosion. Careful control of water chemistry to minimize corrosion can play an important role in making plant life extension (PLEX) possible. Major components, such as the prima[illegible] em piping in BWRs and steam generators in P[illegible], have already had to be replaced at several plants due to excessive corrosion. Ways to avoid this corrosion are discussed in the paper. In addition, there are certain critical components within the core of the reactor which cannot be replaced without essentially rebuilding the plant. Chemical control to preserve the life of these components is also discussed, as is the life of the pressure vessel itself.

1391. DEVELOPMENT OF HIGH TEMPERATURE ADSORBENT IN PWR PRIMARY SYSTEM. H. KITAO, T. MIYAZAKI, K. HATA, and N. NAKAZIMA. Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP. CONF-8804174--, Apr 13 1988, v. 2.

Radiation exposure reduction in PWR is one of the most important problems to be solved. We have developed a high temperature Co adsorbent (HTA), which could be directly applied under primary reactor coolant conditions. This adsorbent was Fe-Ti-O system ceramics, and was fabricated to a suitable form for using in a packed column. Through those experiments of adsorption tests, compatibility tests, leaching tests and hot loop tests, it was found that HTA had superior adsorption capability to not only Co and Ni-ion but also many other transition metal ions. And it was also found that HTA was compatible with high temperature water, as well as advantageous for its waste solidification. Based on the experimental results, dose reduction effect was evaluated by a computer code. From this evaluation, it was found that more than 50 % dose reduction could be expected, when an advanced reactor coolant clean-up (RCC) system with HTA would be realized.

1392. PASSIVATION OF RE ^TOR SYSTEM PIPING. PREFILMING EXPERIL CE USING AIR OXIDATION. R.H. ASAY and R.L.A. ROOFTHOOFT. Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP. CONF-8804174--, Apr 3 1988, v. 2.

A pretreatment process for passivating new or decontaminated materials to reduce future radiation level buildup has been successfully demonstrated. The process has been implemented on a full scale basis at three boiling water reactors. In addition, in-plant tests at several pressurized water reactors indicate similar, or improved benefits with this pretreatment process. The benefits in reducing radiation buildup can be achieved in a cost effective manner commercial light water reactor systems.

1393. DEVELOPMENT OF CHEMICAL DECONTAMINATION TECHNOLOGY FOR THE FUGEN. S. OHKUBO, T. OHTA, Y. MAEKAWA, K. FURUKAWA, and H. NAKAYAMA. Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP. CONF-8804174--, Apr 13 1988, v. 2.

A study of chemical decontamination of the reactor cooling system of the Fugen has been made with KURIDECON 203, a dilute decontamination reagent. The heat exchangers of the clean up water system and of the residual heat removal system were decontaminated with KD203. As a result, decontamination factor (DF) was about 20, and it has proved that chemical decontamination contributes to reduce the radioactive exposure.

1394. EXPERIMENTAL DECONTAMINATION OF A REACTOR-WATER CLEAN-UP LINE AT THE DODEWAARD NUCLEAR POWER PLANT BY MEANS OF THE LOMI PROCESS. P.J.C. LETSCHERT, L.M. BUTTER, W.M.M. HUIJBREGTS, and M.G. SEGAL. Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP. CONF-8804174--, Apr 13 1988, v. 2.

An experimental decontamination of a line of the reactor-water clean-up (RWCU) system at the Dodewaard BWR was carried out. A decontamination unit was designed and constructed for this purpose. The RWCU line was divided into two pieces; each part was decontaminated twice. The dose rate of the RWCU line decreased from about 11.5 to 0.45 mSv/h, which resulted in an overall decontamination factor of 25. In all, 65.566 g of metal with a total ^{60}Co activity of 13 GBq was removed. A linear correlation was found between activity and concentration of dissolved metals in the decontamination solution. After decontamination, several pieces of the RWCU line were examined at the laboratory. No corrosion attack of the stainless steel was found.

1395. STUDIES ON FUEL SURFACE CRUD REMOVAL TO REDUCE OCCUPATIONAL RADIATION EXPOSURES. H. HARIYAMA, S. OKUDA, M. MATSUURA, K. ICHIKAWA, S. YAMAMOTO, and S. KASHIWAGI. Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP. CONF-8804174--, Apr 13 1988, v. 2.

This paper presents an unique approach to reduce occupational radiation exposures by removing CRUD deposit attached on fuel surface. The design concept of CRUD removal system is highlighted in conjunction with performance data obtained from series of field experiments. Evaluation of dose reduction effect as a consequence of CRUD removal is also discussed.

1396. PLANNING FOR A BOILING WATER REACTOR CHEMICAL DECONTAMINATION. S.W. JR. WILCZEK and T.R. LAVOY. Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP. CONF-8804174--, Apr 13 1988, v. 2.

Nine Mile Point Unit 1 is a 620 MWe General Electric Boiling Water Reactor. The unit entered commercial service in December of 1969. Several chemical decontaminations of the reactor recirculation system have been performed in the past to support maintenance activities and recirculation piping replacements. These decontaminations were very successful in reducing radiation exposure rates in the work area.

1397. RELEASE AND DEPOSITION OF CORROSION PRODUCTS IN A SIMULATED BWR COOLANT SYSTEM. SHUNJI KATO and TORU IWAHORI. Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP. CONF-8804174--, Apr 13 1988, v. 2.

Information on corrosion products release from structural material is useful for radiation control in reactor coolant systems. The influence of dissolved oxygen on corrosion products release from AISI 304 stainless steel was studied in high purity water from 30 °C to 280 °C which corresponds to the state of BWR coolant system. The release rates of Fe, Ni and Co from stainless steel increased as the temperature was raised from 30 °C and reached the maximum at about 2000 C,then decreased as the temperature was raised. Their release rates were reduced by the addition of dissolved oxygen in about 100 μg/kg at the temperatures in this experiment. On the other hand, the release rate of Cr was maintained at low levels at temperature up to 280 °C, when the concentration of dissolved oxygen was below 1 μg/kg. With the concentration of dissolved oxygen exceeding 1 mg/kg, the Cr was released as chromium anion, especially at high temperatures. This release behavior was possibly caused by the dissolution of chromium-rich films. Accumulation of ionic Co on the zircaloy surface was examined under simulated BWR primary coolant condition using ^{58}Co as a tracer. Among Fe, Cr and Ni components, Cr component showed highest deposition rate on the zircaloy surface. ^{58}Co was likely to be present in this Cr deposits. The rate of ^{58}Co accumulation was reduced on the pretreated zircaloy with overheated steam to form stable ZrO_2 films. Behavior of ionic Co can be explained by the interaction with Cr component.

1398. CORROSION RELEASE. THE PRIMARY PROCESS IN ACTIVITY TRANSPORT. D. LISTER Japan Atomic Industrial Forum, Inc., Tokyo. Proceedings of a Symposium - 21. JAIF annual conference on water chemistry in nuclear power plants, Tokyo, JP. CONF-8804174--, Apr 13 1988, v. 2.

Radiotracing techniques in out-reactor loops have provided kinetic information; coupled with detailed analyses of corrosion product films they have been invaluable in indicating mechanisms. Studies of release from stainless steel, Inconel and Stellite over a rangeof PWR conditions have demonstrated the influence of oxide film formation and mass transport in the fluid on apparent release as measured by radiotracing. When the mechanisms elucidated for cobalt release in PWRs are coupled with those for cobalt deposition on system surfaces, a model to describe the variation of cobalt in a PWR circuit can be formulated. Such a model indicates that, after an initial rapid drop, cobalt levels fall progressively throughout a reactor's lifetime, suggesting that PWR radiation fields will fall after some period of operation. The influence of cobalt content in the materials of construction and of system parameters can be seen. Under normal BWR conditions, cobalt is released in preference to other elements from stainless steel, with a dependence upon $(\text{time})^{0.64}$. In general, release rates are low and comparable with those measured under PWR conditions. If the BWR coolant contains zinc, oxide films are thin and protective so that corrosion release is reduced to even lower values. The release of cobalt, still preferential to that of other elements, then depends upon $(\text{time})^{0.24}$. To impart the maximum benefit, zinc must be present in the coolant during the initial exposure of the stainless steel; its effect then persists during subsequent exposure without zinc. An injection of zinc into the coolant after a few hundred hours' exposure has only a slight effect on film growth and release. (J.P.N.).

1399. EVIDENCE FOR AN INFECTIVE CAUSE OF CHILDHOOD LEUKAEMIA: COMPARISON OF A SCOTTISH NEW TOWN WITH NUCLEAR REPROCESSING SITES IN BRITAIN. L. KINLEN. Lancet (UK), Vol 2, Dec 10 1988, pp. 1323-1327.

Increases of leukaemia in young people that cannot be explained in terms of radiation have been recorded near both of Britain's nuclear reprocessing plants at Dounreay and Sellafield. These were built in unusually isolated places where herd immunity to a postulated widespread virus infection (to which leukaemia is a rare response) would tend to be lower than average. The large influxes of people in the 1950s to those areas might have been conducive to epidemics. The hypothesis has been tested in Scotland in an area identified at the outset as the only other rural area that received a large influx at the same time, when it was much more cut off from the nearest conurbation than at present - the New Town of Glenrothes. A significant increase of leukaemia below age 25 was found (10 observed, expected 3.6), with a greater excess below age 5 (7 observed, expected 1.5).

1400. SURVEY OF PWR WATER CHEMISTRY. J. GORMAN Nuclear Regulatory Commission, Washington, DC (USA). Office of Nuclear Regulatory Research ; Argonne National Lab., IL (US. NUREG/CR--5116, Feb 1989, 137 pp.

This report surveys available information regarding primary and secondary water chemistries of pressurized water reactors (PWRs) and the impact of these water chemistries on reactor operation. The emphasis of the document is on aspects of water chemistry that affect the integrity of the primary pressure boundary and the radiation dose associated with maintenance and operation. The report provides an historical overview of the development of primary and secondary water chemistries, and describes practices currently being followed. Current problems and areas of research associated with water chemistry are described. Recommendations for further research are included.

1401. APPARATIVE DEVELOPMENT FOR INSERVICE INSPECTIONS OF REACTOR PRESSURE VESSELS (in German). H.K. STANGER, W. KAPPES, R. NEUMANN, O.A. BARBIAN, H. BOHN, and K. RUTHROF. Safety and reliability of pressure components with special emphasis on advanced methods of NDT. Vol.1. Staatliche Materialpruefungsanstalt, Stuttgart (Germany,F.R.) ; Fraunhofer-Gesellschaft zur Foerderung der An. CONF-8610283--, Report No. INIS-mf--11896, Oct 9 1986, pp. 18.1-18.27.

These most recent developments of an improved manipulator and completely automated ultrasound data acquisition, processing and evaluation meets all requirements made on in-service and initial inspections of reactor pressure vessels. The ALOK data acquisition and evaluation system improves the capabilities for defect detection and analysis without extending the times for checking the detection data. The fully programmable manipulatur in combination with phased-array probes and ALOK technique result in a major simplification of test procedures, testing times and radiation exposure of inspecting and operating staff.

1402. EVALUATION OF SCC DEFECTS OF STEEL PIPE USING HIGH-ENERGY X-RAY CT SCANNER. SHIGERU MIYOSHI, YOSHINORI TANIMOTO, KIICHIRO UYAMA, and OSAMU TSUJII. Safety and reliability of pressure components with special emphasis on advanced methods of NDT. Vol.1. Staatliche Materialpruefungsanstalt, Stuttgart (Germany,F.R.) ; Toshiba Corp., Fuchu, Tokyo (Japan). Fuchu Wo. OCNF-8610283--, Report No. INIS-mf--11896, Oct 9 1986, pp. 6.1-6.23.

The X-ray computed CT scanner capable of producing sharp tomograms was expected to become practical and revolutional means in nondestructive inspection in industrial field. In Japan, the development of Linac X-ray CT scanner is under way for the inspection of SCC in weld of the nuclear power plant. For development of the Linac-CT, the preliminary experiments for the inspection of SCC artificial defects were performed using 420 kVp industrial X-ray CT scanner (TOSCANER 4200) which had been developed by Toshiba Corporation. This paper includes the background of this program and the summary of preliminary experiments for X-ray CT.

1403 STATUS OF ADVANCED UT SYSTEMS FOR THE NUCLEAR INDUSTRY. M. BEHRAVESH, M. AVIOLI, G. DAU, and S.N. LIU. Safety and reliability of pressure components with special emphasis on advanced methods of NDT. Vol.1. Staatliche Materialpruefungsanstalt, Stuttgart (Germany,F.R.) ; Electric Power Research Inst., Palo Alto, CA. CONF-8610283--, Report No. INIS-mf-11896, 1986, pp. 5.1-5.15.

Advanced systems were developed with the goals of reduced radiation exposure and increased reliability. These systems use scanners for high-density coverage of components, and since some of them are mechanized and remotely controllable, radiation exposures to personnel can greatly be reduced. The AMAPs is a mechanized scanner developed by EPRI. Digital recording and processing of data facilitate repetition of stored scan patterns, data logging for data review, and application of signal processing techniques for noise reduction.

1404. NUCLEAR ACCIDENTS AND INTERNATIONAL OBLIGATIONS. P. STROHL. Newsletter NEA (NEA), Aug 1988, pp. 3-6.

In order to prevent nuclear accidents, the lessons learnt from the Chernobyl catastrophe have shown the need to strengthen international obligations in this field. The adoption of the Conventions on Early Notification and Assistance in case of Emergency in 1986 is a first step in this direction. Also, further efforts should be made concerning public information, harmonization of safety standards and radiation protection as well as compensation for transfrontier damage (NEA).

1405. FUEL PERFORMANCE: ANNUAL REPORT FOR 1987. W.J. BAILEY and S. WU. Nuclear Regulatory Commission, Washington, DC (USA). Div. of Engineering and System Technology ; Pacific Northwest Lab.. NUREG/CR--3950-Vol.5, Mar 1989, 150 pp.

This annual report, the tenth in a series, provides a brief description of fuel performance during 1987 in commercial nuclear power plants and an indication of trends. Brief summaries of fuel design changes, fuel surveillance programs, fuel operating experience, fuel problems, high-burnup fuel experience, and items of general significance are provided. References to more detailed information and related US Nuclear Regulator Commission evaluations are included.

1406. BWR WATER CHEMISTRY (in English). D ANIEL CUBICCIOTTI ROBIN JONES. Nuclear Plant Corrosion Control, Apr-May 1990, p. 42.

Cracking in the recirculation system piping of BWRs has been a costly problem in terms of lost availability. EPRI research has shown that changes in water chemistry can mitigate this problem. In particular, adding hydrogen to plant feedwater has proved to be effective in reducing the oxidizing radiolysis products that promote stress corrosion cracking. Working with the BWR Owners Group and other industry representatives, EPRI has published guidelines for the implementation and use of this remedy, known as hydrogen water chemistry. Ongoing work is aimed at adapting HWC to protect BWR reactor vessels and internals.

AUTHOR INDEX

SUBJECT INDEX

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202 **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	U.S. NUCLEAR REGULATORY COMMISSION	1. REPORT NUMBER (Assigned by NRC. Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG/CR-3469 BNL-NUREG-51708 Vol. 5
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Washington, D.C. 20555

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

One of the functions of the ALARA Center is to collect and disseminate information on dose reduction at nuclear power plants. This is the fifth report in the series of bibliographies of selected readings in radiation protection and ALARA that the Center publishes periodically. The abstracts in this bibliography were selected from proceedings of technical meetings, journals, research reports, searches of information data bases and reprints of published articles provided to us by the authors. The abstracts relate in one way or another to dose reduction at nuclear power plants, whether it is through good water chemistry, improvements in nuclear materials, better control of corrosion, robotics, and remote tooling or good operational health physics.

The report contains 278 abstracts. Subject and author indices are provided. The subject index covers all previous volumes in this series.

All information in the current volume is also available from the ALARA Center's on-line service, which is accessible by personal computer with the help of a modem. The preface of the report explains how the service may be accessed. The on-line service will be updated as new information is received.

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