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Data Base on Dose Reduction Research Projects for Nuclear Power Plants

Prepared by:
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Prepared for
U.S. Nuclear Regulatory Commission

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Data Base on Dose Reduction Research Projects for Nuclear Power Plants

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U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
NRC FIN A3259

ABSTRACT

This is the fifth 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 two of several databases maintained by Brookhaven National Laboratory's ALARA Center for the Nuclear Regulatory Commission.

The research section of the report covers dose reduction projects that are in the experimental or developmental phase. It includes topics such as steam generator degradation, decontamination, robotics, improvements in reactor materials, and inspection techniques. The section on health physics technology discusses dose reduction efforts that are in place or in the process of being implemented at nuclear power plants. A total of 105 new or updated projects are described.

All project abstracts from this report are available to nuclear industry professionals with access to a fax machine through our ACEFAX system or a computer with a modem and the proper communications software through our ACE system. Detailed descriptions of how to access all our databases electronically are in the appendices of the report.

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EXECUTIVE SUMMARY

As part of a project sponsored by the U.S. Nuclear Regulatory Commission, the ALARA Center of Brookhaven National laboratory maintains a number of databases containing information on dose reduction efforts and various aspects of ALARA. Two of these databases are on dose reduction research and health physics technology. The research database consists of the latest findings on ways of reducing doses in nuclear power plants. The information is derived from conferences and industry and government contacts. It includes experimental cobalt reduction methods, primary coolant chemistry studies, progress on full system decontaminations, operational and maintenance techniques, robotics, and other chemical and engineering research programs. The health physics technology database contains information on applied efforts at nuclear power plants which are intended to reduce radiation exposure levels. Examples include radiation shielding during outages, decontaminations, innovative worker training programs, shutdown chemistry guidelines, etc. This report summarizes all the new or updated information from early 1992 to early 1994. It represents 71 research projects and 34 health physics technology projects. (Earlier projects are described in Vol. 1 to 4 of NUREG/CR-4409. They may also be accessed via the ACE or ACEFAX system mentioned below.)

There are two major sections in this volume: research and health physics technology. Each section contains a set of indices that classify the information by title, category, project manager, principal investigator, sponsor, contractor, and subject. The categories are broadly defined areas under which the dose reduction projects are placed. By looking for projects in a particular category the reader will also be made aware of other projects related to the specific subject and yet not be encumbered by projects on topics of less interest.

This report is not designed to give all the details of a project. The purpose is to provide the main objectives, describe some of the most interesting aspects, summarize the conclusions, and provide information on whom to contact for further details. Of particular interest to the ALARA Center is any quantitative implications as far as exposures and exposure rates are concerned. For more information on a project, the reader is referred to the reference section and/or the principal investigators and project managers.

The contents of this report is also available from the ALARA Center in two other forms. First, all the project information is accessible through our ALARA Center Exchange, or ACE, on-line computer database system. This system can be accessed by means of a personal computer, modem, and the appropriate communications software. The user may do a subject search or simply browse through the databases easily and rapidly. One can then print the forms of interest on-line or capture screens containing useful information for off-line printing.

The second method for information retrieval is to request a fax of the forms of interest via ACEFAX, our ALARA Center Exchange FAX-on-demand system. It allows users to call from the handset of a fax machine, choose the project documents of interest in response to voice prompts, and fax the documents to themselves. Both the ACE and ACEFAX systems are described in detail in the appendix of this report.

The ALARA Center also periodically publishes ALARA Notes, a newsletter devoted to issues of interest to the nuclear power/radiation protection community. It is mailed to about 1300 nuclear professionals in industry, government, and academia. A registration form for inclusion in the mailing list for ALARA Notes is in the appendix.

ACKNOWLEDGMENTS

We thank all project managers and principal investigators for providing information on their projects. We acknowledge the contribution of the people who have discussed with us related research activities and health physics programs. We thank Alan Roecklein, the NRC project manager, for his support of this work. We would like to acknowledge the advice and support of John Baum of the Brookhaven National Laboratory. We also express our thanks to Maria Beckman, our secretary, for her help in producing this report. Finally, we thank the members of the ALARA Center's industry advisory committee for reviewing this document and providing their valuable comments. The members of the advisory committee are:

C. Bergmann	Westinghouse Electric Corporation
R. Giordano	General Electric
C. Hinson	U.S. Nuclear Regulatory Commission
G. Hudson	Tennessee Valley Authority
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F. Roddy	Bechtel Power Corporation
M. Rossler	Edison Electric Institute
J. Schmitt	Nuclear Management & Resources Council
J. Sears	Niagara Mohawk Power Corporation

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SWEDEN/U.S.A. R-333

BWR UNDERWATER DISASSEMBLY/ASSEMBLY - WETLIFT 2000

Keywords: CONTAMINATION PREVENTION; RADIATION SHIELDING; REMOTE SYSTEMS; EXPOSURE REDUCTION

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Objectives: Primary objectives are: to conduct BWR disassembly and assembly operations under water; to reduce contamination and exposure; to speed operations by using remote tooling technology; to improve personnel safety; and to reduce cavity and equipment pool entries.

Other objectives are to reduce critical path time; to reduce exposure; to improve personnel safety; to reduce person-hours; to ease decontamination; and to accomplish this with no physical modifications to existing plant equipment.

Comments: The WETLIFT 2000 system has been used at GPU's Oyster Creek Nuclear Generating Station and allows:

- Reactor cavity flooding immediately after head removal.
- Steam dryer and separator transfer underwater with only one reactor cavity flooding.
- Latching and unlatching of the dryer and separator from refueling bridge.
- Steam plug insertion and withdrawal from refueling bridge.
- Watertight seal of equipment pool shield plugs during reactor vessel maintenance.

WETLIFT 2000 System consists of four major component parts: dryer/separator sling, rigid pole-handli⁻g system, steam line plug tooling, and watertight gate system.

Except for the watertight seal, the WETLIFT 2000 system has also been used at CECO's LaSalle County Station.

Remarks/Potential for dose limitation: The principal benefits of the system are:

- Critical Path Reduction
- One cavity flooding for reactor disassembly and assembly
- All operations performed from refueling bridge
- Remote separator, dryer and shield plug hookup and disconnect
- Accurate positioning, insertion & withdrawal of steam line plugs

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- Remote latching and unlatching of dryer hold downs & shroud head bolts
- No shield plug leakage cleanup Exposure Reduction
- Cavity flooding immediately after head removal
- No cavity work for dryer, separator and steam plug manipulations
- Dryer and separator remain underwater
- Equipment designed for quick decontamination
- Latching and unlatching of dryer hold down and shroud head bolts from refueling bridge
- Steam plug insertion and withdrawal from the refueling bridge
- Significant reduction in respirator and PC work
- Reduced cavity and storage pool entries
- Hook and block stay DRV and contamination-free.

References:

Duration: from: 1990 to: 1992

Funding: N/A

Status: Complete

Last Update: May 27, 1992

U.S.A. R-334

PWR PRIMARY SYSTEM CHEMISTRY: EXPERIENCE WITH ELEVATED PH AT MILLSTONE POINT UNIT 3

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; RADIATION PROTECTION; REACTOR MATERIAL; WATER CHEMISTRY; FUEL

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Objectives: Earlier EPRI measurements of the solubility of simulated fuel crud, together with modeling of crud transport phenomena, led to the conclusion that a pH of 7.4 was superior to a pH of 6.9 for minimizing buildup of radiation fields. Operation of the Ringhals PWRs in Sweden and an earlier cycle of operation at Millstone-3 showed the benefit of operating at higher pH. However, additional data were deemed necessary to determine the effects of long-term operation at higher lithium hydroxide concentrations on the performance of plant components, especially Zircaloy-clad fuel rods.

The objectives are

1. To measure dose rates after operating for one fuel cycle to a terminal pH of 7.2 (following operation for one fuel cycle to a terminal pH of 7.4) and compare these values with other pertinent dose-rate data, and

2. To measure Zircaloy cladding oxide thickness and compare it with similar data obtained from the North Anna PWR.

Comments: Plant chemistry staff maintained the lithium concentration at 3.35 ± 0.15 ppm until the pH reached 7.2; the pH was then held constant until the end of the cycle. The project team then used visual examinations and an eddy current nondestructive evaluation technique to assess oxide buildup on selected fuel rods. Other tasks included dose-rate measurement at standard piping and steam generator locations as well as gamma spectrometry to determine radionuclide concentrations. Dose-rate measurement trends were analyzed using the contractor's CORA code. The beneficial effects of elevated pH operation on radiation fields at Millstone-3 persisted for a second cycle, even though the pH was held at 7.2, rather than the 7.4 value used in the previous cycle. This change halved the time that the unit operated at 3.35 ppm lithium. Operation with elevated lithium appeared to increase Zircaloy cladding corrosion, but the effects are impossible to quantify, given the extensive scatter in the data. Because of the anticipated length of the next fuel cycle, the utility decided to continue operating at a constant pH of 6.9 to minimize the exposure time with lithium concentrations greater than 2.2 ppm.

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Remarks/Potential for dose limitation: The use of elevated pH for a second cycle of operation of the Millstone Point unit 3 PWR continued to reduce radiation-field buildup rates significantly. Dose rates were about 15% lower than in plants operating under coordinated chemistry.

Comparison of the Millstone-3 oxide thickness data and the North Anna high-burnup oxide thickness data suggests that operation with elevated lithium may increase the corrosion rate of Zircaloy cladding. However, the large variation in oxide thickness measurements prevents one from drawing any firm conclusions. Higher pH operation had no observable effect on any other fuel-assembly components. Observed and CORA-calculated dose rates measured here and at the Swedish Ringhals-3 PWR are consistent. Component dose rates are about 15% lower for elevated coolant pH operation compared with operation under coordinated chemistry at a pH of 6.9.

References:

1. EPRI NP-7077, PWR Primary Water Chemistry Guidelines, Revision 2.
2. EPRI TR-100960, PWR Primary System Chemistry: Experience with Elevated pH at Millstone Point Unit 3, Progress Report Number 2, July 1992, 140 pages.

Duration: from: 1990 to: 1994

Funding: N/A

Status: Continuing

Last Update: October 28, 1992

U.S.A. R-335

STEAM GENERATOR DOSE RATES AT BABCOCK & WILCOX REACTORS

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; RADIATION MONITORING; RADIATION DOSE

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Objectives: EPRI has long supported radiation field measurements at operating reactors designed by the different nuclear steam supply vendors. These data are useful in determining trends at an individual plant, in providing data for interplant comparisons, and in evaluating the effectiveness of various radiation control measures.

The objectives are

1. To collect and compare dose-rate data from seven Babcock and Wilcox reactors at five utilities, and

2. To use the data to analyze the effectiveness of various dose-reduction measures implemented at different units.

Comments: Investigators gathered dose-rate data from five sites in mid-1991. The team collected measurements from surveys performed by utility or contractor personnel during plant outages. Researchers then correlated these data with plant design features, operating features, and dose-reduction measures used at each plant. Specific areas addressed in the study included cobalt reduction programs, the cobalt content of steam generator tubing, the use of Zircaloy grid spacers in reload fuel, the number of crud bursts, hydrogen peroxide flushing, primary coolant chemistry, decontamination, and the use of microfiltration.

Remarks/Potential for dose limitation: At the seven units reviewed, upper channel head contact dose rates ranged between 6 and 21 R/hr, while lower channel head contact dose rates ranged between 4 and 20 R/hr. The two most effective techniques in reducing dose rates at these units have been the use of Zircaloy grid spacers in reload fuel and the use of elevated Li/pH primary coolant chemistry. This information should prove valuable to utility personnel assigned the responsibility for reducing occupational radiation exposure of maintenance personnel.

References: EPRI TR-100348, Final Report, July 1992, 64 pages.

Duration: from: 1990 to: 1994

Funding: N/A

Status: Continuing

Last Update: October 28, 1992

PRECONDITIONING OF PWR STEAM GENERATORS TO REDUCE RADIATION BUILDUP

Keywords: CONTAMINATION PREVENTION; STEAM GENERATOR; RADIATION BUILDUP; REACTOR MATERIAL; INCONEL

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Objectives: Examine potential passivation (preconditioning) techniques to determine which form the most stable oxides on alloys used in PWR replacement steam generators, thus reducing corrosion-product rates to the primary coolant.

Comments: Inconel 600 and Inconel 690, the most common steam generator tubing alloys, were chosen for the study along with type 316 stainless steel. Samples were preoxidized under a variety of conditions thought to form adherent, stable oxides. They were then activated and placed in an experimental water-circulating loop. The test facility simulates PWR primary conditions within well-controlled chemistry limits. On-line instrumentation, filters, and resin (ion-exchange) columns collected corrosion products or detected their movement as a function of running time and other key parameters.

Study results were disappointing. At the start of the tests, corrosion release rates differed for samples preoxidized under different conditions. But after 2,000 hours, release rates were reduced, and there were no discernible differences among samples. Preoxidized specimens released more corrosion products than those tested in the unpassivated condition, indicating that reducing conditions should be used. In all cases with Inconel 690, release rates were lowest at pH 7.2-7.4

Remarks/Potential for dose limitation: This study relates only to release of corrosion products, and its results apply only to PWR chemistry. Studies of the effects of preoxidation on deposition of activated corrosion products are described in EPRI reports NP-6616 and TR-100059. Prefilming replacement components for BWRs is more beneficial because deposition is reduced under oxidizing chemistry.

The results do support experience in other countries showing that for PWR startups -- which include hot functional tests and the presence of normal reactor coolant chemicals (LiOH and boric acid) -- the coolant must be deaerated and hydrogen added as early as possible.

References: EPRI TR-100217, Final Report, August 1992, 60 pages.

Duration: from: 1991 to: 1992

Funding: N/A

Status: Completed

Last Update: October 28, 1992

WELDING OF NOREM IRON-BASE HARDFACING ALLOY WIRE PRODUCTS - PROCEDURES FOR GAS TUNGSTEN ARC WELDING

Keywords: COMPONENT RELIABILITY; IRON-BASE ALLOY; COBALT; NOREM; HARDFACING ALLOY; VALVE

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Objectives: Cobalt-base hardfacing alloys used in valves and other critical components in nuclear reactors are a significant contributor to radiation field buildup and to the occupational radiation exposure of plant maintenance personnel. Laboratory studies, described in EPRI report NP-6466-SD, show that NOREM iron-base alloys exhibit wear properties matching those of cobalt-base alloys. However, wire products and welding procedures were needed for field applications.

The objectives of this project are:

- To secure metal core and homogenous weld wire of NOREM alloys suitable for deposition by automatic gas tungsten arc welding (GTAW).
- To develop welding procedures for these wire products.
- To measure the galling wear resistance of specimens prepared using EPRI's newly developed welding procedures.

Comments: Researchers obtained metal-core and homogeneous NOREM weld wire from a number of sources that relied on standard commercial practices in fabricating the product. They next used GTAW to deposit the wire on carbon and stainless steel plate and piping as well as on residual strips of cobalt-base alloy. Finally, they deposited the wire on pins and plates specially made for galling wear tests.

Researchers successfully used GTAW to deposit sound weld overlays of NOREM wire products on Type 304 stainless steel and SA-515 carbon steel substrates without any preheating. The NOREM alloy can also be deposited over a continuous or intermittent layer of cobalt-base Stellite 21 previously deposited by plasma arc welding (PTAW). Wire products with lower Mn and Si contents than used in gas atomized powder exhibited the best results. Wire cleanliness also proved a critical factor in achieving sound welds. In wear testing, NOREM PTAW overlays were as resistant to galling as specimens prepared using Stellite 21.

Remarks/Potential for dose limitation: Laboratory studies reported in EPRI report NP-6466-SD showed that the wear resistance of NOREM iron-base alloys matched that of cobalt-base alloys and possessed other properties required of hardfacing alloys. PTAW of gas atomized powder was used to successfully deposit these alloys on 3-inch valves, which were then subjected to long-term endurance tests under simulated PWR chemistry conditions. Extensive nondestructive and destructive evaluations showed that these valves performed better than one with Stellite 6 trim (report TR-100601). Similar valve tests are now being completed under BWR chemistry conditions. The need for field hardfacing replacements and repairs led to this NOREM wire development and welding evaluation program for in-situ applications. The study demonstrated that the NOREM alloy can be fabricated using standard hardfacing wire production practices and deposited successfully on stainless and carbon steel substrates by automatic GTAW. The NOREM alloy is weldable and exhibits wear and resistance equivalent to or better than cobalt-base alloys. Overall, NOREM alloys should be considered for field applications in both nuclear and fossil plant valves.

References:

1. EPRI NP-6466-SD.
2. EPRI Report TR-101094s
3. EPRI Report TR-100601

Duration: from: 1990 to: 1994

Funding: N/A

Status: Continuing

Last Update: November 17, 1992

ENDURANCE TESTS OF VALVES WITH COBALT-FREE HARDFACING ALLOYS

Keywords: CONTAMINATION PREVENTION; COBALT; IRON; VALVE; HARDFACING ALLOY; DOSE REDUCTION; RADIATION BUILDUP; ALLOY

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Objectives: To develop weld procedures for the candidate alloys on 3-in. gate valves. To determine if the iron-based hardfacing alloys have wear properties matching those of cobalt-based alloys under simulated BWR conditions.

Comments: The performance of the alloys was assessed as follows:

- A number of diverse techniques were employed, including metallography, chemical analysis, hardness, and dye penetration tests
- The iron-based alloys used were NOREM, EB 5183, and EVERIT 50.
- The cobalt-based alloy Stellite 6 was used as a standard.
- The valves were operated under simulated normal BWR primary coolant chemistry conditions (200 ppb oxygen) for 970 full-stroke cycles.

After the first 500 cycles, the valves were characterized by using nondestructive examination techniques: visual examinations, leak rate measurements, and profilometry. After 970 cycles, these examinations were complemented by detailed metallurgical characterization of the valve trim.

Remarks/Potential for dose limitation: The results indicate that with the exception of EB 5183, which is susceptible to pitting attacks, the corrosion resistance of the iron-based alloys is equal or superior to that of Stellite 6. Cobalt-based hardfacing alloys are used in valves and other critical components in nuclear reactors. They are a significant contributor to radiation-field buildup and occupational exposure to plant maintenance personnel. Using cobalt-free alloys to replace or refurbish valves will significantly reduce released cobalt.

References: "Endurance Testing of Valves With Cobalt-Free Hardfacing Alloys," EPRI-TR-101847, Final Report, January 1993.

Duration: from: 1991 to: 1993

Funding: N/A

Status: Completed

Last Update: May 28, 1993

REPLACEMENT OF PINS AND ROLLERS IN IRRADIATED BWR CONTROL BLADES

Keywords: CONTAMINATION REMOVAL; CONTAMINATION PREVENTION; REMOTE SYSTEM; COBALT; CONTROL BLADE; RADIATION FIELD

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Objectives: To design, fabricate, and demonstrate remotely operated equipment that could be used in the spent-fuel pool at BWR sites and would remove the upper pins and rollers in irradiated control blades, replacing them with stainless steel buttons.

Comments: The equipment uses electrical discharge machining (EDM) to remove the upper roller, most of the pin, and a small portion of the control blade. All cut material is collected in a waste container, and fine dust from the EDM operation is collected on a filter. The cut surface is brushed, finished, and inspected. Two stainless steel button halves are remotely riveted together, and their installation is checked with a torque test. The equipment was designed and tested on unirradiated control blades. Modifications were made to accommodate various control blade designs, and the system was then tested at Commonwealth Edison's LaSalle Station.

Remarks/Potential for dose limitation: The equipment was successfully set up and demonstrated at the LaSalle Station. However, work was limited to one wing of one blade due to high dose readings on the filters used to collect radioactive cobalt dust from the EDM operation. These readings would not present a problem if the filters were placed in high-integrity containers. The cobalt-based alloys used in the pins and rollers of BWR control blades are a significant source of the released cobalt that contributes to occupational radiation exposure. Remotely operated equipment permits replacement of the radioactive upper pins and rollers in control blades in the spent-fuel pool.

References: "Replacement of Pins and Rollers in Irradiated BWR Control Blades," EPRI-TR-101837 Vol. 1, Final Report, February 1993.

Duration: from: 1991 to: 1993

Funding: N/A

Status: Completed

Last Update: September 3, 1993

SECONDARY HYDRIDING OF DEFECTED ZIRCALOY-CLAD FUEL RODS

Keywords: COMPONENT RELIABILITY; CONTAMINATION PREVENTION; HYDROGEN; ZIRCALOY; CORROSION; FUEL ROD CLADDING

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Objectives: To examine the secondary hydriding of Zircaloy cladding in a breached LWR fuel rod and evaluate its role in the fuel rod degradation process.

Comments: This study began with a literature review to summarize the current status and understanding of processes pertinent to breached cladding. Analysis of these processes led to initial governing equations, which when solved, enable investigators to determine the location at which secondary hydriding can occur in a fuel rod.

- The source of steam in secondary hydriding is the flashing of coolant through the primary defect. However, the fuel pellets are unlikely to be oxidized by the steam at prevailing temperatures.
- Fission fragments can generate substantial quantities of reactive species such as hydrogen peroxide, which are quite capable of oxidizing the fuel pellets. Such oxidation is a potentially larger source of hydrogen that can cause secondary hydriding in the cladding.

Remarks/Potential for dose limitation: It was shown that a primary defect in the Zircaloy cladding can lead to conditions that may cause secondary hydriding of the cladding. The cladding may subsequently become hydrogen-embrittled, or local massive hydride regions may form. In either case, a rupture in the cladding is possible, often far removed from the primary defect location. Thus, a small through-wall breach in the Zircaloy cladding of an LWR fuel rod can lead to a rapid degradation of the fuel rod. This causes a large quantity of radioactivity to be released into the primary coolant system.

References: "Secondary Hydriding of Defected Zircaloy-Clad Fuel Rods," EPRI TR-101773, Final Report, January 1993.

Duration: from: 19 to: 1993

Funding: N/A

Status: Completed

Last Update: May 28, 1993

TESTING OF AN ORGANIC REMOVAL PROCESS IN BWR RADWASTE SYSTEMS

Keywords: CONTAMINATION PREVENTION; CORROSION; ORGANIC SUBSTANCES; WATER QUALITY; OZONE; ULTRAVIOLET RADIATION; RADWASTE SYSTEM

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Objectives: Naturally occurring organic plus lubricating oils and cleaning solvents are present in plant water systems. These compounds are not removed by the usual plant purification systems. A process based on ozone-ultraviolet radiation has been demonstrated to be capable of degrading and removing organics. The objectives of the study are:

- To optimize the ozone-ultraviolet treatment system parameters.
- To evaluate the effect of residual ozone on ion-exchange resins.
- To determine the reaction time required to effectively treat organic transients.
- To evaluate the efficiency of the process for removal of organics in both high- and low-purity wastes.
- To quantify the expected reduction in radwaste ion-exchange resin usage resulting from reduced organic concentration.

Comments:

- Pilot-scale tests were carried out at Vallecitos Nuclear Center followed by tests on radwaste systems at Dresden and Susquehanna.
- Process streams containing common organic contaminants were introduced into a vessel outfitted with ultraviolet lamps.
- Oxygen and ozone were fed continuously into the vessel.
- Oxidation of the organic was monitored as a function of time, and downstream ozone concentration was also measured.

The treatment system parameters were optimized as follows:

- 100 ppm ozone in combination with ultraviolet light oxidized 90% of the organic in 15-30 min.
- Ion-exchange resins helped remove the oxidized ions of organic acids, nitrates, and sulfates.
- In simulated organic transients or spills, 300 ppm ozone oxidized >90% of organic compounds even in turbid waters in times of up to 60 min.

- Electrohydraulic control fuel was efficiently removed, nonreactive silica was converted to a removable, reactive form.
- Essentially no ozone was detected in the effluent of the process, and there was no damage to the downstream resins.

Remarks/Potential for dose limitation: In reactor water, thermal decomposition and oxidation of organics may yield corrosive species such as acids. The development of a method for their removal reduces the danger to components and relieves the burden on plant purification systems. The operating savings in exposure and funds from this process are sufficient for utilities to consider a full-scale installation. At Susquehanna, the ozone treatment would reduce the liquid waste discharge by 50% and the resin usage by 65%, thereby decreasing operating costs.

References: "Testing of an Organic Removal Process in BWR Radwaste Systems," EPRI NP-7195, Final Report, February 1993.

Duration: from: 1988 to: 1991

Funding: N/A

Status: Completed

Last Update: September 3, 1993

U.S.A. R-342

EVALUATION OF REACTOR PRESSURE VESSEL HEAD CRACKING IN TWO DOMESTIC BWRS

Keywords: COMPONENT RELIABILITY; REACTOR PRESSURE VESSEL; IGSCC; STAINLESS STEEL; LOW-ALLOY STEEL

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Objectives:

1) Determine the cause of cracking in the reactor pressure vessel (RPV) top head at two domestic BWRs
2) use structural analysis methods for assessing the consequences of cracking on continued top head operation

Comments: Two BWRs, Quad Cities unit 1 and Vermont Yankee, experienced cracking in the stainless steel cladding of the RPV top head. The following tests were done at Quad Cities:

- surface penetrant testing (PT)
- ultrasonic testing (UT)
- metallurgical failure analyses
- finite-element analyses

At Vermont Yankee, only visual analysis, PT, and UT were performed. In both cases, structural analyses were done to determine the consequences of cracking on the structural reliability of the RPV head.

Remarks/Potential for dose limitation: The cracking in the austenitic stainless steel cladding was a result of IGSCC. This was caused by the oxidizing nature of the BWR environment, which can be particularly severe in the top head region. The study also showed that cracking in low-alloy steel was not associated with the coolant environment, but rather it resulted from a reheat cracking mechanism that can affect certain low-alloy steels. Structural analyses demonstrated that both plants had considerable structural margin, meeting ASME Code Section XI requirements for continued operation without repair. The vessel head region can be made less susceptible to IGSCC by improved water chemistry controls, and by using hydrogen water chemistry wherever practical.

References: "Evaluation of Reactor Pressure Vessel Head Cracking in Two Domestic BWRs" EPRI TR-101971, Final Report, February 1993.

Duration: from: 1992 to: 1993

Funding: N/A

Status: Completed

Last Update: September 3, 1993

RELATIONSHIP OF RADIATION-INDUCED SEGREGATION PHENOMENA TO IRRADIATION-ASSISTED STRESS CORROSION CRACKING (IASCC)

Keywords: COMPONENT RELIABILITY; IASCC; STAINLESS STEEL; ELECTRON MICROSCOPY; NICKEL ALLOY

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Objectives: The aim of this study was to enhance the understanding of IASCC in austenitic stainless steels and nickel base alloys by examining microstructural changes and grain boundary segregation as a function of irradiation at LWR temperatures.

Comments: The following alloys were studied:
1) Type 304 stainless steel
2) Type 316 stainless steel
3) high-purity Type 304 stainless steel
4) commercial purity alloy X-750

The techniques used for the microstructural examination and the grain segregation studies were scanning transmission electron microscopy (STEM) and auger electron spectroscopy (AES). Each of the four alloys were examined after being irradiated, and nonirradiated specimens served as controls.

Remarks/Potential for dose limitation: The investigations demonstrated that exposure to high neutron fluences at LWR temperatures produces microstructural and grain boundary changes in austenitic stainless steels and nickel base alloys. The STEM studies revealed black spot damage in the Type 316 stainless steel specimens at both high and low fluences. Such damage is responsible for the large increase in yield stress observed in steels irradiated at low temperatures. The AES analysis revealed evidence of chromium depletion at the grain boundaries of the high-purity and commercial-purity Type 304 stainless steel specimens as well as the alloy X-750 specimen. Nickel enrichment occurred in both the high-purity Type 304 and alloy X-750 samples. Some evidence of phosphorus and sulfur segregation was visible in the grain boundaries of the commercial-purity specimens.

References: "Relationship of Radiation-Induced Segregation Phenomena to Irradiation-Assisted Stress Corrosion Cracking (IASCC)," EPRI TR-101987, Final Report, February 1993.

Duration: from: 19 to: 1993

Funding: N/A

Status: Completed

Last Update: June 4, 1993

NORWAY R-344

LIGHT WATER REACTOR MATERIALS AND WATER CHEMISTRY STUDIES AT HALDEN

Keywords: COMPONENT RELIABILITY; WATER CHEMISTRY; PRIMARY ANT CHEMISTRY; LITHIUM; IASCC; CORROSION

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Objectives: The objective of the PWR Test Facility experiments is to determine the effects of high lithium concentrations on the corrosion behavior of Zircaloy-4. The objective of the IASCC facility studies is to assess the effect of water chemistry environment on the cracking propensity of in-core structural materials commonly found in BWRs.

Comments: The Halden Reactor Project test facilities are designed to produce the radiation, thermal hydraulic, and water chemistry conditions representative of those found in commercial Li--t Water Reactors. The PWR facility is being used to determine the effects of high lithium c... tration on the corrosion behavior of high burnup Zircaloy-4 fuel rods. 4-4.5 ppm lithium a... .000 ppm boron were added to the water. The pH was 7.2-7.3 at 300°C, and after 250 full power days, the boron concentration was reduced to 700 ppm (pH 7.4). Oxide thickness was measured in order to determine the effect of lithium concentrations on corrosion rates. Early results indicate that increased lithium concentrations have little effect on enhancing corrosion rates. The IASCC loop system is designed to operate under both Normal BWR Water Chemistry (NWC) and Hydrogen Water Chemistry (HWC) conditions. In order to determine the benefits of HWC in mitigating crack propagation, the specimens are exposed to NWC with the introduction of HWC at various stages during the course of irradiation. Corrosion potential and solution conductivity are closely monitored.

Remarks/Potential for dose limitation: It is hoped that future investigations at Halden will address the feasibility of applying chemical additives such as zinc to control radiation buildup in PWRs. Further investigations relating to IASCC behavior of structural component materials are also expected.

References: Karlsen, T., Gunnerud, P., and Vitanza, C., "Light Water Reactor Materials and Water Chemistry Studies at Halden," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, British Nuclear Energy Society, London, 1992.

Duration: from: 1991 to: 1992

Funding: N/A

Status: In Progress

Last Update: June 4, 1993

PWR IN-PILE LOOP STUDIES IN SUPPORT OF COOLANT CHEMISTRY OPTIMIZATION

Keywords: COMPONENT RELIABILITY; CONTAMINATION PREVENTION; PRIMARY COOLANT CHEMISTRY; PH; CORROSION CONTROL; RADIATION BUILDUP

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Objectives: The aim of this study is to investigate the effects of primary coolant pH on corrosion product oxide mass and radionuclide inventories on loop component surfaces.

Comments: A series of three 3000 hour in-pile loop runs are in progress using the MIT PWR Coolant Chemistry Loop (PCCL) to verify the selection of pH for use in PWR coolant. The following measurements and examinations are made:

- corrosion product oxide mass
- radionuclide inventories
- surface SEM examination before and after crud removal
- waterborne radionuclide concentration
- crud filter assays pH is optimized through adjustment of relative concentrations if LiOH and H_3BO_3

Remarks/Potential for dose limitation: Essentially, all the work to date supports operation at about pH(300°C) = 7.3. Future tests will study the efficacy of zinc injection for reduction of corrosion product radionuclide buildup.

References: Kohse, G.E., Cabello, E.C., Doboe, L., Driscoll, M.J., and Harling, O.K., "PWR In-Pile Loop Studies in Support of Coolant Chemistry Optimization," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, British Nuclear Energy Society, London, 1992.

Duration: from: 19 to: 1994

Funding: N/A

Status: In Progress

Last Update: June 4, 1993

CANADA R-346

THE EFFECT OF DISSOLVED OXYGEN IN LITHIATED COOLANT

Keywords: CONTAMINATION PREVENTION; RADIATION BUILDUP; OXIDE FILM; OXYGEN CONTENT; PRIMARY COOLANT CHEMISTRY

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Objectives: The aim of this study was to determine the effect of slightly oxidizing conditions on cobalt-60 activity buildup on 403 stainless steel, carbon steel, and iron oxide pellets. It has been hypothesized that oxidizing conditions lead to a higher rate of field growth.

Comments: Experiments were performed in a high-temperature experimental loop constructed of 304 stainless steel. The measurements and examinations done included:

- average Co-60 activity in the water
- Co-60 pickup by oxide pellets and steel surfaces
- scanning electron microscopy (SEM)
- conversion electron Mössbauer microscopy
- transmission electron microscopy (TEM)
- energy dispersive X-ray (EDX)
- scanning auger microprobe (SAM)

Remarks/Potential for dose limitation: The conclusions of the study were as follows:

- Oxide films formed on steel materials in deoxygenated and slightly oxygenated water were seen to have different structures and different affinities for Co-60.
- Oxide films formed under oxidizing conditions were thinner, but had higher Co-60 activities per unit volume of oxide.

Because of these two properties working against each other, the net result was similar Co-60 activity per unit area of base metal under reducing and oxidizing conditions.

References: Allsop, H.A., Sawicki, J.A., Lister, D.H., and Godin, M.S.L., "The Effect Of Dissolved Oxygen in Lithiated Coolant," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, pp. 25-32, British Nuclear Energy Society, London, 1992.

Duration: from: 19 to: 1992

Funding: N/A

Status: Completed

Last Update: September 3, 1993

CHEMISTRY PARAMETERS INFLUENCING THE DOSE RATE BUILD-UP IN BWR PLANTS

Keywords: CONTAMINATION PREVENTION; RADIATION BUILDUP; PRIMARY COOLANT CHEMISTRY; PH; OXYGEN CONTENT; NICKEL; IRON; STRIPPING VOLTAMMETRY; ION CHROMATOGRAPHY

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Objectives: The purpose of this paper is to discuss several parameters that are known for influencing the dose rate buildup in BWRs. These factors are related to primary coolant chemistry with certain aspects related to the steam water cycle, including:

- zinc chemistry
- Ni/Fe ratio
- oxygen concentration
- pH

Comments:

- Traces of zinc in the reactor coolant reduce dose rate levels. Zn addition requires a product depleted in Zn-64 to avoid an increased formation of the activation product Zn-65.
- Co-58 and Co-60 activities on the surface of piping are reduced by the influence of the Ni/Fe ratio. However, when increasing the iron concentration too much, the tendency of mobilizing undissolved and activated corrosion products will increase. This is caused by the increased crud deposition on the surfaces of the fuel elements.
- When the oxygen content is too low in medium of the steam water cycle in areas with low alloyed steels, then erosion corrosion of the materials can be increased. This increases the iron input into the RPV, and thus increases the tendency to higher dose rate.
- Increasing oxygen concentration significantly increases material release from cobalt base alloys.
- The use of cobalt reduced materials as hardfacing alloys can contribute highly to the minimization of the dose rate buildup.
- Increasing the pH decreases the metal release rates and mitigates the solubility of corrosion products. Increasing the pH too far causes the metal release rates and solubilities of corrosion products to increase again. Long term pH modification in BWRs is an option for the future.

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Remarks/Potential for dose limitation:

- Filtration, stripping voltametry, and ion chromatography are analytical methods practicable in the necessary low concentration range for monitoring dominating parameters for dose rate buildup and the effect of appropriate countermeasures.
- In general, it seems to be possible to influence the dose rate buildup in a BWR by changing chemical or operational parameters. However, the measures to be taken depend on the specific conditions of the plants and should therefore be determined individually for each plant. It should be considered that the above-mentioned parameters show long term effects, and that measurable results of the dose rate development of the piping will not appear immediately, but the desired effects will be detectable after one or two cycles.

References: Marchl, T.F. and Reitzner, U., "Chemistry Parameters Influencing the Dose Rate Build-Up in BWR Plants," *Water Chemistry of Nuclear Reactor Systems 6*, British Nuclear Energy Society, London, 1992.

Duration: from: 1985 to: 1992

Funding: N/A

Status: Completed

Last Update: September 3, 1993

OVERVIEW OF ACTIVITIES FOR THE REDUCTION OF DOSE RATES IN SWISS BOILING WATER REACTORS

Keywords: CONTAMINATION PREVENTION; PRIMARY COOLANT CHEMISTRY; ZINC ADDITION; COBALT

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Objectives: Two Swiss BWRs, at Leibstadt (KKL) and Mühleberg (KKM) began to add 0.4 ppb Zn and 0.65 ppb Fe-III respectively to their feedwater. The aim of this study was to research the following three themes: 1) Statistical analysis of KKL reactor water data, 2) KKL reactor water analysis during the annual shutdown, 3) autoclave tests to clarify the role of water additives on the Co-60 deposition on steel surfaces.

Comments: Statistical water analysis showed that zinc has a moderately reducing effect of the Co-60 activity in the reactor water. Without Zn present, Fe has a strong reducing effect, with Zn, a strong increasing effect. Cr, Ni also changed sign: without Zn they have a moderate increasing effect, with Zn, a strong reducing effect.

The reactor water analysis during the annual shutdown in 1991 showed that at 24% power, the total Zn-65 activity increases by a factor of 8, mainly because of dissolved Zn-65. At 0% power, full control rod insertion, the crud particle concentration >1μm size increases by a factor of 50. At 160°C, 10 bar, an increase in the total Co-60 activity by a factor of 12 is measured.

The laboratory autoclave tests were made with different water additives and austenitic steel samples at simulated BWR conditions. After 6 time periods each of 300 h, the Co-58 buildup (%) on steel samples exposed to different water additives (ppb) was measured. The results were as follows:

20 ppb Zn	>	Reference	>	20 ppb Fe	>	1.5 ppb Zn
107 to 135%		100%		64 to 77%		47 to 69%

Remarks/Potential for dose limitation: In terms of the statistical water analysis, other BWR data sets have to be analyzed in order to determine whether these KKL-specific statements are of general significance. In terms of shutdown observations, it is not yet clear how the plant cooldown procedure should be modified to take these observations into account.

References: Alder, H.P, and Brélaz, P., "Overview Of Activities For the Reduction Of Dose Rates in Swiss Boiling Water Reactors," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, pp. 45-50, British Nuclear Energy Society, London, 1992.

Duration: from: 1990 to: 1992
Status: Completed

Funding: N/A
Last Update: June 8, 1993

OPERATING EXPERIENCE OF JAPANESE IMPROVEMENT AND STANDARDIZATION BWRs AND BEHAVIOR OF RADIOACTIVITY IN REACTOR WATER

Keywords: CONTAMINATION PREVENTION; COBALT; DOSE REDUCTION; PRIMARY COOLANT CHEMISTRY

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Objectives: This paper describes the increasing concentration of radioactivity in reactor water at Japanese BWRs and the results of studies to clarify this phenomenon.

Comments: In new Japanese BWRs, increased Co-60 concentrations are occurring which may cause an increase in plant dose rates as operation continues year after year. To suppress such dose rate increases at new plants, the investigators have begun to study the causes of increased Co-60 concentrations. The following possibilities were suggested:

1) dissolution speed of fuel deposits accelerating due to reduced reactivity of Fe and Co
2) dissolution speed of fuel deposits accelerating due to change of oxidized surface conditions of new fuel cladding
3) dissolution speed of fuel deposits increasing due to production of high Cr content deposits by increased feedwater Cr concentration
4) highly corrosive fuel spacers

Remarks/Potential for dose limitation: Measures implemented by new plants:

1) use of corrosion-resistant materials for the turbine system
2) dual condensate purification facility
3) oxygen injection into feedwater system
4) use of low-Co materials in feedwater-heater tubes, fuel-spacer springs, and control rod pins and rollers
5) pre-filming
6) Fe/Ni ratio control

References: Aizawa, M., Ohsumi, K., Asakura, Y., Morikawa, Y., Hirahara, Y., Sakai, T., and Haraguchi, K., "Operating Experience of Japanese Improvement and Standardization BWRs and Behavior of Radioactivity in Reactor Water," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, pp. 39-44, British Nuclear Energy Society, London, 1992.

Duration: from: 19 to: 1994 **Funding:** N/A

Status: In Progress **Last Update:** June 8, 1993

FEEDWATER IRON CRUD REDUCTION FOR CHINSHAN NUCLEAR POWER STATION

Keywords: CONTAMINATION PREVENTION; IRON; CRUD; RADIATION BUILDUP; FILTRATION; RETUBING

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Objectives: This paper describes the operating history of Chishan Nuclear Power Station and the methods of iron crud identification and reduction used there.

Comments: Chinshan Nuclear Power Station consists of twin 636 megawatt BWRs located about 40 km north of Taipei. They were put into commercial operation in 1978 and 1979. In 1985, aluminum brass condenser tubing was replaced with titanium tubing. The iron crud concentration subsequently rose from less than 1 ppb to higher than 2 ppb. This increase was attributed to the retubing. Crud samples were taken from the condensate pump discharge header, condensate demineralizer influent, common condensate demineralizer effluent, and the feedwater pump outlet. The iron content and other elemental concentration were analyzed using X-Ray diffraction. Crud particle size was determined using scanning electron microscope (SEM) and particle size analyzer.

Remarks/Potential for dose limitation: The investigators concluded that the following improvement measures should be proposed:

- additional prefiltering
- increase the cation to anion volume ratio from 2:3 to 1:1 or 2:1
- decrease the shutdown rate (soft shutdown)
- dry lay up during shutdown period
- recirculation wet lay up
- condenser hot well cleaning
- start up recirculation
- condensate demineralizer performance improvement
- improve backwash procedure
- investigate the height of resin bed
- proper control of dissolved oxygen in feedwater
- material replacement

References: Wen, T.J. et al, "Feedwater Iron Crud Reduction for Chinshan Nuclear Power Station," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, pp. 57-62, British Nuclear Energy Society, London, 1992.

Duration: from: 1991 to: 1992

Funding: N/A

Status: Completed

Last Update: September 3, 1993

REACTIONS OF IRON CRUD WITH METALLIC IONS UNDER BWR WATER CONDITIONS

Keywords: CONTAMINATION PREVENTION; OXIDE FILM; ION

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Objectives: Formation mechanisms and formation rates of $NiFe_2O_4$ and $CoFe_2O_4$ from amorphous Fe(III) hydroxide and α-Fe_2O_3, with Ni(II) and Co(II) ions, were studied experimentally to clarify the formation of spinel oxide on BWR fuel rod surfaces.

Comments: The reactions of the amorphous Fe(III) hydroxide with Ni(II) and Co(II) could be explained by a reaction model incorporating two phenomena: the dehydration of Fe(III) hydroxide, and the diffusion of ions into it. Cobalt (II) ions diffused into α-Fe_2O_3 to form $CoFe_2O_4$. Apparent activation energy for Co(II) diffusion into a α-Fe_2O_3 was obtained as 2.72-e5 J/mol. Formation of $NiFe_2O_4$ from α-Fe_2O_3 and Ni(II) was promoted by crystallization of Ni(II) and Fe(III) ions from dissolved α-Fe_2O_3 and NiO. The apparent activation energy for the nucleation of $NiFe_2O_4$ crystal was obtained as 6.38-e5 J/mol. When $Co(OH)_2$ coexisted with $Ni(OH)_2$ and α-Fe_2O_3, Co_3O_4 which has a spinel structure was formed and those particles became nuclei for $NiFe_2O_4$. This lowered the activation energy for the nucleation.

Remarks/Potential for dose limitation: The behavior of metallic ions (Ni, Co, etc.) and Fe crud (Fe(III) hydroxides and oxides) which enter the primary water by corrosion of structural materials is closely related to radioactivity in the reactor water. Most of these impurities are deposited on the fuel cladding surface in the reactor water where they become activated through neutron irradiation and represent a major source of radioactivity. Boiling on the fuel rod surface accelerates the deposition of Fe crud and ions. It is desirable to suppress the amounts of parent nuclides flowing into the reactor. Then main radioactive species are Co-58 and Co-60, which are produced by the reactions Ni-58(n,p)Co-58 and Co-59(n,γ)Co-60 respectively. The amount of Co can be reduced by using low Co content materials. However, reduction of the Ni amount is difficult due to dissolution from Ni based alloy in the reactor. Then, it is important to change the chemical form of the deposits into spinel oxides ($NiFe_2O_4$, $CoFe_2O_4$, etc.), which have lower solubility than mono-oxides (NiO, CoO, etc.). When spinel oxides form on the fuel rod surface, the release of radioactive species from them is depressed and those concentrations in the reactor water can be kept low.

BNL ALARA Center Data Base

JAPAN R-351

References: Nishino, Y. et al, "Reactions of Iron Crud With Metallic Ions Under BWR Water Conditions," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, pp. 63-68, British Nuclear Energy Society, London, 1992.

Duration: from: 19 to: 1992

Funding: N/A

Status: Completed

Last Update: June 9, 1993

DECOMPOSITION OF HYDROGEN PEROXIDE IN BWR COOLANT CIRCUIT

Keywords: CONTAMINATION PREVENTION; PRIMARY COOLANT CHEMISTRY; HYDROGEN PEROXIDE; CORROSION

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Objectives: In a BWR primary coolant circuit, the coolant flow velocities and volume-to-surface ratios at various locations are taken into account for the estimation of the decomposition rate of hydrogen peroxide in the system. The decomposition half-times are estimated ranging from a few seconds in the core region to a few minutes in the recirculation piping system.

Comments: Although hydrogen peroxide is believed to exist in the coolant at approximately 280°C during power operation, the measurements has not been successful due mostly to surface catalytic effects in the sample line. Thus the actual level of hydrogen peroxide in an operating BWR is still not accurately known. The rate of a heterogeneous catalyzed reaction is dependent upon both mass transfer and chemical activation processes. In a recent laboratory study of hydrogen peroxide decomposition in aqueous solutions, it has been observed that at temperatures lower than ~200°C, the decomposition reaction is mostly activation-controlled, and above ~200°C the mass transfer process becomes an important factor in determining the overall reaction rate.

Remarks/Potential for dose limitation: By combining the above results and the laboratory test results for the activation-controlled rate constants extrapolated to 280°C, the overall rate constants at various locations in a BWR primary coolant circuit have been calculated. The decomposition half-times are estimated ranging from a few seconds in the core region to a few minutes in the recirculation piping system. Hydrogen peroxide is one of the stable products radiolytically produced in BWR coolant. Understanding the chemical properties of hydrogen peroxide in the BWR coolant has become an important factor in dealing with the material corrosion problems in the BWR primary system.

References: Lin, C.C., "Decomposition of Hydrogen Peroxide in BWR Coolant Circuit," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, pp. 85-88, British Nuclear Energy Society, London, 1992.

Duration: from: 1989 to: 1992

Funding: N/A

Status: Completed

Last Update: September 3, 1993

FULL PRIMARY SYSTEM CHEMICAL DECONTAMINATION QUALIFICATION PROGRAM

Keywords: CONTAMINATION REMOVAL; CAN-DEREM; LOMI; DECONTAMINATION; FULL SYSTEM DECONTAMINATION

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Objectives: Determine the technical acceptability of using certain dilute chemical solvent processes for full reactor coolant system (RCS) chemical decontamination. Two processes, CAN-DEREM and LOMI, were selected as candidates to be qualified for use in a PWR.

Comments: The study of the two decontamination methods was divided into seven tasks:

1. Process Qualification Test Program
2. Fluid Systems Evaluation of Decontamination Process Integration With RCS and Auxiliary Systems
3. Engineering Evaluation of RCS Components and Systems
4. Waste Management Methodology and Waste Characteristics
5. Evaluation of Long-Term Benefit of Full RCS Decontamination
6. Preparation of Topical Report and Generic Safety Evaluation
7. Full RCS Decontamination Project Conceptual Design

Additionally, a detailed review was made of previous evaluations and laboratory assessments relevant to the CAN-DEREM and LOMI Decontamination Process in order to identify potential corrosion consequences following a full RCS chemical decontamination.

Remarks/Potential for dose limitation: The only economically feasible way of significantly reducing the source term of a PWR is to chemically decontaminate the entire primary system. As a result of the evaluations performed, it has been demonstrated that full RCS chemical decontamination, using either the CAN-DEREM or LOMI process, can be performed with a high degree of confidence without significant impacts on plant equipment.

References: Miller, P.E., "Full Primary System Chemical Decontamination Qualification Program," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, pp. 89-96, British Nuclear Energy Society, London, 1992.

Duration: from: 1988 to: 1992

Funding: N/A

Status: Completed

Last Update: June 9, 1993

FULL SYSTEM DECONTAMINATION AND COUNTERMEASURES AGAINST RECONTAMINATION OF THE FUGEN NUCLEAR POWER STATION

Keywords: CONTAMINATION REMOVAL; CONTAMINATION PREVENTION; DOSE REDUCTION; DECONTAMINATION; FULL SYSTEM DECONTAMINATION

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Objectives: This study describes the full system decontamination experiences and effects of endeavors against recontamination at Fugen Nuclear Power Station in Japan.

Comments: The system decontamination procedure consisted of four processes:

1. heating - the temperature of the reactor coolant was raised to 120°C after nitrogen had been injected to deoxidize the coolant
2. decontamination - the decontamination reagent Kuridecon-203 (KD-203) was added and circulated for 24 hours
3. purification - the decontaminate was purified to remove the reagent and radionuclides; conductivity of the coolant was reduced to 10µS/cm
4. purification and flushing - drain and bent tubes were flushed out or rinsed with pure water, and the primary coolant was purified completely until its conductivity was less than 1µS/cm; all the coolant was then discharged and refilled, and the purification, flushing and draining procedures were repeated.

Remarks/Potential for dose limitation: The two decontaminations in 1989 and 1991 saved occupational radiation doses of 6.6 and 7.8 man-Sv respectively. The ultrasonic fuel crud cleaning slightly lowered the recontamination rate after the decontamination of 1991 compared to that of 1989 without the cleaning. As a further countermeasure against recontamination, high-efficiency crud removal resins which reduce crud iron concentration in feedwater to less than 1 ppb are under evaluation at Fugen.

References: Naoi, Y. et al, "Full System Chemical Decontamination and Countermeasures Against Recontamination of the Fugen Nuclear Power Station," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, pp. 97-104, British Nuclear Energy Society, London, 1992.

Duration: from: 1989 to: 1992
Status: Completed

Funding: N/A
Last Update: September 3, 1993

ELECTROCHEMICAL CORROSION POTENTIAL MEASUREMENT WITH A ROTATING CYLINDER ELECTRODE IN 288°C WATER

Keywords: COMPONENT RELIABILITY; IGSCC; ELECTROCHEMICAL CORROSION POTENTIAL; ROTATING CYLINDER ELECTRODE

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Objectives: This study focuses on a description of the test apparatus and the effects of water flow velocity on the electrochemical corrosion potential (ECP) behavior of stainless steel in 288°C water under simulated BWR conditions.

Comments: The ECP of 316 stainless steel was measured by using the rotating cylinder electrode under simulated BWR water chemistry conditions. It was demonstrated that the rotating cylinder electrode (RCE) is useful for measuring the ECP under hydrodynamic conditions in 288°C water. Because of the practical limitation and undefined hydrodynamic conditions by the pipe loop and the paddle wheel, the RCE is useful for practical evaluation of the effect of flow velocity on the kinetics of electrochemical processes.

Remarks/Potential for dose limitation: It has been shown that IGSCC susceptibility can be markedly decreased if the ECP can be decreased below a critical value. ECP behavior is known to be controlled by the dissolved H_2, O_2, and H_2O_2 concentrations in the BWR coolant and subsequently mass transfer rates of these species play an important role on ECP. Therefore, an acceleration of the electrochemical reactions of H_2 and O_2 caused by hydrodynamic water flow is expected to alter ECP behavior of stainless steel under various water chemistry conditions. The preliminary data have shown that the increase of the water flow velocity accelerates the oxygen reduction rate under various dissolved oxygen conditions and subsequently results in a positive ECP shift.

References: Kim, Y.J, Lin, C.C., and Pathania, R., "Electrochemical Corrosion Potential Measurement With a Rotating Cylinder Electrode in 288°C Water," *Water Chemistry in Nuclear Reactor Systems 6*, Vol. 1, pp. 139-144, British Nuclear Energy Society, London, 1992.

Duration: from: 1991 to: 1993

Funding: N/A

Status: In progress

Last Update: September 3, 1993

U.S.A. R-356

EFFECTS OF ZINC ADDITIONS ON THE CRACK GROWTH RATE OF SENSITIZED STAINLESS STEEL AND ALLOYS 600 AND 182 IN 288°C WATER

Keywords: COMPONENT RELIABILITY; STRESS CORROSION CRACKING; ZINC; STAINLESS STEEL; ALLOY

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Objectives: The goal of this study was to evaluate the effects of 5 to 10 ppb Zn^{2+} addition on the stress corrosion crack growth rates of sensitized Alloy 600 using 25 mm compact type (CT) specimens, and sensitized type 304 stainless steel and Alloy 182 weld metal using double cantilever beam (DCB) specimens.

Comments: Zn additions of 5 to 10 ppb were consistently beneficial in reducing crack growth rates of sensitized Alloy 500, sensitized type 304 steel, and Alloy 182 weld metal in ~288°C water containing 0 or 200 ppb O_2 and 0 to 0.4 μM H_2SO_4. The reduction in crack growth rate from Zn addition ranged from about a factor of 5 for sensitized type 304 stainless steel or Alloy 182 weld metal in 282°C pure water at low corrosion potential, to about a factor of 2 for sensitized Alloy 600 in 288°C water containing 200 ppb O_2 and 0.3 μM (0.267 μS/cm) H_2SO_4. Approximately a factor of 2 was also observed in a four-inch diameter, weld sensitized type 304 stainless steel pipe.

Remarks/Potential for dose limitation: In the last several years, ZnO additions to BWR water have been studied, primarily because of their beneficial influence in reducing buildup of radioactive species such as Co-60 in the oxide film of structural components. Its success and lack of notable adverse side effects has resulted in its implementation in a few BWRs. However, no prior evaluation of Zn additions on crack growth has been performed.

References: Andresen, P.L., and Diaz, T.P., "Effects of Zinc Additions On the Crack Growth Rate of Sensitized Stainless Steel and Alloys 600 and 182 in 288°C Water," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, pp. 169-175, British Nuclear Energy Society, London, 1992.

Duration: from: 1991 to: 1992

Funding: N/A

Status: Completed

Last Update: September 3, 1993

ON-LINE MEASUREMENT OF PARTICLES IN REACTOR WATER OF BWRS

Keywords: CONTAMINATION PREVENTION; ON-LINE MEASUREMENT; RADIATION FIELD; WATER CHEMISTRY

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Objectives: To measure the number, size, and composition of particles in the primary cooling water of a BWR.

Comments: A high temperature and pressure cell (290°C, 90 bar) was developed and tested in an out-of-pile loop. The equipment was subsequently used in cooled reactor water in the NPP Leibstadt. During steady state, power reduction, and the shutdown operation, the number and size of particles were measured. Additional chemical and radiological analyses were done to give information regarding particle composition. A POLYTEC HC 70 was used to measure suspended material in the reactor water. By measuring on-line, the investigators were able to determine at what times during the reactor cycle the activated products were bound to particles, and at what times they were "dissolved" and not bound.

Remarks/Potential for dose limitation: In the primary circuits of water cooled reactors, activated corrosion products such as Co-58, Co-60, and Mn-54 are transported and deposited on the walls. Since the deposited activity causes a radiation field that makes maintenance work more difficult, it would be of great importance to control the deposition process.

References: Schenker, E, Francioni, W., and DeGueldre, C., "On-line Measurement of Particles in Reactor Water of BWRs," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, pp. 133-139, British Nuclear Energy Society, London, 1992.

Duration: from: 1990 to: 1992 **Funding:** N/A

Status: Completed **Last Update:** June 10, 1993

TAIWAN R-358

THE INTEGRITY OF INCONEL ALLOYS IN HIGH TEMPERATURE WATER CHEMISTRY

Keywords: COMPONENT RELIABILITY; NICKEL; ALLOY; WATER CHEMISTRY; INCONEL

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Objectives: To investigate the electrochemical behavior of nickel-based alloys using potentiodynamic technique in sulfate and/or chloride environments at 316°C.

Comments: A low chromium alloy was designed to simulate the grain boundary composition of sensitized Inconel 600. High pressure and temperature electrochemical technique was applied to evaluate the effect of sulfate and chloride ions on the corrosion behavior of nickel-based alloys. Anodic polarization scans were performed with a Solartron 1286 Electrochemical Interface that communicated with an HP computer at a scan rate of 1 mV/s. The platinum counter electrode and Ag/AgCl reference electrode were maintained in de-aerated electrolytes. It was found that the trend of high temperature electrochemical polarization curves is a little different for nickel-based alloys in neutral, acidic, and alkaline solutions. Chloride ions tend to corrode metals catalytically, especially in highly acidic media, and sulfate ions are less damaging in the corrosion process, providing sulfate salts can be formed on active sites.

Remarks/Potential for dose limitation: High temperature water chemistry is a critical issue in determining the life assessment of heat exchanger tubes made of nickel-based alloys. There are two major contaminants, sulfate and chloride anions, that may be present in PWRs and affect the material performance. In creviced regions, the level of impurity concentration may be elevated by as much as four orders of magnitude. At high concentrations of sulfate and chloride ions, nickel-based alloys quite possibly suffer from a detrimental corrosion problem

References: Yang, I.-J., "The Integrity of Inconel Alloys in High Temperature Water Chemistry," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, pp. 177-180, British Nuclear Energy Society, London, 1992.

Duration: from: 1991 to: 1992

Funding: N/A

Status: Completed

Last Update: June 11, 1993

U.K. R-359

ENRICHED BORON PRODUCTS

Keywords: COMPONENT RELIABILITY; BORON; ENRICHED BORIC ACID; WATER CHEMISTRY; STRESS CORROSION CRACKING

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Objectives: Document the usefulness of enriched boron products in nuclear power plants, specifically enriched boric acid at PWRs and enriched sodium pentaborate at BWRs.

Comments: In PWRs, boric acid is used as a solubility reactivity control agent and is referred to as a chemical shim because of the high capture cross section (3815 barns) for thermal neutrons exhibited by the boron 10 isotope contained in the boric acid. Using enriched boric acid (EBA), which is enriched to 99% B-10, allows for the decrease of total boron concentration without changing the boron 10 content of the reactor cooling water. In order to maintain the pH level of the water, the quantity of lithium hydroxide needed is dramatically reduced as well. This keeps the lithium limits below concentrations thought to accelerate Zircaloy corrosion and may eliminate or delay initiation of primary water stress corrosion cracking of steam generator tubes.

In order to upgrade the stand-by liquid control systems in BWRs, a solution more concentrated in boron 10 was needed. Enriched sodium pentaborate was developed for this purpose.

Remarks/Potential for dose limitation: The use of EBA could allow longer fuel cycles, reduced man-rem exposure, reduced maintenance costs, and prevent major plant modifications such as enlarging the boric acid tanks when going to lower concentrations or higher burn-up fuels. Greater safety, reduced corrosion and plant life extension are just some of the possible benefits of using enriched boron products.

References: Goehlich, V., "Enriched Boron Products," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, pp. 187-189, British Nuclear Energy Society, London, 1992.

Duration: from: 19 to: 1992

Funding: N/A

Status: In Progress

Last Update: September 3, 1993

VARIABILITIES IN THE CALCULATION OF PWR PRIMARY COOLANT pH

Keywords: CONTAMINATION PREVENTION; PH; WATER CHEMISTRY

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Project Manager:

Objectives: pH values vary greatly with temperature and can vary significantly with different methods of calculation. In this paper, these variations are quantified in order to aid cross-comparison of literature values.

Comments: In the past, a variety of methods have been used leading to differences in values obtained for pH. Up to the present, calculations have usually been at the inlet temperature (often approximated to 285°C) or at 300°C. The Electric Power Research Institute (EPRI) recommends calculation at the average primary coolant temperature, which is different in each plant and may vary with time. The results from two methods of pH calculation are graphed in this paper, and the discrepancies between the two can be seen.

Remarks/Potential for dose limitation: The lithium/boron regime adopted for primary coolant chemistry has an important effect on corrosion product activity transport and hence on dose rates around the primary circuit.

References: Polley, M.V., "Variabilities in the Calculation of PWR Primary Coolant pH," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, pp. 192-193, British Nuclear Energy Society, London, 1992.

Duration: from: 1991 to: 1992

Funding: N/A

Status: Completed

Last Update: September 3, 1993

CONSTRUCTION AND OPERATION OF AN IN-PILE LOOP FOR BWR COOLANT CHEMISTRY STUDIES

Keywords: CONTAMINATION PREVENTION; COMPONENT RELIABILITY; PRIMARY COOLANT CHEMISTRY; CORROSION; NITROGEN

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Objectives: This paper discusses the construction and operation at the MIT Research Reactor of an in-pile loop which simulates BWR coolant conditions. The loop was designed to carry out coolant radiolysis studies, with a focus on O_2 and H_2O_2 generation, electrochemical corrosion potential, and N-16 chemistry.

Comments: The BWR Coolant Chemistry Loop (BCCL) had once-through flow. Neutron and gamma dose rates and core exit quality are comparable to those in an actual BWR. With the exception of the in-core Zircaloy tubing, the system is constructed almost entirely of titanium to insure water purity, and a chemical injection system is provided for controlled addition of chemicals of interest. Charging tank cover gas composition can be varied to simulate a range of conditions between NWC and HWC, or to add a wide variety of gaseous species for test purposes.

Remarks/Potential for dose limitation: Coolant chemistry in a BWR has important effects on materials integrity, ex-core radionuclide deposit, and steam plant dose rates. Radiolysis dominates and is hard to quantify fully by computation alone. This motivated the construction and operation of this experimental facility.

References: Kohse, G.E. et al, "Construction and Operation of an In-Pile Loop for BWR Coolant Chemistry Studies," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, pp. 190-191, British Nuclear Energy Society, London, 1992.

Duration: from: 1991 to: 1992

Funding: N/A

Status: Completed

Last Update: June 11, 1993

WATER CHEMISTRY DURING THE SHUT-DOWN OF THE BOILING WATER REACTOR LEIBSTADT

Keywords: CONTAMINATION PREVENTION; WATER CHEMISTRY; SHUTDOWN CHEMISTRY; PARTICULATE; HYDROGEN PEROXIDE; CORROSION PRODUCT

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Objectives: In order to better understand the reasons for activity increase in reactor water during shut-down, an extensive measuring campaign was carried out during the shut-down of the BWR Leibstadt (KKL).

Comments: The measurements began at 72% total power, or 70 hours before zero power, and lasted until 80 hours after zero power. Particle size, size distribution, corrosion product concentrations (Fe, Ni, Cr, Mn, Co, and Zn), and hydrogen peroxide concentration were measured.

Remarks/Potential for dose limitation: Peaks in activity are mainly caused by undissolved corrosion products. The concentration of hydrogen peroxide increased rapidly when the temperature dropped below 160°C.

References: Wedda, H., Loner, H. and Schenker, E., "Water Chemistry During the Shut-Down of the Boiling Water Reactor Leibstadt," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, pp. 194-195, British Nuclear Energy Society, London, 1992.

Duration: from: 1991 to: 1992

Funding: N/A

Status: Completed

Last Update: June 11, 1993

FRANCE R-363

SOLUBILITY OF COBALT IN PRIMARY CIRCUIT SOLUTIONS

Keywords: CONTAMINATION PREVENTION; PRIMARY COOLANT CHEMISTRY; COBALT; SOLUBILITY

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Objectives: The solubility of cobalt ferrite ($CoFe_2O_4$) was measured in PWR primary circuit conditions in the temperature range 250-350°C. The results were compared with the ones obtained on magnetite and nickel ferrite.

Comments: It was found that in prevailing primary circuit conditions, the solubility of the cobalt ferrite was minimum at temperatures around 300°C. The equilibrium iron concentration is significantly lower than in the case of magnetite. The results are discussed in relation with the POTHY code, based only on thermodynamic laws and data, which was used for the prediction of the primary circuit chemistry.

Remarks/Potential for dose limitation: The agreement generally observed between equilibrium constants issued from experimental data and from direct calculation by POTHY confirm the validity of this code for any application in the primary circuit chemistry.

References: Lambert, I. and Joyer, F., "Solubility of Cobalt in Primary Circuit Solutions," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, pp. 196-197, British Nuclear Energy Society, London, 1992.

Duration: from: 1991 to: 1992

Funding: N/A

Status: Completed

Last Update: June 14, 1993

SWITZERLAND R-364

STATISTICAL ANALYSIS OF REACTOR WATER DATA

Keywords: CONTAMINATION PREVENTION; COBALT; RADIATION BUILDUP; MATHEMATICAL MODELS

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Objectives: To show that regression analysis is a simple tool to get an idea of which impurities in a reactor are important for the transport and deposition of Co-60 for further mechanistical studies.

Comments: The reactor water in BWRs is analyzed regularly and it is possible to obtain much information from these analyses. In this work, the regression analysis was chosen to produce empirical models for the Co-60 activity in the reactor water, taking account of other impurities over a time period where the reactor water chemistry was changed dramatically. The most simple regression is the linear regression:

$y = Xa + b$, where

- y vector of dependent variable
- X matrix including one constant predictor and m vectors x_i of explanatory variables
- a estimated parameters
- b residuals of the model compared to the measured values

Remarks/Potential for dose limitation: Mathematical modeling of the primary cooling system helps to see effects in the activity buildup due to changes in the reactor water chemistry. The primary cooling system of a BWR is a very complex system and difficult to model, but various efforts have been made creating mathematical models for radiation control in the primary system.

References: Loner, H., Alder, H.-P., Covelli. B., "Statistical Analysis of Reactor Water Data," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, pp. 200-201, British Nuclear Energy Society, London, 1992.

Duration: from: 1991 to: 1992

Funding: N/A

Status: Completed

Last Update: June 11, 1993

MIXED OXIDE-ALLOY-WATER SYSTEMS UNDER LWR CONDITIONS

Keywords: COMPONENT RELIABILITY; CORROSION; OXIDE FILM; PH

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Project Manager:

Objectives: To calculate the potential-pH diagrams for Fe-Cr-Ni alloys and for Fe-Zn systems, showing the regions of stability for mixed oxides.

Comments: The diagrams are only first approximations to the water-alloy equilibrium. They emphasize the fact that mixed oxides can form on the metal surface, with substantial impact on the pH-potential fields of stability and hence on the corrosion behavior of the metal. Better diagrams can be calculated when thermodynamic data become available for the Gibbs energy of formation of: a) the mixed oxides, especially those containing three or more metals, b) the ions in solution, which were approximated values.

Remarks/Potential for dose limitation: The corrosion of construction materials causes problems for reactors in the hot water in BWRs and PWRs, and in the room temperature water of service water systems. In corrosion processes, one must consider the kinds of films formed on the surface of the corroding metal, which are different for alloys from pure metals because of the formation of mixed oxides involving two or more of the constituent elements. To help in the corrosion analysis of alloys, the potential-pH diagrams were calculated.

References: Cubicciotti, D., "Mixed Oxide-Alloy-Water Systems Under LWR Conditions," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, pp. 206-207, British Nuclear Energy Society, London, 1992.

Duration: from: 1991 to: 1992

Funding: N/A

Status: Completed

Last Update: September 3, 1993

U.K. R-366

MAXIMUM ALLOWABLE CHLORIDE LEVELS ON STAINLESS STEEL COMPONENTS AT THE SIZEWELL "B" PWR

Keywords: COMPONENT RELIABILITY; STRESS CORROSION CRACKING; INTERGRANULAR ATTACK; CHLORIDE; PITTING

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Objectives: Assess the environmental conditions and chloride contamination levels under which corrosion (stress corrosion cracking, intergranular attack, and pitting) could occur at the Sizewell "B" PWR.

Comments: There are two distinct sets of conditions where a stainless steel plant could experience corrosion, namely those associated with storage when the plant is part-constructed and stored on site before final fabrication, and those which would be experienced by the plant during operation. As Sizewell is a coastal location, it was anticipated that the vendor-specified maximum chloride level of 0.015 $\mu g/cm^2$ would be difficult to achieve and maintain during construction and operation. In order to define what levels could be allowed, work was commissioned to establish levels of chloride that could initiate corrosion on typical Sizewell "B" stainless steel material over a range of typical conditions simulating storage and plant operation.

Remarks/Potential for dose limitation: The level of chloride contamination allowed on PWR stainless steel is governed by requirement to avoid corrosion associated with chlorides, including stress corrosion cracking, intergranular attack, and pitting. Based on the results of this study, a maximum chloride level of 0.1 $\mu g/cm^2$ would be appropriate for Sizewell "B". This will give a margin of safety and is attainable with appropriate care.

References: Allan, S.J., Atherton, D. and Lawson, W.F., "Maximum Allowable Chloride Levels on Stainless Steel Components at the Sizewell 'B' PWR," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, pp. 210-211, British Nuclear Energy Society, London, 1992.

Duration: from: 1991 to: 1992 **Funding:** N/A

Status: Completed **Last Update:** June 14, 1993

U.K. R-367

INORGANIC SEED MATERIALS FOR THE DECONTAMINATION OF PWR AQUEOUS WASTES

Keywords: CONTAMINATION REMOVAL; WASTE FILTRATION; INORGANIC SEED MATERIAL

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Objectives: Study the use of several inorganic sorbents, used in combination with crossflow membrane filtration, for the reduction of Cr-51 and Sb-125 levels in a PWR waste stream.

Comments: By adding to the waste effluent small quantities of solid "seed" materials onto which dissolved radionuclides can become absorbed, soluble contaminants can be removed to produce a clean effluent. Seeds specific to a number of different radionuclides have been identified (e.g. nickel hexacyanoferrate (II) for Cs-137). By employing a mixture of different seed materials, a wide spectrum of radionuclides can be dealt with. A mixture of titanium oxide, zirconium phosphate, and sodium nickel hexacyanoferrate (II) gave an overall decontamination factor of 20 at a solution pH of 4.5.

Remarks/Potential for dose limitation: Ultrafiltration is a filtration process that enables particles as small as 2nm in size to be removed from liquid suspension. Some aqueous waste streams arising at nuclear power plants are contaminated with very low levels of radioactive nuclides typically in the sub-μg kg^{-1} range. Many of the radionuclides in the wastes are present as colloidal or insoluble material and can therefore be potentially removed using ultrafiltration. Soluble species cannot be filtered directly, and inorganic seed materials must be used.

References: Hooper, E.W., Kavanagh, P. and Sellers, R.M., "Inorganic Seed Materials for the Decontamination of PWR Aqueous Wastes," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, pp. 214-215, British Nuclear Energy Society, London, 1992.

Duration: from: 1991 to: 1992

Funding: N/A

Status: Completed

Last Update: June 14, 1993

EASY INEXPENSIVE HYDROGEN WATER CHEMISTRY PREDICTIVE METHODS

Keywords: COMPONENT RELIABILITY; HYDROGEN WATER CHEMISTRY; IGSCC; COST REDUCTION

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Project Manager:

Objectives: This study investigated inexpensive and simple hydrogen water chemistry (HWC) predictive methods that eliminate the need for HWC minitests.

Comments: HWC is the addition of hydrogen to the feedwater of a boiling water reactor. The hydrogen reduces the concentration of dissolved oxygen in the primary coolant to a level that will no longer facilitate intergranular stress corrosion cracking of stainless steel components. Each BWR responds differently to such hydrogen injections which lead to expensive HWC minitests. The main difference between the Aptech HWC predictive methods and a minitest is that no hydrogen needs to be added to the feedwater. Rather, the predictive methods utilize the indigenous hydrogen and oxygen generated by radiolysis in the core.

Remarks/Potential for dose limitation: The ability of any specific downcomer/jet pump region to facilitate the recombination of hydrogen and oxygen varies from plant to plant. This has required expensive HWC minitests in order to determine the precise amount of feedwater hydrogen needed to reduce the ability of the recirculating water to facilitate IGSCC of the external recirculation piping, typically made of welded steel. Such mini tests cost upwards of $1.0 million.

References: Fox, M., "Easy Inexpensive Hydrogen Water Chemistry Predictive Methods," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 1, pp. 222-223, British Nuclear Energy Society, London, 1992.

Duration: from: 1990 to: 1992

Funding: N/A

Status: Completed

Last Update: June 16, 1993

POTENTIAL-pH DIAGRAMS FOR ALLOY-WATER SYSTEMS UNDER LWR CONDITION

Keywords: COMPONENT RELIABILITY; POURBAIX DIAGRAM; CORROSION; OXIDE FILM; MIXED OXIDE

Principal Investigator:

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Project Manager:

Objectives: To calculate from thermodynamic data the regions of oxide compound stability (Pourbaix diagrams) for the water-Fe-Cr-Ni system at room temperature and at typical LWR temperature. Diagrams for Fe-Cr-water, Fe-Ni-water, and Cr-Ni-water systems are also given.

Comments: Potential-pH or Pourbaix diagrams present the regions of stability of metallic species with water. They indicate the areas of potential and pH in which the dissolved species, the metal oxides, or the metal itself is stable. Regions for stable dissolved species are ones in which the metal can undergo corrosion. Where the metal itself is the stable form, the region is one of immunity. In the regions where the metal oxide is stable, the oxide can form on the metal as a protective layer.

Remarks/Potential for dose limitation: Corrosion of the PWR or BWR infrastructure contribute greatly to the overall radiation level. Controlling the pH of the primary water coolant is one method for reducing corrosion of metal surfaces. Pourbaix diagrams can contribute to this effort by identifying the ideal pH for metal or metal oxide stability.

References: Cubicciotti, D., "Potential-pH diagrams for alloy-water systems under LWR conditions," *Journal of Nuclear Materials*, Vol. 201, p. 176-183, 1993.

Duration: from: 1992 to: 1992

Funding: N/A

Status: Completed

Last Update: August 25, 1993

IN-PILE LOOP STUDIES OF CLOSE REDUCTION TECHNOLOGIES FOR PWRs AND BWRs; INVESTIGATIONS OF MATERIAL SUSCEPTIBILITY TO CRACKING

Keywords: CONTAMINATION PREVENTION; COMPONENT RELIABILITY; WATER CHEMISTRY; PH; N-16 CARRYOVER; STRESS CORROSION CRACKING

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Objectives:
1. For BWRs, reduce N-16 carryover by optimizing chemistry.
2. For PWRs, reduce radioactive corrosion product build up on the primary cooling surface by pH optimization of the water chemistry.
3. Reduce irradiation assisted stress corrosion cracking on core structural materials.

Comments:

Remarks/Potential for dose limitation: All of the research have the potential for major reductions in operational dose commitments in LWRs.

References:

1) "In-Pile Facilities for LWR materials and Chemistry Studies at the MIT Research Reactor," O.K. Harling, G.E. Kohse, M.J. Driscoll, R.G. Ballinger, JAIF Conference, Fukui City (1991)

2) "In-Pile PWR Loop Coolant Chemistry Studies in Support of Dose Reduction," G.E. Kohse, R.G. Sanchez, M.J. Driscoll, M. Ames, and O.K. Harling, JAIF Conference, Fukui City (1991)

3) "Development and Use of an In-pile Loop for BWR Chemistry Studies", EPRI TR-102248, July 1993.

Duration: from: 1985 to: 1993

Funding: $ 0.5 to 1.5 M / yr

Status: In progress

Last Update: August 26, 1993

EVALUATION OF FACTORS AFFECTING RADIATION FIELD TRENDS IN WESTINGHOUSE-DESIGNED PLANTS

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; RADIATION FIELD TREND; RADIATION FIELD CONTROL; RADIATION FIELD; COBALT; WATER CHEMISTRY; STEAM GENERATOR; DECONTAMINATION

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Objectives: Quantify the effects of cobalt input variations, operational chemistry, steam generator replacements, and decontamination on plant dose rates.

Comments: This is a summary of the third report of the Standard Radiation Monitoring Program initiated by Westinghouse and EPRI in 1977. It discusses the evaluation techniques and presents the results of the evaluation.

Remarks/Potential for dose limitation: Some of the changes in radiation fields are:

* Plant startup with Zircaloy vs Inconel fuel grids: 19% reduction
* Coordinated vs Uncoordinated Chemistry: 15% reduction
* Inconel fuel grids without high cobalt impurity vs ones with high impurity: 18% reduction
* Overall improvement with initial Zircaloy fuel grids and coord. chemistry: 45% reduction
* channel head radiation levels in plants that have replaced steam generators with low Co impurity tubing are 65% of previous levels
* The dose rate in channel heads that have been chemically decontaminated builds up to only 60% of pre-decontamination levels
* Modified primary coolant chemistry (PCC) results in about 25% lower component dose rates compared with coordinated PCC
* An elevated coolant pH results in lower radiation levels by a factor of two compared to modified or coordinated pH
* Plants started in 1985 and after have a factor of 2 less doses compared to pre-1985 plants

References: Bergmann, C.A., Perock, J.D., "Evaluation of Factors Affecting Radiation Field Trends in Westinghouse-Designed Plants," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 2, pp. 16-23, British Nuclear Energy Society, London, 1992.

Duration: from: 1977 to: 1991

Funding: N/A

Status: Completed

Last Update: August 27, 1993

BNL ALARA Center Data Base

CANADA R-372

THE MECHANICS AND KINETICS OF CORROSION PRODUCT RELEASE FROM CARBON STEEL IN LITHIATED HIGH TEMPERATURE WATER

Keywords: CONTAMINATION PREVENTION; COBALT RELEASE; CORROSION PRODUCT RELEASE; OXIDE FILM

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Objectives: A Formulation of the Cobalt Release from Carbon Steel

Comments: Data has been analyzed. Paper in preparation for publication.

Remarks/Potential for dose limitation: Understanding cobalt transport will have a large bearing on reducing radiation fields.

References: In preparation

Duration: from: 1991 to: 1993

Funding: N/A

Status: Completed

Last Update: August 30, 1993

CANADA R-373

INVESTIGATION OF THE CHEMICAL AND PHYSICAL PROPERTIES OF SPINEL OXIDES

Keywords: CONTAMINATION PREVENTION; COBALT RELEASE; CORROSION PRODUCT RELEASE; OXIDE FILM

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Objectives: To determine the affinity of various corrosion product spinels to the oxide layers formed by corrosion product release in high temperature water

Comments:

Remarks/Potential for dose limitation: Possible mechanisms for trapping various radioactive species will be studied. These directly influence the buildup of radiation fields.

References: In preparation

Duration: from: 1993 to: 1993

Funding: N/A

Status: Initiated

Last Update: August 30, 1993

U.K. R-374

OVERVIEW OF THE IMPACT OF STELLITE REMOVAL ON RADIATION FIELDS IN KWU PWRs

Keywords: CONTAMINATION PREVENTION; STELLITE; HARD FACING ALLOY; COBALT SOURCE; STAINLESS STEEL; RADIATION FIELD

Principal Investigator:

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Project Manager:

Objectives: Determine the following based upon data from Siemens PWRs:
1) the effect of a progressive reduction in Stellite on radiation fields
2) the consequences of replacing Inconel 718 gridded fuel by Zircaloy
3) the effect of eliminating antimony from the main coolant pump bearings
4) the effect of an increase in the pH of the primary coolant

Comments: This study updates an earlier analysis covering data collected up to 1989 on primary circuit activity levels and radiation fields from Siemens (KWU) PWRs. The work covers Co-60, Co-58, and Sb-124 concentrations in the primary coolant and deposition on primary circuit surfaces.

Remarks/Potential for dose limitation:

1. Stellite, other than in the control rod drive mechanisms, is the major contributor of Co-60
2. Cobalt in the primary circuit structural materials are minor contributors to radiation fields
3. In-vessel Stellite is the most important Stellite source
4. Co-60 contributions to dose rates correlate with cycle-averaged total Co-60 concentrations in the primary coolant
5. Co-58 production showed no particular trend from plant to plant as Stellite was reduced
6. Co-58 contributions to dose rates were reduced on replacing Inconel 718 gridded fuel with Zircaloy gridded fuel
7. Sb-124 contributions to dose rates and circulating coolant concentrations were low for plants commissioned with antimony-free main coolant pump bearings
8. Sb-124 is a significant contributor to dose rates
9. Operation with the KWU/VGB high pH regime in Siemens PWRs reduced dose rates
10. The later KWU PWRs, which have eliminated both Stellite and antimony sources, have extremely low primary circuit radiation fields and yearly personnel doses

References: Garbett, K., "Overview of the Impact of Stellite Removal on Radiation Fields in KWU PWRs," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 2, pp. 31-38, British Nuclear Energy Society, London, 1992.

Duration: from: 1989 to: 1992
Funding: N/A
Status: Completed
Last Update: August 30, 1993

ACTIVITY TRANSPORT AND CORROSION PROCESSES IN PWRs

Keywords: CONTAMINATION PREVENTION; WATER CHEMISTRY; PH; PWR COOLANT; COBALT; COBALT SOURCE; CORROSION; CRUD; CRUD TRANSPORT; CORROSION PRODUCT TRANSPORT; CORROSION PRODUCT DEPOSITION; CORROSION PRODUCT

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Project Manager:

Objectives: Outline current understanding of activity processes in PWRs.

Comments: This study discusses the major scientific principles underlying the reduction of radioactivity within reactor systems and the controlling of radiation fields around components. The following topics are covered:

I. Basic Theory of Activity Transport
II. The Production of Radioactive Species
- A) Particle Transport
- B) Particle Formation
- C) In-core Processes Involving Dissolved Cobalt
- D) Sources of Cobalt

III. The Activation of Out-of-Core Surfaces

Remarks/Potential for dose limitation:

1) Solubility differences move corrosion product oxides between in-core and out-of-core surfaces.
2) The source of dissolved corrosion products, especially cobalt, is described in terms of "corrosion release" from corroding surfaces.
3) The contamination of out-of-core surfaces is dependent upon the contamination parameter, which describes the properties of the oxides growing on the surfaces.
4) From information on cobalt adsorption-desorption processes on in-core materials, coupled with the release and incorporation processes on out-of-core surfaces, a simple model for the evolution of total cobalt concentration in PWR coolant can be devised.

References: Lister, D.H., "Activity Transport and Corrosion Processes in PWRs," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 2, pp. 49-60, British Nuclear Energy Society, London, 1992.

Duration: from: 1990 to: 1992
Status: Completed

Funding: N/A
Last Update: August 31, 1993

FEASIBILITY OF ON-LINE MONITORING OF STRESS CORROSION CRACKING IN ROTATING COMPONENTS

Keywords: COMPONENT RELIABILITY; STRESS CORROSION CRACKING; CORROSION MONITORING; CORROSION TESTING; ON-LINE MONITORING

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Objectives: To demonstrate the feasibility of sending electrochemical signals telemetrically between an instrumented stress corrosion cracking (SCC) specimen and a data acquisition system.

Comments: A method of SCC monitoring involving the measurement of electrochemical current noise (ECN), electrochemical potential noise (EPN), and zero resistance ammeter (ZRA) signals has been developed recently. In this demonstration, the test apparatus consisted of a working electrode - a compact tension specimen made from 3CrMo steel - stressed in a 30 wt% sodium hydroxide solution at 75°C. Unstressed counter-electrodes of the same material, also immersed in the caustic solution close to the precracked region of the compact tension specimen, allowed measurement of the ECN, EPN, and ZRA signals generated during stress corrosion crack growth. A prototype single-channel data transmission unit received ECN signals from the electrodes and transmitted them telemetrically to the antenna of a signal receiver unit positioned a short distance away.

Remarks/Potential for dose limitation: SCC was detected in the specimens by means of the ECN signals received. The ECN signals were successfully digitized and transmitted at radio frequencies across an air gap to a receiver and decoder, where they were converted to signals suitable for conventional processing and storage. Future SCC detection systems based on this principle can significantly reduce the manual labor needed for SCC inspection, thereby reducing worker radiation dosage.

References: "Feasibility of On-Line Monitoring of Stress Corrosion Cracking in Rotating Components," EPRI TR-102537 Final Report, Electric Power Research Institute, Palo Alto, CA, June 1993.

Duration: from: 1991 to: 1993

Funding: N/A

Status: Completed

Last Update: August 31, 1993

CONCEPT AND EXPERIENCE OF SYSTEM DECONTAMINATION WITH CORD

Keywords: CONTAMINATION REMOVAL; CORD PROCESS; DECONTAMINATION; FULL SYSTEM DECONTAMINATION; SUB-SYSTEM DECONTAMINATION; COBALT REMOVAL

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Objectives: Outline the Siemens concept for the decontamination of systems with the CORD process. The results of sub-system and full-system decontaminations of a PWR and a BWR is presented.

Comments: The latest decontamination processes used by Siemens AG KWU may be characterized as follows:

- low chemical concentrations
- simple analytical monitoring
- short treatment cycles of a few hours with continuous adjustment of the chemical concentration
- waste reduction to the virtual elimination of secondary waste

The use of permanganic acid as oxidation agent in the process makes it possible to perform a decontamination cycle (oxidation, reduction, and decontamination) with only one system fill, thereby avoiding an intermediate cleanup and rinsing and reheating step.

Also, the CORD process does not leave chelating agents in the final waste.

There is virtually no external equipment required for the decontaminations of Siemens PWR primary loop. But for Westinghouse built 3 loop units and BWRs, additional purificaition equipment is required.

Remarks/Potential for dose limitation:

- A recirculation loop decontamination at the 640 MWe NPP Wurgassen led to a saving in personnel dose of 2000 mSv.
- During the decontamination of the sealing water supply system of the internal axial flow pumps with CORD in a German BWR (KKI), the personnel dose was reduced by 300 - 400 mSv.
- CORD was applied in 1990 and 1991 to Swedish BWR (OKG) subsystems. The two decontaminations resulted in savings of 250 and 200 mSv for the repair and inspection work respectively.

- In 1991, a FSD with CORD at the BR3 in Mol/Belgium resulted in personnel dose savings between 4000 and 8000 mSv for subsequent dismantling operations.

References: Wille, H., Bertholdt, H.O., "Concept and Experience of System Decontamination with CORD," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 2, pp. 161-167, British Nuclear Energy Society, London, 1992.

Duration: from: 1991 to: 1992

Funding: N/A

Status: Completed

Last Update: September 1, 1993

ELOMIX: A BETTER WAY OF HANDLING THE WASTE FROM DECONTAMINATION

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; LOMI; ELOMIX; RADWASTE MINIMIZATION; WASTE VOLUME REDUCTION

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Objectives: To develop and verify the electrochemical LOMI Ion Exchange (ELOMIX) process for subsystem and possibly full system decontamination.

Comments: The ELOMIX process has the advantage of reducing the volume of secondary decontamination waste and transferring the contaminants into a stable inorganic form. This is accomplished by means of an electrochemical cell consisting of resin sandwiched between the anode and cathode. The radioactive debris are deposited at the cathode.

A small prototype ELOMIX cell was successfully tested at Commonwealth Edison's Dresden unit 2 in Oct. 1990. In May 1992, a larger scale cell capable of processing 30 litres/hr was constructed and demonstrated at Gulf States Utilities River Bend plant. The larger demo has shown that: 1) the ELOMIX system can be cleaned and transported, and 2) the metallic waste can be transferred hydraulically, enabling efficient treatment. Work is ongoing to build one full scale cell for design verification.

Remarks/Potential for dose limitation: The benefits of this process are:

1) Reduction in waste volume by factors of up to 140.
2) Conversion of the radioactive waste into a chemically more stable inorganic form.
3) The possibility of long term on-site storage of the waste.

References: Tucker, P. M., "ELOMIX: A Better Way of Handling the Waste From Decontamination", *Nuclear Engineering International*, Vol. 38, No. 463, pp. 18-21, Feb 1993.

Duration: from: 1989 to: 19

Funding:

Status: In progress

Last Update: August 19, 1993

U.S.A. R-379

BWR/5 FULL-SYSTEM DECONTAMINATION FEASIBILITY STUDY

Keywords: CONTAMINATION REMOVAL; FULL SYSTEM DECONTAMINATION; DECONTAMINATION; RADIATION FIELD CONTROL

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Objectives: To determine the engineering feasibility and cost-effectiveness of complete reactor system decontamination of a BWR/5 plant using the LOMI process.

Comments: The EPRI report TR-100049 concluded that BWR full-system decontamination was technically and economically feasible on a BWR/3 plant design. EPRI conducted another FSD study on a BWR/5 plant because of several design differences compared with the BWR/3. The approach was to determine the applicability of decontamination studies at Commonwealth Edison Company's Quad Cities BWR/3 to its LaSalle County BWR/5.

Remarks/Potential for dose limitation: The conclusions are:

1) Full System Decontamination of the BWR/5 at LaSalle would require approximately 30% less reagent and ion-exchange resins compared with the BWR/3.
2) No new, untested materials would be exposed to the decontamination solvent.
3) Only minor changes would occur in the proposed operation of BWR systems during the decontamination, with less decontamination equipment required.
4) Estimated costs and benefits would be similar to the BWR/3 ($7.8 million cost, $12.6 million benefit).
5) Exposures during the decontaminations would increase slightly (54 vs 42 rem).

References: "BWR/5 Full-System Decontamination Feasibility Study", EPRI TR-102332 Final Report, May 31, 1993.

Duration: from: 1991 to: 1993

Funding:

Status: Completed

Last Update: August 19, 1993

SWEDEN R-380

MOVING FROM ULTRA-PURE BWR WATER TO PLANT-TAILORED WATER CHEMISTRY

Keywords: CONTAMINATION PREVENTION; WATER CHEMISTRY; PRIMARY COOLANT CHEMISTRY; CORROSION; CORROSION CONTROL; CORROSION PRODUCT; IGSCC; STRESS CORROSION CRACKING; HYDROGEN WATER CHEMISTRY; OXYGEN CONTENT

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Project Manager:

Objectives: Determine the impact of chemistry on activity build-up and on materials behaviour.

Comments: The study investigated whether material corrosion can be reduced in Swedish BWRs by controlling the water chemistry are described. Some of the topics covered include:

- Problem with the behaviour of copper corrosion products in the reactor core
- Combating pipe cracking by adjusting the concentration of dissolved oxygen in the main recirculation water
- Cobalt reduction by controlling the iron-nickel concentration ratio and by zinc addition
- Dealing with sulphate and chloride, which have a strong enhancing effect on IGSCC, in the reactor water
- Hydrogen Water Chemistry
- pH adjustment by addition of alkali metal hydroxides to the reactor water
- Condensate clean-up system related aspects

Remarks/Potential for dose limitation: The water chemistry activities still aim at achieving low radioactive contamination of the reactor systems; control of identified harmful chemical compounds in the process waters; and reduction of radioactive wastes. The way to improvements routes via better information and profound understanding of the fundamental scientific background of operating experiences.

References: Fejes, P. "Moving From Ultra-Pure BWR Water to Plant-Tailored Water Chemistry," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 2, pp. 90-95, British Nuclear Energy Society, London, 1992.

Duration: from: 1991 to: 1992

Funding: N/A

Status: Completed

Last Update: September 13, 1993

FRANCE R-381

EFFECTS OF pH OF PRIMARY COOLANT ON PWR CONTAMINATION

Keywords: CONTAMINATION PREVENTION; WATER CHEMISTRY; PRIMARY COOLANT CHEMISTRY; COOLANT; PH; PH CONTROL; DOSE RATE

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Project Manager:

Objectives: Investigate the effects of increasing pH(300)=7.0 to pH(300)=7.2 on six 900 MWe French PWRs.

Comments: Basic research has converged on an optimum pH lying between 7.2 and 7.4 with little difference between the two pH values. Thus, tests on six 900 MWe reactors were started in 1987 by increasing the Lithium concentration to obtain a pH(300) of 7.2. A systematic program for measuring the dose-rate around the primary piping of the reactors was set up. Each time there was a refueling shutdown, dose rate measurements were taken around the primary piping.

Two approaches to analysis were taken: 1) Comparing contamination in two groups of reactors with different chemical conditioning of the primary water. 2) Comparing the development of reactor contamination before and after changing the conditioning.

Remarks/Potential for dose limitation: The comparison in the two groups of reactors was inconclusive. The mean dose rate index for pH=7.0 was 59 (mRem/H) with a standard deviation of 19. For pH=7.2, it was 66 with SD of 21. The results were similarly inconclusive when comparing each reactor before and after changing the pH. Tests carried out on French EdF reactors did not reveal a significant impact due to changing primary coolant conditions to give a value greater than 7.0. But they did not contradict the tendencies revealed by the basic research. However, the effect of changing the pH is slight when compared with changes in design such as the composition of fuel assembly grids or steam generator production techniques.

References: Anthoni, S., Ridoux, P., Menet, O. and Weber, C., "Effects of pH of Primary Coolant on PWR Contamination," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 2, pp. 9-15, British Nuclear Energy Society, London, 1992.

Duration: from: 1987 to: 1993

Funding: N/A

Status: In progress

Last Update: September 14, 1993

BEHAVIOR OF PWRs IN SPAIN FOLLOWING CHANGES TO MODIFIED CHEMISTRY AND FUEL SPECIFICATIONS

Keywords: CONTAMINATION PREVENTION; PRIMARY COOLANT CHEMISTRY; WATER CHEMISTRY; PH; PH CONTROL; COBALT; RADIATION FIELD CONTROL

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Project Manager:

Objectives: Determine the impact on radiation fields at the large Spanish PWR plants of Asco 1&2, Almaraz 1&2, Trillo and specially Vandellos 2 from primary coolant chemistry control and fuel grid material replacement.

Comments: The main areas of the work presented include:

- Tight follow up of coolant chemistry and radiochemistry.
- Gamma spectrometry of the system surfaces.

Remarks/Potential for dose limitation:

- Three factors were found to contribute to system surfaces activation: 1) Cobalt impurity and Nickel content in plant materials, 2) Operating chemistry, 3) Cobalt ingress due to maintenance operations
- The optimum pH(T avg) for Vandellos 2 is just below 7.4
- Different response of Co-58 with respect to Co-60 and Fe-59 to pH changes suggests different source oxide in the core for them.
- There may be different optimum pH values for different radioisotopes.
- Trillo low radiation fields are related to both Co-58 and Co-60 activity.
- The high Co-60 activity in Almaraz 1 and Vandellos 2 is probably due to high Cobalt input from old specification fuel element grids and valve lapping respectively.
- The reduction of radiation fields in Almaraz 2 and Asco 1 and 2 has been due to adoption of the elevated pH chemistry and the fuel substitution by low Cobalt impurity at Almaraz and Zircaloy grids at Asco.

References: Fernandez Lillo, E., Boronat, M., Cascante, C., Adrada, J. and Ortega, A., "Behaviour of PWRs in Spain Following Changes to Modified Chemistry and Fuel Specifications," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 2, pp. 24-30, British Nuclear Energy Society, London, 1992.

Duration: from: 19 to: 1993

Funding: N/A

Status: In progress

Last Update: September 15, 1993

REVIEW OF EFFECT OF LITHIUM ON PWR FUEL CLADDING CORROSION

Keywords: COMPONENT RELIABILITY; FUEL CLADDING; ZIRCALOY; CORROSION; LITHIUM; WATER CHEMISTRY; PRIMARY COOLANT CHEMISTRY; OXIDE FILM

Principal Investigator:

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Project Manager:

Objectives: Review the reactor corrosion data on Zircaloy-4 which is currently available and to place this within the context of expectations derived from laboratory tests in an attempt to assess whether reactor corrosion rates are enhanced by operating an elevated-lithium regime.

Comments:

Laboratory Experience:
- Early isothermal tests showed that LiOH additions resulted in an increase in the rate of Zircaloy corrosion, especially in the post-transition region.
- Recent isothermal autoclave tests show that the deleterious influence of lithium additions is ameliorated even at quite low concentrations of boric acid.
- Tests done under boiling conditions indicate that realistic concentrations of boron can ameliorate the effects of high concentrations of lithium, even though boiling concentrates the lithium at the oxide surface

Experience From Reactors Operating Under Elevated-Lithium Regimes:
- Millstone Point 3: somewhat higher corrosion rates
- Ringhals: no significant enhancement in oxide thickness
- St. Lucie 1: seem to show a faster increase in corrosion rate
- Calvert Cliffs: no significant enhancement in corrosion rates
- The Halden In-Pile Loop: little effect on enhancing corrosion rates

Remarks/Potential for dose limitation: The reactor experience summarized show that NO gross deterioration in Zircaloy corrosion behaviour has been found from operation under elevated-lithium conditions. In cases where there seem to be lithium enhanced corrosion rates, more studies need to be done to ascertain whether the increased rates are due to high lithium concentration or some other cause.

References: Polley, M.V. and Evans, H.E., "Review of Effect of Lithium on PWR Fuel Cladding Corrosion," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 2, pp. 61-66, British Nuclear Energy Society, London, 1992.

Duration: from: 1991 to: 1992
Funding: N/A
Status: Completed
Last Update: September 15, 1993

SHUTDOWN CHEMISTRY IN SPANISH PLANTS

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; SHUTDOWN CHEMISTRY; OUTAGE TIME; DOSE RATE; BORATION

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Project Manager:

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Objectives: Investigate shutdown procedure improvements implemented in Westinghouse designed Spanish PWRs in the areas of reduced critical path time, in-core and ex-core activated corrosion products solubilization and reactor coolant system radiation measurements.

Comments: Since 1986 the five large Spanish Westinghouse PWR plants have instituted a series of modifications to the refueling shutdown procedures that have achieved significant time optimization and potential benefits in terms of ex-core activity reduction. These changes were:

- Boration to refueling shutdown concentration prior to cooldown.
- Use of a new charge of resin in the Chemical and Volume Control System mixed bed demineralizer in the H^+/OH^- form for coolant purification during shutdown.
- Reactor Coolant System draindown as soon as the Co-58 peak in solubility has been confirmed, with a coincident hydrogen peroxide residual, and has been observed to be undergoing reduction by ion-exchange purification.
- Out-of-core radiation fields are believed to be reduced by application of this optimized procedure in several plants. Calculations performed at Vandellos 2 during the 2nd refueling shutdown yield an estimate of 20% reduction.
- Critical path time for the refueling outages has been reduced significantly. Calculations at Asco 2 2nd refueling indicated times savings about 2.5 days.

Remarks/Potential for dose limitation: This evaluation has demonstrated the benefit in terms of dose rate reduction of establishing acid-reducing environments and maintaining them for certain time periods prior to establishing acid-oxidizing chemistry.

References: Llovet, R., Kormuth, J.W., Fernandez Lillo, E., Boronat, M., Ortega, A., "Shutdown Chemistry in Spanish Plants," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 2, pp. 67-73, British Nuclear Energy Society, London, 1992.

Duration: from: 1986 to: 1992

Funding: N/A

Status: Completed

Last Update: September 16, 1993

EFFECT OF SURFACE TREATMENT ON RADIOACTIVITY DEPOSITION ON STAINLESS STEEL COUPONS EXPOSED IN DOEL 2

Keywords: CONTAMINATION PREVENTION; COMPONENT RELIABILITY; PRECONDITIONING; SURFACE PREPARATION; SURFACE TREATMENT; SURFACE CONDITIONING; STAINLESS STEEL; CHROMIUM; PALLADIUM COATING; ELECTROPOLISHING; PASSIVATION

Principal Investigator:

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Project Manager:

Objectives: Present the results of a joint LABORELEC-EBES/Nuclear Electric/EPRI study of surface pre-conditioning and coatings to reduce radioactivity uptake on surfaces undertaken on the Doel 2 PWR in Belgium.

Comments: Stainless steel 304L, 309L, 316L and CF8M coupons have been exposed for between one and three cycles in the hot and cold legs of a steam generator channel head on Doel 2 PWR between 1986 and 1991. In the present paper results from examination of coupons exposed up to 1991 during Cycles 14, 15 and 16 are reported. The surface finishes on these coupons included as received, electropolished, electropolished/passivated, chromium plated, chromium plated/passivated and palladium coated. Results from gamma spectrometry and scanning electron microscopy examinations of the coupons are reported. The most dramatic result is the very low activity uptake on the chromium plated coupons.

Remarks/Potential for dose limitation: Occupational radiation exposure arises principally from exposure to radiation fields on out-of-core surfaces during maintenance and inspection operations. Reduction of these radiation fields by treatment of surfaces is in the interest of ALARA.

The main conclusions from this study are:

- Chromium plated coupons showed an order of magnitude lower levels of activity than as received coupons.
- The activity levels of the 309L, 316L and CF8M chromium plated coupons were very similar.
- The addition of a passivation stage after the chromium plating treatment had a detrimental effect by increasing activity uptake by up to a factor of two.
- Electropolished/passivated (E/P) CF8M coupons have a factor of 2 lower activity while E/P 309L coupons have a slightly increased level of activity uptake compared

with the as received coupons.

- After over 3 Cycles of exposure, the palladium coatings yielded no clear reduction of Co-58 and Co-60 uptake and enhanced uptake of Cr-51, Ag-110m, Sn-113 and Sb-125.

References: Pick, M.E., Young, M.A. and Roofthooft, R., "Effect of Surface Treatment on Radioactivity Deposition on Stainless Steel Coupons Exposed in Doel 2," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 2, pp. 74-79, British Nuclear Energy Society, London, 1992.

Duration: from: 1986 to: 1992

Funding: N/A

Status: Completed

Last Update: September 16, 1993

U.S.A. R-386

UTILITY APPROACH TO RADIATION FIELD REDUCTION BY COOLANT CHEMISTRY CONTROL

Keywords: CONTAMINATION PREVENTION; ALARA; PH; PH CONTROL; ZINC INJECTION; WATER CHEMISTRY; PRIMARY COOLANT CHEMISTRY; RADIATION FIELD CONTROL; COBALT REDUCTION; LITHIUM

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Project Manager:

Objectives: Describe the field testing and implementation of zinc injection at Millstone 1 (BWR), and elevated pH control at Millstone 3 (PWR). The results after two cycles of operations are presented.

Comments: In 1986 Northeast Utilities (NU) initiated an aggressive ALARA improvement program for application at all four of its nuclear generating stations. A part of this program was to focus on achieving lower radiation fields by implementing the latest technologies available in coolant chemistry control, cobalt source removal, component surface conditioning and decontamination.

Remarks/Potential for dose limitation: Two cycles of operation with zinc injection at Millstone 1 were successful in reducing surface dose rates by factors up to 2. The zinc injection process does not seem to introduce any unmanageable concerns, while the expected use of zinc depleted in Zn-64 is likely to improve considerably the benefits of this process. Elevated pH control for two cycles at Millstone 3 has also been successful in controlling the increase of dose rates in SG channel heads and associated piping. Unfortunately, the fuel cladding oxide concerns, possibly enhanced by operation at higher lithium levels, have led to a temporary respite from elevated pH operations until cladding oxidation limitations are better defined, and/or more corrosion resistant cladding alloys become generally available.

Decontamination, surface treatments and cobalt source removal programs are in place or are being introduced at all of the NU nuclear stations. It is anticipated that these programs, in conjunction with the coolant chemistry programs, will continue to reduce occupational radiation exposure at all of NU's operating reactor units.

References: Hudson, M.J.B. and Klisiewicz, J.W., "Utility Approachy to Radiation Field Reduction by Coolant Chemistry Control," *Water Chemistry of Nuclear Reactor Systems 6*, Vol. 2, pp. 96-102, British Nuclear Energy Society, London, 1992.

Duration: from: 1986 to: 1992

Funding: N/A

Status: Completed

Last Update: September 17, 1993

LOW PICOLINATE LOMI - UPDATE

Keywords: CONTAMINATION REMOVAL; LOMI; DECONTAMINATION; FULL SYSTEM DECONTAMINATION; RADWASTE MINIMIZATION

Principal Investigator:

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Project Manager:

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Objectives: Review all aspects of the use of low picolinate LOMI.

Comments: It was observed that the molar ratio of picolinic acid to vanadium employed during full system decontamination (FSD) applications at the Winfrith SGHWR was 3:1. This is 2 times lower than the 6:1 ratio normally used in the U.S. Detail cost-benefit analyses indicated significant financial benefits if the lower ratio could be safely applied in BWR FSD applications.

Theoretical calculations and experimental studies have confirmed that reducing the molar ratio of picolinic acid to vanadium during LOMI decontaminations from 6:1 to 3:1 will have no adverse effects whatsoever on the decontamination itself and will result in significant savings in chemicals, ion exchange resins and costs. No changes to the corrosion behavior of LOMI are expected.

Remarks/Potential for dose limitation: Based on the results of this study, the authors recommend that all future applications of LOMI employ a picolinic to vanadium molar ratio of 3:1 instead of 6:1 or 4.5:1. The advantage is that less chemicals and ion exchange resins are required. This translates to savings of approximately $236,000 and $295,000 per FSD application in BWRs and PWRs, respectively.

References: Smee, J.L. and Bradbury, D., "Low Picolinate LOMI -Update," *Fifth Workshop on Chemical Decontamination*, pp. 18.1-18.13, Electric Power Research Institute, Charlotte, North Carolina, 1993.

Duration: from: 1992 to: 1993

Funding: N/A

Status: Completed

Last Update: September 27, 1993

CANADA R-388

DECONTAMINATION CHEMISTRY: CURRENT ISSUES

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; FULL SYSTEM DECONTAMINATION; CAN-DECON; CAN-DEREM; ALKALINE PERMANGANATE; ZINC ADDITION

Principal Investigator: **Project Manager:**

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Objectives: Outline the current state of research and experience with the CAN-DECON and CAN-DEREM processes. The issues addressed include: 1) Qualification in the US, 2) AP decomposition and waste generation, 3) Effects of zinc addition.

Comments: The CAN-DEREM process has been qualified for use in PWR full-system decontaminations (FSD) in a major program carried out by Westinghouse in the US. A five step process (CAN-DEREM/AP/CAN-DEREM/AP/CAN-DEREM) was used as the reference case with the CAN-DEREM step applied at 115 °C (240 °F) for 24 hrs using reagent concentrations of 0.1 wt%. Lower concentrations of reagents and/or fewer steps can be used in order to reduce the volume of resin waste.

For AP/CAN-DEREM applications, examination of the projected ion-exchange resin wastes to be produced during a FSD of a PWR suggests that about 4 times more resin is produced by the AP steps than by the CAN-DEREM steps. In addition, application conditions for AP use have not been optimized, inducing decomposition in some cases.

It was observed that the Co-60 contaminated oxide formed in Zn containing coolant was easily removed without an AP pretreatment. The implication is significant in that ion-exchange resin wastes from decontamination could be dramatically reduced for PWR plants using Zn additions since the AP step could be avoided.

Remarks/Potential for dose limitation: To date, 15 CANDU reactor heat-transport-systems (including cores with fuel in place) have been decontaminated with the CAN-DECON process in Canada. In the US, 19 BWR and 5 PWR sub-systems were decontaminated using CAN-DECON. Since 1987, 15 BWR and PWR sub-systems have been decontaminated using the CAN-DEREM process. The decontamination factors for these 2 processes is about 5 to 10.

References: Speranzini, R.A., Miller, D.G., and Allsop, H.A., "Decontamination Chemistry: Current Issues," *Fifth Workshop on Chemical Decontamination*, pp. 20.1-20.33, Electric Power Research Institute, Charlotte, North Carolina, 1993.

Duration: from: 1991 to: 1993 **Funding:** N/A

Status: In progress **Last Update:** September 28, 1993

RESIN OXIDATION PROCESS IMPROVEMENTS

Keywords: CONTAMINATION REMOVAL; RADWASTE MINIMIZATION; WASTE VOLUME REDUCTION; RESIN OXIDATION

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Objectives: Investigate improvements in the Resin Oxidation Process. This is a method developed by EPRI to reduce the volume of ion exchange resin waste arising from reactor decontaminations.

Comments: The resin oxidation process is a low temperature and low pressure wet oxidation process developed primarily for destroying the organic content of ion exchange resins generated at nuclear power plants. The process is designed to be operated by mobile temporary equipment taken to the site where the resin is generated.

The presence of organic materials in radioactive waste can be detrimental to the stability and long term isolation of waste. The resin oxidation process was conceived with the objective of oxidizing the ion exchange resin waste under water at much lower temperatures than incineration. The process also destroys chelants in the resin wastes arising from decontamination operations.

A pilot-scale resin oxidation system was built by LN Technologies and operated at their premises and later at the EPRI NDE Center in Charlotte, NC. Several laboratory tests have been done to verify the process.

Remarks/Potential for dose limitation: This process has the potential to significantly reduce the cost of decontaminations by minimizing the ion exchange resin waste needed to be disposed.

References: Bradbury, D., Elder, G.R., and Kalinauskas, G.L., "Resin Oxidation Process Improvements," *FifthWorkshop on Chemical Decontamination*, pp. 23.1-23.7, Electric Power Research Institute, Charlotte, North Carolina, 1993.

Duration: from: 1986 to: 1993

Funding: N/A

Status: In progress

Last Update: September 29, 1993

DEVELOPMENT OF FULL SYSTEM DECONTAMINATION FOR BWRs

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; FULL SYSTEM DECONTAMINATION; LOMI; IGSCC; CORROSION; STRESS CORROSION CRACKING; ALKALINE PERMANGANATE

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Objectives: Initial evaluation of the feasibility of a full system decontamination (FSD) of a BWR (with fuel removed) using the LOMI process with specific emphasis on:

1) Long term materials compatibility and performance
2) LOMI process review and system interaction/flow paths
3) Radwaste management
4) Recontamination and dose savings
5) Cost benefit analysis
6) Safety review and licensing

Comments: The primary concern with BWR FSD is corrosion of the reactor internals. This report evaluates the applicability of FSD based on:

1) work previously performed to qualify the LOMI process
2) results of LOMI system and subsystem decontaminations
3) experience worldwide of FSD with LOMI

Based on this study, there appears to be no significant corrosion or materials concern for the application of LOMI for BWR FSD. Tests have shown that LOMI neither causes intergranular attack (IGA) or intergranular stress corrosion cracking (IGSCC), nor does it exacerbate existing IGA or IGSCC.

However, the NP-LOMI process is not recommended for FSD use because of some instances of IGA, enhanced crack growth rates, and SCC. The AP-LOMI process appears promising but additional testing is required before it can be considered fully FSD qualified.

Remarks/Potential for dose limitation:

- The program was successful in qualifying LOMI for BWR FSD with the fuel removed. No BWR plant or fuel material has been identified to be incompatible with LOMI.
- Documentation is now available to utilities for performing a BWR FSD. (EPRI Report TR-100049)

- Instead of being a potential risk to reactor internal integrity, FSD may become an integral part of the Optimum Water Chemistry's strategy for SCC minimization.

References: Gordon, B.M., "Topical Report on FSD for BWRs," *Fifth Workshop on Chemical Decontamination*, pp. 30.1-30.31 Electric Power Research Institute, Charlotte, North Carolina, 1993.

Duration: from: 1989 to: 1993

Funding: $719,000

Status: Completed

Last Update: September 29, 1993

U.S.A. R-391

FUEL DECONTAMINATION QUALIFICATION PROGRAM

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; FUEL DECONTAMINATION; FULL SYSTEM DECONTAMINATION; FUEL CLADDING; FUEL ASSEMBLY; CORROSION

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Objectives: Perform a fuel decontamination qualification program at the V.C. Summer site which will qualify full reactor coolant system (RCS) chemical decontamination with fuel in-place while maintaining existing warranties and incorporating improved fuel technologies.

Comments: At the request of South Carolina Electric and Gas Company, Westinghouse in 1989 developed a program to qualify nuclear fuel for full RCS Decon application. This program involved the chemical decontamination of actual fuel assemblies in a specialized canister in the fuel handling building at the V.C. Summer Nuclear Station with the same dilute chemical solvent parameters as were employed in the Full FCS Qualification Program.

Four fuel assemblies were decontaminated: two using CAN-DEREM and two using LOMI. Approximately 20 curies of Co-58 and Co-60 were removed from each assembly. The four decontaminated assemblies and two control assemblies were reinserted in the V.C. Summer plant for one more cycle with inspections at the next outage.

Remarks/Potential for dose limitation: Preliminary evaluation of the cladding corrosion oxide thickness measurements on the decontaminated and control assemblies indicates that the decontamination treatments have had no adverse affect on the post decontamination cladding corrosion behavior. However, a number of decontamination process application anomalies were observed. The principal conclusions are:

- The activity level in-core based on prior visual crud deposition and sampling data were underestimated.
- EDTA depletion was observed in the CAN-DEREM reducing step. As a result, the estimated waste volumes for full RCS decon need to be recalculated.
- No detectable quantities of carbon dioxide gas were observed. Thus, CO_2 generation is no longer expected to be a problem for full system decon.
- Full plant fuel-in decon should not be limited by fuel cladding corrosion performance.

References: Miller, R.S., Miller, P.E., and Peffer, D.R., "SCE&G Fuel Decontamination Qualification Program," *Fifth Workshop on Chemical Decontamination*, pp. 33.1-33.8, Electric Power Research Institute, Charlotte, North Carolina, 1993.

Duration: from: 1989 to: 1993

Funding: N/A

Status: In progress

Last Update: September 30, 1993

U.S.A. R-392

PACIFIC NUCLEAR FIELD IMPLEMENTATION

Keywords: CONTAMINATION REMOVAL; FULL SYSTEM DECONTAMINATION; DECONTAMINATION

Principal Investigator:

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Project Manager:

Objectives: Summarize Pacific Nuclear's scope of work for field implementation of the PWR full reactor coolant system (RCS) decontamination at Consolidated Edison of NY's Indian Point 2 Nuclear Power Station.

Comments: Pacific Nuclear has turnkey responsibility for all engineering, equipment, construction, decontamination processing and waste processing services, including craft labor for implementation of the full RCS decontamination project. The tasks being performed include:

1) Engineering for plant interface modifications
2) Engineering and fabrication of decon process system
3) Development of operating procedures for decon process
4) Construction of plant interface modifications
5) Decon process system testing and installation
6) Tie-in to reactor coolant system
7) Decon implementation (5 step CAN-DEREM/AP process)
8) Waste equipment supply and waste processing
9) Plant restoration and demobilization

Remarks/Potential for dose limitation: A full RCS decontamination final report will be prepared and issued describing the results of the decontamination process and the results of several tests which will be performed as part of the national demonstration of PWR full system decontamination.

References: Sheffield, J., "Pacific Nuclear Field Implementation," *Fifth Workshop on Chemical Decontamination*, pp. 29.1-29.15, Electric Power Research Institute, Charlotte, North Carolina, 1993.

Duration: from: 1992 to: 1995

Funding: N/A

Status: In progress

Last Update: October 4, 1993

FRANCE R-393

UTILITY DECONTAMINATION EXPERIENCE

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; EMMA

Principal Investigator:

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Project Manager:

Objectives: Develop the EMMA decontamination process used in France.

Comments:

- To apply it on stainless steel, the EMMA process uses 2 cycles: an oxidizing step (15 hrs) and a reducing step (5 hrs) at a temperature of 80°C.
- The oxidizing solution is a mixture of $KMnO_4$ (0.1% wt), nitric acid (0.013% wt) and sulfuric acid (0.005% wt).
- The reducing solution contains 0.1% wt ascorbic acid and 0.05% wt of citric acid.
- When conditions permit, ultrasound is applied during the treatment.
- The process is effective and relatively easy to implement (stable chemical products, operation at atmospheric pressure).
- It has been used since 1989 for the decontamination of primary pump hydraulic systems.
- Dose rate reduction factor ranges from 6 to 20.

Remarks/Potential for dose limitation:

References: Dupin, M., "Utility Decontamination Experience," *Fifth Workshop on Chemical Decontamination*, pp. 14.1-14.4, Electric Power Research Institute, Charlotte, North Carolina, 1993.

Duration: from: 1989 to: 1993

Funding: N/A

Status: Completed

Last Update: October 6, 1993

SWEDEN R-394

ABB ATOM PROJECT - ALARA 2000

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; EXPOSURE REDUCTION; FUEL FAILURE; EVALUATION OF DOSE RATE; NUCLIDE SPECIFIC MEASUREMENT; CORROSION PRODUCT MASSBALANCE

Principal Investigator:

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Project Manager:

Tor Ingemansson

Objectives: To reduce the annual exposures to 50% of the 1993 level. Three to five parameters that dominate the annual exposure shall be identified. Suggestions to plant specific measures for reducing the dose rates shall be provided to the outages in 1994.

Comments: The project is ordered by TVO, Oskarshamn 1, 2 and 3, Barseback 1 and 2, and Ringhals 1. The motivation for the project are increasing annual exposures. The project is separated into three parts: mapping of the future expected radiological conditions, mass balances of corrosion products, and evaluation of dose rate and nuclide specific in-site measurements.

Remarks/Potential for dose limitation: The effect of fuel failures on the radiation dose rates in a plant will be estimated. Different types of failures will be treated.

References: A presentation of the ABB Atom ALARA projects will be given at the "International Workshop on Implementation of ALARA at Nuclear Power Plants," to be held on Long Island, New York, May 1994. Proceedings will be available from the NRC, the BNL ALARA Center, and the National Technical Information Center after the workshop.

Duration: from: 1993 to: 1995

Funding: 2.5 py

Status: In progress

Last Update: December 9, 1993

U.S.A. R-395

DEVELOPMENT AND USE OF AN IN-PILE LOOP FOR BWR CHEMISTRY STUDIES

Keywords: CONTAMINATION PREVENTION; COMPONENT RELIABILITY; WATER CHEMISTRY; NITROGEN-16; ELECTROCHEMICAL POTENTIALS

Principal Investigator:

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Objectives: Measure and evaluate changes in radiolysis product generation as well as electrochemical corrosion potential (ECP) levels and nitrogen-16 (N-16) behavior as a function of BWR core coolant inlet water chemistry.

Comments: An in-pile loop has been successfully constructed and operated to simulate BWR coolant chemistry conditions. Results from a series of runs demonstrated the effects of a wide variety of organic and inorganic chemical additions on radiolysis product generation, N-16 behavior, and ECP. These results were consistent with similar measurements in the reactor coolant of full-scale BWRs, providing data useful in building more reliable predictive models to explain actual plant experience.

Water radiolysis products, such as oxygen and hydrogen peroxide, and their effect on ECP play an important role in the stress corrosion cracking of reactor plant materials and the behavior of important species such as N-16. The purpose of this study was to obtain quantitative cause and effect data important to the evaluation of BWR chemistry conditions, both inside and outside current guideline values.

Remarks/Potential for dose limitation: The observed effects of changing from normal water chemistry to hydrogen water chemistry included reduced oxygen and hydrogen peroxide concentrations, more negative ECPs, and increased N-16 carry-over in the steam phase. Organic additives and hydrogen had similar effects, increasing N-16 steam phase activity by a factor of approximately five while lowering ECP to levels that protect against stress corrosion cracking (below -230 mV). However, measured concentrations of O_2, H_2, and H_2O_2 were higher by a factor of two or more than those calculated using the MIT radiolysis code. This result is of potential significance for peroxide generation where BWR plant data are lacking. Molybdate was identified as a promising additive, capable of significantly lowering ECP without increasing N-16 carry-over. Improvements were identified for future runs to measure H_2O_2 and determine ECP more accurately.

References: "Development and Use of an In-Pile Loop for BWR Chemistry Studies," *EPRI TR-102248 Final Report*, Electric Power Research Institute, Palo Alto, CA, September 1993.

Duration: from: 1992 to: 1993

Funding: N/A

Status: Completed

Last Update: December 13, 1993

REPLACEMENT OF PINS & ROLLERS IN IRRADIATED BWR CONTROL BLADES

Keywords: REMOTE SYSTEMS; PINS AND ROLLERS; COBALT REDUCTION

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Objectives: Design, fabricate, and demonstrate remotely operated equipment that could be used in the spent-fuel pool at BWR sites and would remove the upper pins and rollers in irradiated control blades, replacing them with stainless steel buttons.

Comments: Equipment was designed and tested on unirradiated control blades at ABB Combustion Engineering's facility in Windsor, Connecticut. Modifications were made to accommodate various control blade designs, and the system was then tested at Commonwealth Edison Company's LaSalle Station.

The equipment uses electrical discharge machining (EDM) to remove the upper roller, most of the pin, and a small portion of the control blade. All cut material is collected in a waste container. Fine dust from the EDM operation is collected on a filter. The cut surface is brushed, finished, and inspected. Two stainless steel button halves are remotely installed and riveted together. Their installation is checked with a torque test. A test on unirradiated control blades led to minor modifications to the equipment and was followed by a test on an irradiated control blade at the LaSalle Station.

Remarks/Potential for dose limitation: The release of cobalt from the pins and rollers has been calculated to be responsible for up to 40% of shutdown radiation fields. Thus, a strong incentive exists to develop equipment capable of removing this significant cobalt inventory. Computer simulations has shown the possibility of a 20% to 40% field reduction.

References:

"Replacement of Pins and Rollers in Irradiated BWR Control Blades," *EPRI TR-101837s Vol. 1*, Electric Power Research Institute, Palo Alto, CA, February 1993.

Ruoss, C., "Replacement of Pins & Rollers in Irradiated BWR Control Blades," *Radiation Field Control Seminar*, Electric Power Research Institute, Seattle, Washington, 1993.

Duration: from: 1991 to: 1993

Funding: N/A

Status: In progress

Last Update: March 18, 1994

COBALT SOURCE REDUCTION - CONTROL ROD PIN & ROLLER REPLACEMENT

Keywords: COMPONENT RELIABILITY; COBALT REDUCTION; PINS AND ROLLERS; CONTROL ROD

Principal Investigator:

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Project Manager:

Objectives: Describe the General Electric program for replacing the irradiated control rod pins & rollers as a mean of reducing cobalt sources from the reactor core.

Comments:

Design:

- No EDM or welding
- Use spacer pads
- Top pins & rollers only
- Minimize implementation time

Process:

- Remove roller with hydraulic punch
- Install posi-lock spacer pad

Design of Spacer Pads:

- Two piece self locking device
- Primary retention: 7/16 inch thread
- Secondary retention: snap ring
- Material: inconel X-750 (same as current rollers)

Attributes of Spacer Pads:

- Positive self-locking device
- Retains pin segments
- Minimizes primary water circulation around pin segments
- Sized to be compatible with standard and GE 10 channels

Remarks/Potential for dose limitation:

Advantages:

- Eliminates majority of Co source
- Minimizes risk of damage to CR
- Minimizes risk of pool contamination
- Minimizes amount of waste: no EDM Swarf
- Simple/fast process
- Minimum effect on pool space

Status of program:

- BWR 2-4 (D-lattice) spacer pad complete
- Spacer pad qualification testing completed
- Spacer pad safety evaluation complete
- BWR 2-4 (D-lattice) tooling design and qualification complete
- Site demonstration successfully completed at KKM - March 1993
- C-lattice tooling design and qualification in progress
- Initial production at KKM August 1993

References: Cearley, J.E., "Cobalt Source Reduction - Control Rod Pin & Roller Replacement," *Radiation Field Control Seminar*, Electric Power Research Institute, Seattle, Washington, 1993.

Duration: **from:** 1992 **to:** 1993

Funding: N/A

Status: In progress

Last Update: January 3, 1994

PERFORMANCE OF IRON-BASE HARDFACING ALLOYS IN GATE VALVES TESTED UNDER SIMULATED BWR CHEMISTRY CONDITIONS

Keywords: COMPONENT RELIABILITY; HARDFACING ALLOYS; EVERIT 50; NOREM; STELLITE; COBALT REDUCTION; MATERIALS

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Project Manager:
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Phone: 415-855-2055

Objectives: Determine the behavior of several hardfacing alloys under autoclave BWR conditions.

Comments:

1) Both the PWR and the BWR phases of the valve hard facing testing program are completed.
2) The testing program has demonstrated the welding practicality of the alloys based on deposit hardness and chemical composition.
3) All alloys tested in the BWR phase had zero cold and hot leakage, with the exception of the valve with NOREM 04 which showed a persistent hot leakage. However, based on the various examinations it was concluded the hot leakage was the result of a fit-up problem rather than a deficiency of the deposit.
4) All alloys tested had similar resistance to sliding wear damage and galling which was comparable to that of the STELLITE 6 control standard.
5) With the exception of EB 5183, the candidate alloys had equal or superior corrosion resistance to STELLITE 6. EB 5183, however, was susceptible to pitting attack and therefore not suitable for applications in BWR primary circuits.
6) EVERIT 50, NOREM 01, and NOREM 04 meet or surpass the performance of the STELLITE 6 standard with respect to corrosion and material loss due to wear and maintenance of the valves sealing function. They have met the acceptance criterion established for this program and can be considered to be acceptable alternatives to STELLITE 6 for BWR valve hardfacing applications.

Remarks/Potential for dose limitation:

References: Murphy, E.V. and Inglis, I., "Performance of Iron-Base Hardfacing Alloys in Gate Valves Tested Under Simulated BWR Chemistry Conditions," *Radiation Field Control Seminar*, Electric Power Research Institute, Seattle, Washington, 1993.

Duration: from: 1992 to: 1993

Funding: N/A

Status: Completed

Last Update: January 4, 1994

NOREM WEAR-RESISTANT ALLOYS: AN EPRI PROGRAM UPDATE

Keywords: COMPONENT RELIABILITY; NOREM; WEAR-RESISTANT ALLOY; HARDFACING ALLOY; COBALT REDUCTION

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Objectives:
1) Reduce major source of background radiation
2) Provide utilities with a cobalt-free hardfacing alloy which exhibits wear & corrosion properties equivalent to Alloy 6
3) Develop welding product forms suitable for in-situ application

Comments:
Existing NOREM Product Forms:
- 0.045 diameter solid & metal-cored wires for automatic GTAW deposition
- 3/32 & 1/8 diameter rod for manual GTAW deposition
- gas atomized powder for PTAW & HIP sintering

NOREM Weldability Program - Project Goals:
- Develop & successfully deposit NOREM with the GTAW, GMAW, & FCAW processes
- Demonstrate multi-layer crack-free welds on carbon steel & stainless steel substrates which yield wear properties equal to Stellite 6
- Develop welding parameters which require no preheat
- Demonstrate a localized repair employing NOREM

NOREM GTA Weldability (B1) Current Status:
- Successfully demonstrated on stainless steel at ambient temperature
- Successfully demonstrated on wrought carbon steel with 309 Butter and with 200°F preheat
- Successfully demonstrated repairs on SS & CS with 200°F preheat

Remarks/Potential for dose limitation:
NOREM PTA Weldability Current Status:
- B1 & B4 successfully demonstrated on SS at ambient temperature
- B1 & B4 successfully demonstrated on wrought CS with 800°F preheat
- B1 successfully demonstrated on cast CS with 309 Butter layer and 800°F preheat

NOREM Weldability Program 1993 Milestones:
- Develop a chemistry which can be deposited without preheat or a Butter layer on CS with the automatic GTAW process
- Establish utility advisory committee

- Support Dragon Valve & Union Electric with a "first use" application
- Provide technical support to EPRI membership
- Conduct a NOREM demonstration workshop

References: Phillips, M., "NOREM Wear-resistant Alloys: An EPRI Program Update," *Radiation Field Control Seminar*, Electric Power Research Institute, Seattle, Washington, 1993.

Duration: from: 1992 to: 1995

Funding: N/A

Status: In progress

Last Update: January 4, 1993

CANADA R-400

THE EFFECT OF ZINC ON CARBON STEEL AND STAINLESS STEEL IN LITHIATED COOLANT

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; ZINC ADDITION; WATER CHEMISTRY; CORROSION; ZINC

Principal Investigator:
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Project Manager:

Objectives: Study the effects of zinc on carbon and stainless steel in a nuclear power plant.

Comments:

Potential Benefits of Zinc Addition:
- Lower rates of Co-60 buildup
    - lower dose rates; reduced man-rem
- Reduced corrosion rates
    - thinner oxides; less ion exchange resin required during a decontamination
    - less activity transport
- Decontamination performance
    - 410 SS was decontaminated without an oxidizing pretreatment

Comparison With PWR Results - Corrosion Rate Reduction:

Westinghouse (Esposito et al., 1991)
- zinc borate 50 ppb
- corrosion rate reduction by factor of
    - 3 on 304 and 316 SS
    - 1.7 to 11 on nickel based alloys

AECL
- corrosion rate reduction by factors of
    - 3 on carbon steel
    - 8 on 410 SS

Remarks/Potential for dose limitation: When 15 to 60 ppb of zinc was added to lithiated coolant:
- Corrosion was reduced by a factor of 3 on carbon steel, and 8 on 410 SS
- Corrosion-product release was reduced by a factor of appro. 18 on both CS and 410 SS
- The affinity for Co-60 was reduced by a factor of 3 to 5 for CS and appro. 60 on 410 SS
- 410 SS was effectively decontaminated using CAN-DECON without an oxidizing pretreatment
- Similar decontamination effectiveness was obtained with CAN-DECON on CS with and without zinc

References: Allsop, H., Godin, M., and Miller, D., "The Effect Of Zinc On Carbon Steel and Stainless Steel In Lithiated Coolant," *Radiation Field Control Seminar*, Electric Power Research Institute, Seattle, Washington, 1993.

Duration: from: 1992 to: 1993

Funding: N/A

Status: Completed

Last Update: January 4, 1994

OPTIMUM WATER CHEMISTRY IN RADIATION FIELD BUILDUP CONTROL

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; WATER CHEMISTRY; ZINC ADDITION; HYDROGEN WATER CHEMISTRY; COBALT REDUCTION

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Project Manager:

Objectives: Provide a summary of the latest water chemistry techniques in reducing radiation fields in a BWR.

Comments:

Countermeasures for Radiation Field Buildup:
- Cobalt source reduction
- Feedwater Fe reduction/control: controlling Co-60 transport
- Ionic impurity reduction: reducing material corrosion
- Depleted zinc oxide (DZO) addition: reducing Co-60 in water
- Decontamination: removing radioactivities on piping
- Prefilming: reducing initial activity buildup
- HWC: reducing material corrosion

Effects of Zn Addition in Radiation Buildup Control:
- Zn (depleted in Zn-64) is recommended to avoid Zn-65 production
- Lab data confirm that Zn at 5-10 ppb in water slows down the corrosion rate and reduces Co-60 deposition on steel surface
- Reactor data show Zn also reduces the Co-60 release rate from fuel deposit, resulting in lower Co-60 concentration in water

Effects of HWC on Radiation Field Buildup:
- Shutdown dose rate increase due to increased Co-60 deposition
- Magnitude of effect different among HWC plants
- Soluble and filterable Co-60 concentrations vary among plants
- Dose rate buildup varies among plants, from none to substantial
- At some plants with both HWC and GEZIP, shutdown dose rate increase was dominated by Zn-65

Remarks/Potential for dose limitation:

Background:
- Shutdown radiation fields in BWRs have been reduced significantly in recent years, and average personnel exposure continues to decrease slowly

- Personnel exposure goals continue to be lowered
- Further reduction in radiation field by water chemistry control is a formidable challenge
- Plant specific strategies aimed at controlling radiation field buildup and minimizing personnel exposure have to be developed and implemented

Summary and Conclusion:
- The concept of optimum water chemistry can be realized in radiation field reduction
- Co/Co-60 model calculation is helpful to define effective approaches to control and reduce radiation field buildup
- Effects of HWC on radiation field buildup have been clearly observed, but the magnitude may be minimized with source term reduction and proper operational procedure

References: Lin, C.C., "Optimum Water Chemistry in Radiation Field Buildup Control," *Radiation Field Control Seminar*, Electric Power Research Institute, Seattle, Washington, 1993.

Duration: from: 1992 to: 1993

Funding: N/A

Status: Completed

Last Update: January 5, 1994

U.S.A. R-402

STATUS REPORT ON BWR FULL SYSTEM DECONTAMINATION

Keywords: CONTAMINATION REMOVAL; LOMI; FULL SYSTEM DECONTAMINATION; DECONTAMINATION

Principal Investigator:

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Project Manager:

Objectives: BWR FSD Program Objective - initial evaluation of the feasibility of a full system decontamination (FSD) of a BWR (with fuel removed) using the LOMI process with specific emphasis on:
1) Long term materials compatibility and performance
2) LOMI process review and system interaction/flow paths
3) Radwaste management
4) Recontamination and dose savings
5) Cost benefit analysis
6) Safety review and licensing

Comments: Summary of Corrosion Data on LOMI, AP/LOMI, and NP/LOMI:

LOMI
- No BWR plant or fuel material has been identified to be incompatible with LOMI
- LOMI does not cause IGA or IGSCC, nor does it exacerbate existing IGA or IGSCC
- LOMI does not effect IGSCC UT detectability

AP/LOMI
- No crack extension in precracked stainless clad low alloy steel
- No corrosion of irradiated fuel materials
- No IGA or IGSCC in stainless steel or nickel-base alloys
- Cracking/corrosion of Cr plated parts
- Westinghouse study in AP-LOMI-AP-LOMI at higher temperatures, concentrations, and times:
  - IGA on 17-4 pH and type 410 SS
  - shallow pitting of many alloys

NP/LOMI
- Crack extension in pre-cracked stainless clad low alloy steel
- IGA on welded 316L/321, FS A600 U-bends
- IGA of FS 304
- Enhanced crack growth (~30x) in SA508-2

U.S.A. R-402

Remarks/Potential for dose limitation: GE Nuclear Energy Engineering Positions on LOMI and Oxidation Step Options:

LOMI - The risks associated with the application of LOMI for FSD of BWRs with the fuel removed appears to be very low provided that the process is applied in accordance with the approved process specifications.

AP - The application of AP for FSD of BWRs appears promising. However, additional testing is required before AP can be approved for FSD. For non-FSD applications, AP should be evaluated on a case by case basis.

NP - The application of NP for FSD of BWRs is not acceptable. For non-FSD applications, NP should be evaluated on a case by case basis.

References: Gordon, B.M., "Status Report on BWR Full System Decontamination," *Radiation Field Control Seminar,* Electric Power Research Institute, Seattle, Washington, 1993.

Duration: from: 1992 to: 1993

Funding: N/A

Status: Completed

Last Update: January 5, 1994

ACTIVITY PICKUP BY COATED COUPONS EXPOSED IN THE DOEL REACTOR

Keywords: CONTAMINATION PREVENTION; SURFACE PRETREATMENTS; SURFACE CONDITIONING; ELECTROPOLISHING; PREFILMING; PREOXIDATION

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Objectives: Study various surface pretreatment techniques for reducing out-of-core radiation level buildup.

Comments: Over the past several years, a cooperative program between EPRI and the Belgian utility has been underway to study various surface pretreatment techniques for reducing out-of-core radiation level buildup. The program involves exposure of pretreated coupons made from materials similar to those used in the construction of PWR primary systems. After exposure in the plant for a fuel cycle, the coupons are analyzed to determine the level of deposited activation corrosion products.

Remarks/Potential for dose limitation: Early test results showed a distinct advantage in first electropolishing a surface to reduce true surface area and leave it in a clean, microscopically smooth state. After electropolishing, pre-filming the electropolished surfaces further reduces corrosion and deposition on the surface. Pre-filming methods involved applying thin films of inert materials (e.g., palladium) and pre-oxidation of the base metal itself after electropolishing. The electropolished and preoxidized surface greatly reduces the activity buildup in BWR plants and the initial tests showed similar results in the PWR primary system. The main reason for the corrosion and deposition resistance is believed to be due to the chromium enrichment of the oxide film grown. A technique of first applying a very thin chromium film and then incorporating this chromium into a protective oxide coating proved very successful at Doel. Reduction factors in activity buildup of up to 20 were observed. A second test was run wherein only a chromium film was tested (without the pre-oxidation or stabilizing step) which was also very encouraging. The stabilized chromium film has been applied to new coupons for longer term testing at Doel as well as to plant piping for full-scale evaluations.

References: Asay, R.H., "Activity Buildup by Coated Coupons Exposed in the Doel Reactor," *Radiation Field Control Seminar*, Electric Power Research Institute, Seattle, Washington, 1993.

Duration: from: 1991 to: 1993

Funding: N/A

Status: In progress

Last Update: January 12, 1994

LIST OF HEALTH PHYSICS TECHNOLOGY PROJECTS

NUMBER	TITLE
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H-176	INDIAN POINT 2 SUB-SYSTEM DECONTAMINATIONS
H-177	NATIONAL DEMONSTRATION OF A FULL RCS CHEMICAL DECONTAMINATION
H-178	PLANT E.I. HATCH CHEMICAL DECON 1991
H-179	CHEMICAL DECONTAMINATION OF THE RESIDUAL HEAT REMOVAL SYSTEM
H-180	ONTARIO HYDRO DECONTAMINATION EXPERIENCE
H-181	RESOURCE MANAGEMENT AS AN ALARA TOOL
H-182	PWR UPPER/LOWER INTERNALS SHIELD
H-183	INTERNAL DOSE, RESPIRATORY PROTECTION AND REVISED 10CFR20 AT DAVIS-BESSE NUCLEAR POWER STATION
H-184	ZION UNIT 2 CYCLE 12 SHUTDOWN AND EARLY BORATION RESULTS
H-185	HEALTH PHYSICS SERVICES ON THE PLATFORM AT SALEM USING ROMMRS
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H-187	AN AUTOMATED PROGRAM IMPLEMENTING NEW 10CFR20 REQUIREMENTS AT SOUTHERN NUCLEAR PLANTS
H-188	STEAM GENERATOR REPLACEMENT PROJECT AT NORTH ANNA POWER STATION
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H-191	S/G REPLACEMENT AT BEZNAU 1: EXPERIENCE AND RESULTS IN RADIOLOGICAL PROTECTION
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H-195	A TEAM APPROACH FOR THE MANAGEMENT OF RADIOACTIVE LIQUID EFFLUENTS
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H-197	RADIOLOGICAL ASSESSMENT OF DECOMMISSIONING AT FORT ST. VRAIN
H-198	PERSONNEL RADIATION EXPOSURE REDUCTION DURING REMOTE STUD HANDLING AT INDIAN POINT 2
H-199	REPLACEMENT OF RWCU PIPING WITH STATE-OF-THE-ART MATERIALS
H-200	EVALUATION OF ZIRCALOY FUEL CLAD OXIDATION AT MILLSTONE 3 PWR
H-201	AN UPDATE ON CHEMISTRY RELATED DOSE REDUCTION EFFORTS AT MILLSTONE NUCLEAR GENERATING STATIONS
H-202	SURFACE CHARACTERIZATION OF THE STEAM GENERATOR CHANNEL HEAD FOLLOWING MECHANICAL/ ELECTROPOLISHING AT MILLSTONE POINT UNIT 2
H-203	CHROMIUM TREATMENT OF RHR PIPING
H-204	ON-LINE MONITORING OF DOSE RATES AND SURFACE ACTIVITY DURING THE CYCLE 17 SHUTDOWN OF RINGHALS 2 1993
H-205	CORROSION PRODUCTS BEHAVIOR IN FRENCH PRESSURISED WATER REACTOR DURING SHUTDOWN OPERATION
H-206	PWR STARTUP AND SHUTDOWN CHEMISTRY GUIDELINES
H-207	UPDATE ON DOSE RATES IN SIEMENS-DESIGNED PWRS

Category Index for Health Physics Technology Projects

Project Manager Index for Health Physics Technology Projects

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A METHOD FOR OPTIMIZING THE USE OF RESPIRATORY PROTECTION IN RADIATION AREAS

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; RESPIRATORS; RESPIRATORY PROTECTION; RADIATION PROTECTION; INTERNAL DOSE; EXTERNAL DOSE; COST-BENEFIT ANALYSIS; ALARA

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Objectives: If decisions on the use of respiratory protection equipment are made solely on the basis of airborne radioactivity levels, total dose equivalent (the sum of external and internal exposures) may not be as low as reasonably achievable (ALARA).

The objective of this work was to develop an optimization methodology for deciding when to use respiratory protection equipment. The method should take into account the reduction in worker efficiency that results from the use of respirators (and the increase in external dose equivalent) as well as the costs associated both with using respirators and not using respirators (surveillance, bioassay, record keeping, etc.). It should also allow a range of values to be used for relative worker efficiency and the assumed cost of the detriment of a person-rem.

Comments: The method indicates that the decision on whether to use respirators should be based on the following:

- If the airborne radionuclide concentration is significant with respect to the external dose rate, respirators should generally be worn in order to minimize total effective dose equivalent. The exception is when the cost of wearing respirators is excessive.
- If the external dose rate is significant with respect to the airborne radionuclide concentration, respirators should generally not be worn so that the total effective dose equivalent is minimized. The exception is when wearing respirators significantly reduces administrative and bioassay costs.
- If it is costly to issue respirators, they should not be worn unless the airborne radionuclide concentration is significant with respect to the external dose rate.
- If it is costly to not issue respirators when a potential for inhalation of radioactive material exists, they should be worn unless the external dose rate is significant with respect to the airborne radionuclide concentration.

Remarks/Potential for dose limitation: Work in airborne radioactivity areas often requires the use of respiratory protection equipment. In many cases, the decision to require respiratory protection is based solely on the actual or potential airborne radioactivity concentration. At nuclear facilities in the United States, it is common practice to issue respirators if airborne radioactivity concentrations exceed specified action levels (25% of 10CFR20 Appendix B values) or if surface contamination levels are very high (i.e., potential airborne radioactivity). This practice is based primarily on two factors: (1) current U.S. Nuclear Regulatory Commission regulations discourage significant internal exposures, and (2) the use of respirators greatly reduces the potential for large accidental intakes. However, this ignores two other important factors that relate to the use of respiratory protection equipment -- external radiation levels and the costs of using (or not using) respirators.

References: Merwin, S.E. and J.B. Martin, "A Method for Optimizing the Use of Respiratory Protection in Radiation Areas," *Radiation Protection Management*, Vol. 6, pp. 64-71, January/February 1991.

Duration: **from:** 1990 **to:** 1991

Funding: N/A

Status: Completed

Last Update: May 6, 1992

INDIAN POINT 2 SUB-SYSTEM DECONTAMINATIONS

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; SUB-SYSTEM DECONTAMINATION; CAN-DEREM/AP; CITROX/AP; COBALT REMOVAL; RECONTAMINATION; INDIAN POINT 2

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Project Manager:

Objectives:

I. Analyze sub-system decontaminations experience at:
   A) Indian Point 2 residual heat removal system
   B) Indian Point 1 steam generators (SG)

II. Examine recontamination rates at Indian Point 2's Regenerative Heat Exchanger, which was first decontaminated in 1989.

Comments:

Indian Point 2 residual heat removal system:

- A five step Can-Derem/AP decontamination process was used.
- One unusual feature was that the Co-58 and Co-60 were not removed until the anion resin was used at the end of the process. The cation resin did not remove the cobalt.
- It may not be worth the cost to perform a 4th and 5th step, since 75% of the activity was removed after the 1st step.
- An auxiliary pump was installed that injected a 0.1 gpm flow rate into the residual heat removal system (RHR) pump seals during the decon in order to avoid pumping decon chemicals into or through the RHR seals.

Indian Point 1 steam generators (in safe store mode since 1974):

- Used a five step Citrox/AP decon process.
- The decon chemicals could be reused on the 2nd pair of SG after the 1st pair were decontaminated due to the regenerative nature of the Citrox process.
- To simplify matters, the electrical power to operate the decon equipment was obtained via a cable from a reactor coolant pump.
- The asbestos insulating material on the SGs were found to be a problem.

Remarks/Potential for dose limitation:

Indian Point 2:

- The average exposure rate was 105 mrem/hr before and 7.1 mrem/hr after the decon.
- Approximately 30 rem was avoided during the 1993 outage due to the decontamination.
- The recontamination rate is being tracked between now and the next outage in 1995.

Indian Point 1:

- On the first pair of SGs 39 curies of Co-60 were removed.
- On the second pair 49 curies were removed.
- The final decontamination factor was approximately 4.

References: Parry, John O. "Indian Point 2 Sub-System Decontaminations," *Fifth Workshop on Chemical Decontamination*, pp. 15.1-15.22, Electric Power Research Institute, Charlotte, North Carolina, 1993.

Duration: from: 1992 to: 1993

Funding: N/A

Status: Completed

Last Update: September 23, 1993

U.S.A. H-177

NATIONAL DEMONSTRATION OF A FULL RCS CHEMICAL DECONTAMINATION

Keywords: CONTAMINATION REMOVAL; FULL SYSTEM DECONTAMINATION; DECONTAMINATION; INDIAN POINT 2

Principal Investigator:

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Project Manager:

Objectives: Provide a status report on plans for a full system decontamination (FSD) of Indian Point 2 in 1995 and detail several tasks that Consolidated Edison is working on toward that goal.

Comments: In 1988 Consolidated Edison, EPRI, ESEERCO and nine other utilities began a qualification program on how to chemically decontaminate the entire Reactor Coolant System of a Westinghouse PWR. A FSD is planned for Indian Point 2 in 1995. The majority of the services for the decon will be performed by Pacific Nuclear and Westinghouse. Some of the areas that Con Ed is working on are:

1) BMI Seals - Obtain and train personnel to install new high pressure seals for use while the Bottom Mounted Instrumentation is removed during FSD
2) Clean Seal Injection for reactor coolant pump (RCP) Seal - Prevent plugging of the seal injection cartridge filter or damage to the RCP seals at high particulate levels during FSD
3) Develop a test plan to monitor every aspect of the FSD
4) Dead leg flushing - Identify areas where the piping may have to be flushed after the FSD.
5) Reactor Water Storage Tank Clean Up
6) Calculate and track decontamination factors and recontamination rates
7) Demobilization of decontamination equipment

Remarks/Potential for dose limitation: The plans to perform the first chemical decontamination of a PWR in early 1995 are on schedule. There are no major technical concerns with the implementation plan.

References: Parry, John O., "National Demonstration of a Full RCS Chemical Decontamination," *Fifth Workshop on Chemical Decontamination*, pp. 27.1-27.27, Electric Power Research Institute, Charlotte, North Carolina, 1993.

Duration: from: 1988 to: 1995

Funding: N/A

Status: In progress

Last Update: September 30, 1993

BNL ALARA Center Data Base

U.S.A. H-178

PLANT E.I. HATCH CHEMICAL DECON 1991

Keywords: CONTAMINATION REMOVAL; HATCH UNIT 1; SUB-SYSTEM DECONTAMINATION; DECONTAMINATION; RECIRCULATION SYSTEM

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Project Manager:

Same as Principal Investigator

Objectives: Outline the results of the 1991 Hatch Unit 1 Reactor Recirculation System (RRS) decontamination using AP/LOMI. Share the lessons learned from this decon operation.

Comments:

- The decon operation consisted of a two-step LOMI/LOMI application to the entire RRS, with an additional AP/LOMI application to the RRS discharge piping.
- Project duration from start of cavity drain to cavity refilled was 10 days.
- The process removed 28 kg of oxide and 72 curies of activity from the RRS. Dose rates in the immediate vicinity of the RRS were reduced by an average factor of 8.9.
- The average dose rates on recirc piping was reduced from >350 mr/hr to <40 mr/hr. Drywell dose rate went from 20.46 mr/hr previously to 12 mr/hr.
- The estimated man-rem savings for two outages post decon is >700 man-rem.
- The recontamination rate for 1 cycle was 50%

Remarks/Potential for dose limitation: Lessons learned from this decon:

1) Choose equipment that is easy to assemble
2) Maximum temperatures, flows and flow reversals will increase decon factors (DF)
3) Location of decon taps is important - injection points and flow paths directly effect the DF
4) Use pressure gauges for level indication as opposed to tygon tubing
5) Run recirculation as soon as possible after decon.
6) Tear down of decon equipment was slowed due to low spots containing water. More low point drains were needed.
7) Include procedure steps that allow repeat of steps, if needed.
8) Hang shielding as normally done.
9) Use dedicated health physics technicians.

References: Warren, W., "Plant E.I. Hatch Chemical Decon 1991," *Fifth Workshop on Chemical Decontamination*, pp. 9.1-9.14, Electric Power Research Institute, Charlotte, North Carolina, 1993.

Duration: from: 1991 to: 1991

Funding: N/A

Status: Completed

Last Update: October 4, 1993

CHEMICAL DECONTAMINATION OF THE RESIDUAL HEAT REMOVAL SYSTEM

Keywords: CONTAMINATION REMOVAL; SUB-SYSTEM DECONTAMINATION; DECONTAMINATION; CITROX; RESIDUAL HEAT REMOVAL SYSTEM; BRUNSWICK UNIT 1

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Project Manager:

Objectives: Outline the procedure and summarize the results of a recent chemical decontamination performed on the CP&L Brunswick Unit1 Residual Heat Removal (RHR) system using a one step CITROX process.

Comments: The CITROX solvent was chosen because the process is regenerative and determined to be more effective in removal of high levels of iron oxide formed on the carbon steel. The process is one of acidic dissolution and reductive dissolution with the metal ions being removed from solution by the cation resin.

The solvent was injected into each loop at the two four inch flanges located on the suction side of the RHR pumps. The heat exchangers were bypassed for approximately the first six hours of the decontamination because of concern about corrosion of the 70/30 Cu/Ni heat exchanger tubes. A flange was installed in the demineralized water bypass line around the RHR loop isolation valve for return of the chemicals back to the decon equipment skid. At the decon skid, the solvent flowed through the in-line filters ion exchange and the heaters, and was injected back into the RHR loop.

The expected corrosion rate for carbon steel is <1 micrometer/hr and for austenitic stainless steel and nickel based alloys <0.1 micrometers/hr. No measured data for 70/30 Cu/Ni is available.

Remarks/Potential for dose limitation: The following results were achieved: (1) 3.3 Curies (primarily from Co-60) were removed, (2) 152 lbs of iron oxide were removed, (3) Average Decontamination Factor = 26, (4) Average Dose Reduction Factor = 9. A total of 240 cubic feet of ion exchange resin was used during the decon process. The total cost of the RHR decon was $850,000 and 6.6 person-rem was expended to complete the project.

References: Kury, R., Bozeman, J. and Ferguson, J., "Chemical Decontamination of the Residual Heat Removal System," *Fifth Workshop on Chemical Decontamination*, pp. 12.1-12.16, Electric Power Research Institute, Charlotte, North Carolina, 1993.

Duration: from: 1992 to: 1993

Funding: N/A

Status: Completed

Last Update: October 5, 1993

ONTARIO HYDRO DECONTAMINATION EXPERIENCE

Keywords: CONTAMINATION REMOVAL; FULL SYSTEM DECONTAMINATION; DECONTAMINATION; CAN-DECON; CANDU

Principal Investigator:

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Project Manager:

Objectives: Describe the CAN-DECON process and results obtained from past decontaminations. Discuss some key issues of design and operation that must be addressed to ensure a successful decontamination.

Comments: The essence of CAN-DECON involves the addition of the organic acids, citric acid, oxalic acid, and EDTA, to the heavy water coolant to form a 0.08% solution. These reagents dissolve and complex the corrosion product layer and radioactivity deposited on the heat transport system internal surfaces. The resultant solution is circulated through strong acid cation ion exchange resins in the purification circuit to remove the complexed metals and regenerate the reagents.

The success of decontaminations is dependent not only on the chemistry of the process itself, but also on the reliability of the decontamination purification system. A well designed system together with thorough commissioning are essential.

Training of support staff is a crucial area that cannot be overlooked. Both the needs of operating and chemical advisor staff must be addressed.

Remarks/Potential for dose limitation: Most of the Candecon decontaminations have been applied in support of reactor pressure tube maintenance. For Pickering Unit 3, the radiation dose saved is estimated at 1100 Rem for the decontamination carried out at the time of the reactor retubing outage. However, the savings increase to 4500 Rem when an earlier decontamination in support of continued operation is also considered. For Pickering Unit 4, the savings are estimated at 600 Rem, and 1850 Rem considering an earlier decontamination. For the retubing outages in these two units, maintenance personnel have been able to complete the maintenance substantially within the 2 Rem annual dose limit.

References: Lacy, C.S., "Ontario Hydro Decontamination Experience," *Fifth Workshop on Chemical Decontamination*, pp. 13.1-13.6, Electric Power Research Institute, Charlotte, North Carolina, 1993.

Duration: from: 1975 to: 1993

Funding: N/A

Status: In progress

Last Update: October 6, 1993

RESOURCE MANAGEMENT AS AN ALARA TOOL

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; RESOURCE MANAGEMENT; ALARA; SALEM UNIT 2

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Project Manager:

Objectives: Examine the 7th Salem Generating Station Unit 2 refueling outage results.

Comments: The outage results are:

1) 100.028 person-rem
2) 129,734 RCA hours
3) 0.77 mR/RCA hour (effective dose rate)
4) 45 personnel contaminations
5) 45 respirators issued
6) 14.5 cubic meters of radioactive waste

Remarks/Potential for dose limitation: The Resource Tools created a "best ever outage" :

1) Westinghouse integrated outage package
2) Refueling outage schedule
3) Contractor Returnee's
4) "In field" work methods
5) PSP/DCP radiation protection reviews
6) Work order to ALARA review link

"In field" work methods:

1) Valve maintenance program
2) Radwaste reduction
3) Temporary shielding
4) Radiation protection technician utilization

References: Palma, R.A., "Resource Management as an ALARA Tool," *1993 Radiation Exposure Management Seminar*, Westinghouse, Pittsburgh, Pennsylvania, 1993.

Duration: from: 1993 to: 1993

Funding: N/A

Status: Completed

Last Update: November 4, 1993

PWR UPPER/LOWER INTERNALS SHIELD

Keywords: RADIATION SHIELDING; REACTOR INTERNALS; REFUELING; INDIAN POINT 2

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Objectives: The goal of this research and development program was to design, develop, test and demonstrate a shielding system which would use the existing mass of the refueling pool water to provide shielding from the protruding components of the upper internals in order to reduce the radiation exposure of refueling personnel in containment.

Comments: During refueling of a nuclear power plant, the reactor upper internals must be removed from the reactor vessel to permit transfer of the fuel. The upper internals are stored in the flooded reactor cavity. Refueling personnel typically receive radiation exposure from a portion of the highly contaminated upper internals package which extends above the normal water level of the refueling pool. At Con Edison's Indian Point 2 plant, a method of shielding was devised which would use a vacuum pump to draw refueling pool water into an inverted canister suspended over the upper internals to provide shielding from the normally exposed components. The shield system consists of a 72" high cylindrical tank with an open bottom that is suspended from outside the cavity by two I-beams. The tank is positioned to provide 18" of immersion in the existing pool water. After installation most of the air trapped in the upper 54" of the tank is evacuated, and the vacuum draws water from the pool which fills the tank above the pool level. The "stand pipe" of water in the tank encircles the upper internals thereby providing the needed shielding.

Remarks/Potential for dose limitation: The development of the vacuum radiation shielding system resulted in significantly reduced dose rates to personnel. General area dose rates to refueling bridge personnel were reduced from 154 mR/hr to 25 mR/hr. Fourteen person-rem of exposure were saved as compared to the 1991 refueling outage. At 10,000/person-rem, the net savings for Con Edison is approximately $140,000 per use.

References: Homyk, W.A., "PWR Upper/Lower Internals Shield," *1993 Radiation Exposure Management Seminar*, Westinghouse, Pittsburgh, Pennsylvania, 1993.

Duration: from: 1992 to: 1993

Funding: N/A

Status: Completed

Last Update: November 5, 1993

INTERNAL DOSE, RESPIRATORY PROTECTION AND REVISED 10CFR20 AT DAVIS-BESSE NUCLEAR POWER STATION

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; RESPIRATORY PROTECTION; RESPIRATORS; TOTAL EFFECTIVE DOSE EQUIVALENT; INTERNAL DOSE; DAVIS-BESSE

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Project Manager:

Objectives: Investigate limiting the use of respirators in order to reduce the Total Effective Dose Equivalent (TEDE) for workers by allowing a small internal dose. The goal is compliance with the provisions of Title 10, Part 20 of the Code of Federal Regulations.

Comments: The DBNPS program is based on the philosophy that engineering controls are the preferred method of limiting intake of radioactive material as long as this does not result in higher total doses. The principal steps of the Respiratory Protection program at DBNPS are:

1) Determination of the area in which the job will take place.
2) Determination of the Deep Dose Equivalent (DDE) rate in the area of concern.
3) Determination of the expected fractional Derived Air Concentration (DAC) in the area of concern.
4) Determination of the efficacy of respiratory protection of any type.
5) If the use of engineering controls is justified, then apply it.
6) If the resulting air concentration after the application of engineering controls is still greater than 25% of the listed DAC, then a determination of whether a respirator would result in a lower TEDE is required.
7) Health and safety considerations may limit respirator use in areas of high temperature or on scaffolding, etc.
8) If the airborne contamination conditions are not sufficiently known in advance, respirators are used.

Remarks/Potential for dose limitation: A net savings of nearly thirty man-rem was achieved by permitting an internal dose of less than 1.5 man-rem.

This cost of the savings consisted of increased training on the concept of controlling the TEDE rather than controlling each type of dose separately and independently.

Further means of dose reduction of a similar nature can be found in the areas of excess protective clothing leading to a decrease in worker efficiency and therefore to increased stay times in radiation fields.

Education is the key to dose reduction. The worker, and some of the RP personnel, must be made to understand that the cell does not know which direction the particle or photon is traveling.

References: Zibung, B., Greenwood, R.A., and Mason, T., "Internal Dose, Respiratory Protection and Revised 10CFR20 at Davis-Besse Nuclear Power Station," *1993 Radiation Exposure Management Seminar*, Westinghouse, Pittsburgh, Pennsylvania, 1993.

Duration: from: 1992 to: 1993

Funding: N/A

Status: Completed

Last Update: November 5, 1993

ZION UNIT 2 CYCLE 12 SHUTDOWN AND EARLY BORATION RESULTS

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; SHUTDOWN CHEMISTRY; BORATION; ZION UNIT 2

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Project Manager:

Objectives: The Commonwealth Edison (CECo) Zion unit 2 reactor performed early boration and hydrogen peroxide treatment for the cycle 12 refueling outage. This work summarizes the chemistry and radiological observations from Zion 2 cycle 12 and from the refueling (Z2R12) shutdown.

Comments: Some of the results are as follows:

Zion 2 cycle 12 mid-cycle outage 9-27-91:

* Outage extended beyond original end date to 11-12-91
* De-lithiated to < 0.1 ppm
* Borated to pre-determined shutdown margin (1381 ppm B max)
* Removed 127 curies Co-58 and 7.3 curies Co-60 in the 1st 6 days
* No nickel analyses requested (or performed)
* No radiological problems

Zion 2 cycle 12 mid-cycle outage 4-3-92:

* Outage extended beyond original end date to 6-19-92
* Cool to 175 F and maintain 175 F for 38 days
* 1 RCP operating and letdown demins aligned
* De-lithiated to < 0.1 ppm
* Borated to pre-determined shutdown margin (1273 ppm)
* Removed 1637 curies Co-58 and 34.8 curies Co-60
* Removed 4500-5000 grams nickel
* No radiological problems

Remarks/Potential for dose limitation:

Zion 2 cycle 12 end-of-cycle outage 11-12-92:

* De-lithiated to < 0.1 ppm
* Borated to > 2000 ppm within 12 hours
* Removed 822 curies Co-58 and 8.6 curies Co-60. Small removal compared to previous outages.
* Removed > 4200 grams nickel. Large removal compared to previous outages.

* Steam generator channel head center radiation fields decreased 60% from refueling outage 11 to refueling outage 12.

End-of-cycle outage exposure totals:

* Pre-outage estimate 619 person-rem
* Actual 268.8 person-rem
* 36.6 of 268.8 person-rem was emergent work

Summary:

* Mid-cycle outage cleanup and operating chemistry control combined to contribute to radiological improvements
* May provide justification for chemistry and temperature hold during outages for ALARA purposes.

References: Stern, S., "Zion Unit 2 Cycle 12: Shutdown and Early Boration Results," *1993 Radiation Exposure Management Seminar*, Westinghouse, Pittsburgh, Pennsylvania, 1993.

Duration: from: 1991 to: 1993 **Funding:** N/A

Status: Completed **Last Update:** November 24, 1993

HEALTH PHYSICS SERVICES ON THE PLATFORM AT SALEM USING ROMMRS

Keywords: REMOTE SYSTEMS; ROMMRS; ROSA; ROBOTICS; SALEM

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Project Manager:

Objectives: Describe the capabilities of the Remotely Operated Managed Maintenance Robotic System (ROMMRS) and its role in performing Health Physics (HP) tasks.

Comments: ROMMRS is a joint project of Public Service Electric and Gas Company and Westinghouse Electric. The system will perform the Health Physics tasks for the primary side services planned for Steam Generators 12 and 14 at the Salem Nuclear Power Plant. The tasks to be performed at this outage with the system include: visual inspections, radiation surveys, vacuuming, wipe downs, area swiping, air sampling and equipment handling.

The robot is the ROSA I arm, slightly modified to be controlled from the ROSA III control system. The arm has 6 degrees of freedom and can be positioned anywhere within the 6 foot reach volume. It is mounted on a mobile base that travels on a triangular track with 2 degrees of freedom.

The video capabilities include two pan, tilt and zoom (PTZ) cameras and two end point cameras. The gamma radiation surveys are performed with a pair of Ebberline detectors. Beta radiation surveys are performed with a RO-2 meter.

Remarks/Potential for dose limitation: The long range vision for ROMMRS is 2 fold;
1) Reduce the radiation exposure to personnel through:
   - Replacement or augmentation of HP services with robotics
   - Replacement or augmentation of containment support worker services with robotics
   - Track personnel exposure real time with the computer control system
   - Plan tasks using a combination of the survey data base and the 3-D control model
2) Increase the productivity of the containment maintenance services through:
   - Use of the 3-D simulation to provide time-motion studies to optimize the service
   - Use the 3-D simulation to optimize and refresh the personnel training
   - Use the data base of industry wide experience to optimize the contingency management
   - Closely couple health physics services and contract services to minimize conflicts

References: Cruickshank, H., "Health Physics Services on the Platform at Salem Using ROMMRS," *1993 Radiation Exposure Management Seminar*, Westinghouse, Pittsburgh, Pennsylvania, 1993.

Duration: from: 1993 to: 1994
Funding: N/A
Status: In progress
Last Update: November 29, 1993

CHEMICAL DECON OF SYSTEMS: RESULTS AND PROBLEMS

Keywords: CONTAMINATION REMOVAL; FULL SYSTEM DECONTAMINATION; SUB-SYSTEM DECONTAMINATION

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Project Manager:

Objectives:

1) Summarize the work done to date on the National Demonstration of a full system decontamination at Indian Point 2, scheduled for 1995.
2) Summarize the fuel decontamination qualification program at the V.C. Summer site.
3) Summarize the results of the Fuel-In Full RCS Chemical Decontamination Seminar conducted on July 1993.
4) Present recent Westinghouse subsystem decontamination experience.

Comments:

1) Phase 2 - Decon Process Qualification and Detailed Engineering Evaluation - of the National FSD Demonstration Program was completed in Jan. 1991. Waste Certification Program completed by Chem Nuclear Systems in 1993. Program is on schedule for FSD in 1995.
2) Fuel Decon Qualification Program was completed at V.C. Summer in 1993. It was found that the post-decon fuel cladding corrosion behavior was unaffected by decontamination processes.
3) All attending utilities at the Fuel-In Full RCS Chemical Decon Seminar expressed interest in establishing a comprehensive program leading to qualification of fuel-in FSD. Westinghouse will elicit additional support and report back.
4) Westinghouse has conducted 18 sub-system decon applications at 11 different plants from 8/91 to 3/93. Fourteen of the applications were performed at BWRs, the rest at PWRs. The processes used include LOMI, CAN-DEREM, CANDECON and CITROX.

Remarks/Potential for dose limitation: A successful FSD will have significant impact on efforts to reduce worker radiation field exposure. Westinghouse, in conjunction with several utilities and other organizations, are on schedule for a 1995 FSD demonstration.

References: Miller, P. and Schwartz, C., "Chemical Decon of Systems: Results and Problems," *1993 Radiation Exposure Management Seminar*, Westinghouse, Pittsburgh, Pennsylvania, 1993.

Duration: from: 1990 to: 1993

Funding: N/A

Status: In progress

Last Update: November 29, 1993

AN AUTOMATED PROGRAM IMPLEMENTING NEW 10CFR20 REQUIREMENTS AT SOUTHERN NUCLEAR PLANTS

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; 10CFR20; DOSIMETRY SOFTWARE; SOFTWARE

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Project Manager:

Objectives: Describe the development and implementation of an integrated software system developed by Canberra Nuclear for the three Southern Nuclear plants Farley, Hatch, and Vogtle in compliance with the new 10CFR20 regulations.

Comments: Approximately two and a half years ago Southern Nuclear Company (SNC) initiated a task force to investigate a common methodology for its three nuclear plants to implement the new 10CFR20 regulations. A contract was awarded to Canberra Nuclear to develop an integrated software system for this purpose.

There are two software subsystems and several major modules associated with them. Each module is either in use or undergoing final site testing at the plants. They are:

1) Health Physics Data System
   - HIS-20: Health Physics Information System
   - ABACOS Plus Whole Body Counting System

2) Counting and Chemistry Data System
   - CAS: Countroom Analysis System
   - EMS: Effluent Management System
   - CDM: Chemistry Data Mgt System

Remarks/Potential for dose limitation: The lessons learned from this project are:

1) Write a detailed specification on what you want and how you expect it to perform, with a payment schedule linked to satisfactory completion of key tasks.
2) Select knowledgeable personnel to work on the project and empower them with the authority to make decisions. Keep these people through the completion of the project.
3) Work closely with the vendor on every phase of the project.
4) Insist on a detailed vendor test plan and testing before you begin your testing.
5) Perform detailed testing at the vendor's factory and again at the site before acceptance.
6) Train users early enough to get them familiar with the new system.

References: Patton, P., "An Automated Program for Implementing New 10CFR20 Requirements at Southern Nuclear Plants," *1993 Radiation Exposure Management Seminar*, Westinghouse, Pittsburgh, Pennsylvania, 1993.

Duration: from: 1991 to: 1993

Funding: N/A

Status: In progress

Last Update: November 30, 1993

STEAM GENERATOR REPLACEMENT PROJECT AT NORTH ANNA POWER STATION

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; STEAM GENERATOR REPLACEMENT; NORTH ANNA

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Objectives: Summarize the steam generator replacement (SGR) project at North Anna Power Station.

Comments: Some of the problems that led to the S/G replacement were:
- intergranular and primary water stress corrosion cracking
- circumferential cracking
- tubesheet dose rates of 25 to 35 Rem/Hr
- extensive S/G inspection and maintenance scope expending from 80 to 200 man-rem per outage
- outage duration of 55 to 77 days due to S/G activity

Some of the practices contributing to the success of the project:
- early and effective planning
- successful mock up training program
- aggressive radiological protection measures
- extremely low respirator usage through effective HEPA
- ventilation and other engineering controls
- extensive use of remote tooling and robotics
- good engineering
- little rework or out of scope work

Remarks/Potential for dose limitation: The dosage data for the project are:
Projections: (1) 540 Rem for all work, (2) 482 Rem for SGR work only, (3) goal of 110 personnel contamination events for all work, (4) goal of 11,000 cubic ft of radwaste or less

Actual results: (1) 313 Rem for all work, (2) 240 Rem for SGR work only, (3) 59 personnel contamination events for all work, (4) 3,600 cubic ft of radwaste generated

References: Banks, T., "Steam Generator Replacement Project at North Anna Power Station," *1993 Radiation Exposure Management Seminar*, Westinghouse, Pittsburgh, Pennsylvania, 1993.

Duration: from: 1/93 to: 4/93

Funding: N/A

Status: Completed

Last Update: December 31, 1993

ALARA PROGRAMME MANAGEMENT AND ORGANIZATION IN EDF NUCLEAR POWER STATIONS

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; ALARA; EDF

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Project Manager:

Objectives: Implement the ALARA principle during the operational phases of EDF's installations, and especially during maintenance.

Comments: EDF has taken steps to implement ALARA at its plants. Issues considered are:
1) Implementation of the ALARA Process
   - integration of radioprotection into the power station management process
   - consider dosimetric indicators the same as other indicators concerning power station operation (maintenance cost, etc.)
   - the methodology applied is based on two levels: national and local; the fundamental principle is that of local responsibility and initiative
2) Longevity of the ALARA structures
   - The ALARA structures were implemented in late 1991, and it is too early to draw any conclusions concerning their longevity.
3) Incentives to apply the ALARA principle
   - EDF intends to direct motivation toward how to reduce exposure, not how to receive an aware; therefore, no special incentives are considered for EDF staff.
   - Contractual incentives are offered to outside contractors.

Remarks/Potential for dose limitation: The long-term production of nuclear powered electricity requires that two conditions be met: (1) acceptance by public opinion and (2) ability to provide kWh at the lowest cost. Radioprotection plays a part in the fulfillment of both.

Today, EDF is basing its action on the ALARA concept, which is difficult to implement as it is based on the involvement of many participants. However, it is also very motivating, as national and international experience is showing excellent results.

Improved quality = Lower doses, and therefore higher economic efficiency.

References: Perin, M., "ALARA Programme Management and Organization in EDF Nuclear Power Stations," *1993 Radiation Exposure Management Seminar*, Westinghouse, Pittsburgh, Pennsylvania, 1993.

Duration: from: 1990 to: 1994

Funding: N/A

Status: In progress

Last Update: December 1, 1993

ENHANCED RADIATION WORKER TRAINING AT JAMES A. FITZPATRICK NUCLEAR PLANT

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; RADIATION PROTECTION; WORKER TRAINING; ALARA; FITZPATRICK

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Objectives: Investigate the radiation protection problems at the New York Power Authority (NYPA) FitzPatrick Plant and steps taken to improve radiological performance.

Comments: The FitzPatrick Plant experienced a significant decline in radiological performance from early 1987 through early 1992. There were several serious radiological incidents involving unplanned exposure and extremity overexposure to workers. The radiation protection program was criticized by the NRC for (a) weak supervisory oversight, (b) poor adherence to procedures by the plant staff and (c) weak training. Some of the recurring radiation protection (RP) problems were:

1) Inadequate communication with Radiation Protection
2) Improper usage of HEPA ventilation
3) High Radiation Area key control
4) High Radiation Area boundary control
5) Unplanned spread of contamination
6) Lack of dosimetry in the plant Restricted Area

To address this situation, NYPA management instituted a Radiological Improvement Plan. The development of the Enhanced Radiation Worker (ERW) Training Program was one of many items contained in the plan.

Remarks/Potential for dose limitation: In Dec. 1991 a pilot session of the ERW program was conducted.

The instruction techniques include:

- lecture
- practical exercises in the simulated hot laboratory
- computer animation
- video taping of exercises
- facilitation of class critiques

The course contents include:

- basic radiation/contamination concepts
- radiation protection procedures and policies
- industry events

- work planning
- ALARA
- contamination control

During 1992, 350 plant workers were trained and there were significant improvements in radiological awareness and performance. The SALP (systematic assessment of licensee performance) rating was increased from 3 (adequate) to 2 (good).

References: McCarty, J. and Hamblin, J., "Enhanced Radiation Worker Training at James A. FitzPatrick Nuclear Plant," *1993 Radiation Exposure Management Seminar*, Westinghouse, Pittsburgh, Pennsylvania, 1993.

Duration: from: 1991 to: 1994

Funding: N/A

Status: In progress

Last Update: December 1, 1993

S/G REPLACEMENT AT BEZNAU 1: EXPERIENCE AND RESULTS IN RADIOLOGICAL PROTECTION

Keywords: COMPONENT RELIABILITY; STEAM GENERATOR REPLACEMENT; RADIOLOGICAL PROTECTION; BEZNAU

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Project Manager:

Objectives: Describe the preparation, implementation, and results of the steam generator replacement at the Swiss Beznau 1 364 MW Westinghouse reactor in the context of radiological protection.

Comments: The two steam generators and sections of the reactor coolant lines were replaced at Beznau 1 in Spring 1993. This replacement qualifies as a new record in two areas: the time required to complete the principal operation took only 44 days and the radiation exposure was not higher than 1100 mSv (110 man-rem).

This low dose was achieved by using proven techniques and methodology in radiological protection. Some of the most important techniques were:

- Detailed planning of the radiation protection measures and procedures
- Intensive personal training (mock-up training)
- Shielding (a total of 80 t of lead had been installed)
- Installation of special "radiation islands" in the containment by means of shielding to allow a low dose area for workers to discuss problems
- Mechanical decontamination of the piping ends

Remarks/Potential for dose limitation: The effective total radiation exposure was below the projected dose. Some reasons for this were: 1) fewer man-hr than expected for some tasks, 2) unexpected good results about the continuous modifications at the shielding during the replacement operations, 3) the installed "radiation islands" were often used, especially in the first phase of the replacement operation

References: Weidmann, U., "S/G Replacement at Beznau 1: Experience and Results in Radiological Protection," *1993 Radiation Exposure Management Seminar*, Westinghouse, Pittsburgh, Pennsylvania, 1993.

Duration: from: 1985 to: 1993

Funding: N/A

Status: Completed

Last Update: December 3, 1993

PARTNERS IN PERFORMANCE: AN ALARA PERSPECTIVE

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; ALARA

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Objectives: Present the benefits to a partnership between site and vendor ALARA.

Comments: "Partners in Performance" denotes: (1) An agreement to work together for a common goal, (2) An understanding that if one side benefits, the other side benefits, and (3) A decision that both the NSSS vendor and the utility are in this business together. RP and NSSS vendor cooperation leads to increased communication, decreased misperceptions, less stress and less exposure.

An effective means in improving communications between work groups is through the use of a liaison. This had always been done with the production group. Through lessons learned at a utility prior to working with the Diablo Canyon Power Plant (DCPP), the vendor decided to use an ALARA Engineer as the liaison. This proved to be very effective at this utility. It was decided that an RP liaison, an ALARA Engineer, would be utilized at DCPP. This formed the basis for ALARA Partners in Performance. DCPP's ALARA program and Vendor Radiation Exposure Management benefited greatly from this partnership.

Remarks/Potential for dose limitation: Benefits of an ALARA Partnership:
- It provides a "common thread" between utilities performing similar work scopes.
- It provides a single point contact for addressing radiological concerns.
- It provides a pathway for lessons learned to be followed for implementation by both parties.
- It enables the focus of effort to be on "real", not "perceived" problems.
- It insures support activities (ie, scaffolding) be implemented correctly the first time.
- It streamlines radiological control implementation, minimizing impediments to production duration.
- It benefits the utility by enabling planned work scopes to be completed using less schedule time.
- It benefits the vendor by reducing personnel exposures, thereby assuring extensive use of experienced personnel on subsequent jobs and improved job performance.
- It assures that ALARA is more than just hanging lead.

References: Helmen, C. and White, G., "Partners in Performance: An ALARA Perspective," *1993 Radiation Exposure Management Seminar*, Westinghouse, Pittsburgh, Pennsylvania, 1993.

Duration: from: 1993 to: 1994

Funding: N/A

Status: In progress

Last Update: December 3, 1993

STEAM GENERATOR SNUBBER ELIMINATION

Keywords: COMPONENT RELIABILITY; SNUBBER ELIMINATION; STEAM GENERATOR; WOLF CREEK; CALLAWAY

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Objectives: Implement the Westinghouse Steam Generator Snubber Elimination Program at the Wolf Creek and Callaway plants.

Comments: The continuing need to monitor, inspect, and periodically test the performance of the SG snubbers are costly in terms of exposure and man-hours. However, recent advancements in computer technology have made it economically feasible to perform the engineering analyses necessary to eliminate 100% of the SG snubbers in nuclear power plants.

The Westinghouse Steam Generator Snubber Elimination Program implements the load reduction techniques of Leak-Before-Break, Elimination of Arbitrary Intermediate Breaks, and ASME Code Case N-411 damping. The new approach now also includes non-linear time history seismic and pipe break analyses to more accurately predict the loadings on the reactor coolant system components.

The Wolf Creek Nuclear Operating Company & the Union Electric Company are embarking on a program to eliminate all 32 Steam Generator Large Bore Hydraulic Snubbers. The resulting savings in reduced man-rem exposure, maintenance, and inspection is significant. The payback period is estimated to be just two years.

Remarks/Potential for dose limitation: The benefits of SG snubber elimination identified by Wolf Creek and Union Electric are as follows. Similar benefits would be experienced at other nuclear stations.

- Achieves ALARA: exposure reduced by approximately 3 man-rem per outage.
- Eliminates visual inspection and functional testing: saves $35,000 per outage.
- Reduces the risk of outage extension and unplanned outages
- Reduces maintenance and refurbishment costs: saves $15,000 per outage
- Reduces outage activities
- Improves plant reliability by reducing congestion and snubber failures
- Increases plant availability

References: Stirzel, C., "Steam Generator Snubber Elimination," *1993 Radiation Exposure Management Seminar*, Westinghouse, Pittsburgh, Pennsylvania, 1993.

Duration: from: 1992 to: 1994

Funding: N/A

Status: In progress

Last Update: December 6, 1993

FUTURE POWER STATIONS IN THE UNITED KINGDOM: DESIGNING FOR LOW DOSES

Keywords: CONTAMINATION PREVENTION; DOSE REDUCTION; FUEL CYCLE; ZIRCALOY; COOLANT CHEMISTRY; STELLITE; SIZEWELL B

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Project Manager:

Objectives: Utilize the design experience from the Sizewell 'B' nuclear power station nearing completion to develop potential steps that can be taken to improve operator doses on future PWR plants in the U.K.

Comments: Nuclear Electric is currently building its first commercial PWR station, Sizewell 'B', in Suffolk, England. It is based upon the Westinghouse Standard Nuclear Unit Power Plant (SNUPPS). With Sizewell 'B' nearing completion, Nuclear Electric is already looking to the design of any further stations to be built in the UK. These future plants are likely to be Sizewell 'B' replicas but attention would still be given to reducing operator doses further.

An assessment of the operator doses for Sizewell 'B' concluded that for the planned 12 month fuel cycle the annual dose would be 1.97 man-Sv (197 man-Rem). The maximum individual dose was calculated to be 8.5 mSv (0.85 Rem). This was a conservative estimate because it did not reflect all the source reduction steps taken for the plant.

Remarks/Potential for dose limitation: The Sizewell 'B' design was effectively frozen in the mid to late 1980's. However, since then significant improvements has been made in the operation of PWRs and radioactive source reduction. The following are considered to be the most important features for consideration for future plants:

- Adoption of an 18 month fuel cycle	(97 man-Rem)
- Adoption of Zircaloy fuel grids	(76 man-Rem)
- Adoption of high pH chemistry	(57 man-Rem)
- Stellite removal	(35 man-Rem)

The items listed can provide both significant financial advantages and greater operational flexibility in achieving the low dose targets.

References: Willcock, A., "Future Power Stations in the United Kingdom: Designing for Low Doses," *1993 Radiation Exposure Management Seminar*, Westinghouse, Pittsburgh, Pennsylvania, 1993.

Duration: from: 1992 to: 1994 **Funding:** N/A

Status: In progress **Last Update:** December 8, 1993

A TEAM APPROACH FOR THE MANAGEMENT OF RADIOACTIVE LIQUID EFFLUENTS

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; CONTAMINATION REMOVAL; LIQUID EFFLUENTS; RADWASTE DISPOSAL; DAVIS-BESSE

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Project Manager:

Objectives: Streamline the management of liquid effluents at the Davis Besse Nuclear Power Station.

Comments: The management of liquid effluents at most PWRs is complex due to responsibility being shared by various departments. A team approach is necessary to provide effective control of all radioactive liquid effluents to reduce dose to the public and maintain processing costs as low as possible.

At Davis Besse, the administrative control over liquid radwaste processing has been particularly complex for various reasons. They have proposed to form a radiological effluent management team that will develop policy that coordinates plant activities and operations that generate, process and release radioactive liquid effluents. The proposed objectives of the team are: 1) Reduce the source term for radioactive effluents, 2) Minimize unnecessary liquid from entering the radioactive liquid waste processing system, 3) Determine the type of processing media to be utilized, 4) Perform routine review of the processing system, 5) Plan for infrequent events such as resin replacement or system flushing, and 6) Provide a communication route.

Remarks/Potential for dose limitation: Reducing liquid effluent activity requires a management culture which commits the cooperation and coordination of the entire station. Long term vision should be reflected in a written radwaste policy. Communication can affect "buy in" from station personnel. One has to address the source term. Small leaks need to be located and repaired. Modifications and work orders need to be reviewed to eliminate "clean" water from the radwaste stream. Vendors need to be pushed to continue to develop improved technology to achieve better activity removal factors.

References: Priest, J., "A Team Approach for the Management of Radioactive Liquid Effluents," *1993 Radiation Exposure Management Seminar*, Westinghouse, Pittsburgh, Pennsylvania, 1993.

Duration: from: 1992 to: 1994

Funding: N/A

Status: In progress

Last Update: December 13, 1993

ALARA AND WORK MANAGEMENT

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; ALARA; WORK MANAGEMENT; DOSE REDUCTION; EDF

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Project Manager:

Objectives: Examine the effects of work management on occupational exposures.

Comments: Within the framework of its ALARA program for the French nuclear power plants, EDF (Électricité de France) has initiated a pluri-annual research project conducted by CEPN in order to delineate the various factors related to work management which can influence occupational exposures and to evaluate the effectiveness of possible protection actions influencing these factors. Three different categories of factors have been defined: the factors linked to working conditions (ergonomic of work areas, adaptation of tools...), those characterizing the operators (qualification, experience level, motivation...) and the factors directly depending on the operations' organization (tasks planning, general preparation of works...). The results have been complemented by a survey carried out in five French nuclear power plants and focused on three types of operations: primary valves maintenance, decontamination of reactor cavity, and some specialized maintenance operations.

Remarks/Potential for dose limitation: The results of the first 2 years of study tend to confirm that a great potential for reducing exposures can be obtained through work management actions without spending vast sums of money. However, ALARA implementation needs the support of management, especially in the following stages:

1) Preparation of maintenance operations
   - integration of areas "design" into and procedures
   - planning by working area
   - prediction of exposure taking into account real dose rates
2) Follow up
   - implementation of systematic data collection network
   - quantification of doses due to mishaps
3) Feed back experience
   - use of databases to rapidly communicate lessons learned
   - comparison between dosimetric performances must be done by reference to a same level of ambient dose rate

References: Schieber, C., "ALARA and Work Management," *1993 Radiation Exposure Management Seminar*, Westinghouse, Pittsburgh, Pennsylvania, 1993.

Duration: from: 1991 to: 1994

Funding: N/A

Status: In progress

Last Update: December 14, 1993

RADIOLOGICAL ASSESSMENT OF DECOMMISSIONING AT FORT ST. VRAIN

Keywords: CONTAMINATION REMOVAL; DECOMMISSION; DECONTAMINATION; ALARA; FORT ST. VRAIN

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Objectives: Present the current status of the decommissioning effort at the Public Service Co. of Colorado (PSC) Fort St. Vrain (FSV) reactor.

Comments: The decontamination option was chosen over safe storage for decommissioning. The goal is to reduce the amount of residual radioactive material remaining on the Fort St. Vrain site to a level acceptable for the termination of the FSV operating license.

Fort St. Vrain Background Information:
- first and only of its kind in the US: 330 MWE HTGR reactor
- poor operational performance (< 15%)
- outstanding radiological performance (avg 3 person-rem/year)
- high cost, poor performance basis for planned 6/1990 shutdown
- equipment failures prompted premature shutdown in 8/1989

Scope of the FSV decon:
- decontamination and dismantlement of the prestressed concrete reactor vessel (PCRV) containing 90-95% of radioactivity
- decontamination and dismantlement of contaminated balance of plant systems with the remaining 5-10% of radioactivity
- site cleanup and final site release survey
- release of site for unrestricted use, likely as repowered electric generating facility

Remarks/Potential for dose limitation: Radiological Assessment:
- All project person-rem recorded exposures have been under the goals and estimates
- No positive internal body burdens have been recorded
- Workers have submitted over 100 exposure saving ideas with over 80 implemented
- Over 5,000 formal radiological classroom training hours have been conducted
- The project has conducted numerous job specific rad worker training classes to ensure workers understand the job task to be performed.
- No project NRC violations have been recorded.

Project ALARA Program includes:

1) aggressive shielding program
2) contamination control
3) mock up training program
4) radiological engineering controls
5) ALARA awareness programs
6) integrated work package reviews
7) communication usage
8) special instrumentation
9) video camera usage

References: Borst, T., "Radiological Assessment of Decommissioning at Fort St. Vrain," *1993 Radiation Exposure Management Seminar*, Westinghouse, Pittsburgh, Pennsylvania, 1993.

Duration: from: 1989 to: 1995

Funding: N/A

Status: In progress

Last Update: December 14, 1993

PERSONNEL RADIATION EXPOSURE REDUCTION DURING REMOTE STUD HANDLING AT INDIAN POINT 2

Keywords: REMOTE SYSTEMS; REMOTE STUD HANDLING; DOSE REDUCTION; INDIAN POINT 2; REACTOR VESSEL STUDS

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Objectives: Describe the usage of the remote study handling system during the February 1993 refueling outage at Indian Point 2.

Comments: The automated stud handling system employs a transport segment system to remove and install reactor vessel studs, nuts, and washers. The transport segment is a precision fabrication which sits on the reactor vessel covering 1/4 of the bolt circle. A track assembly on the top of the transport segment guides 2 automated stud turning robots. These two robots are connected to a central control panel which is located on the refueling deck.

Using a special lifting beam, the first transport segment along with the two automated stud turning robots is positioned on the reactor vessel head. Two operators are located in the cavity to align the transportation segment on the head. Once the transportation segment is properly positioned the stud removal process is initiated from the remote control panel on the refueling deck. The operators in the cavity detach the lifting beam assembly from the transportation segment and then move to a lower dose area during the stud removal sequence.

Remarks/Potential for dose limitation: Using the remotely operated stud system during refuel 11/12, which started in 2/1993, studs were removed in 6.5 hours with a total exposure of 498 mRem. Stud insertion was completed in 6.5 hours with 1045 mRem exposure.

Total outage exposure results using the remote stud handling system:

OUTAGE	MAN HOURS	DOSE RATE FIELD	CRITICAL PATH	EXPOSURE
1991	160.5	250 mRem/Hr	26.75 Hr	6.258 Rem
1993	65.00	250 mRem/Hr	13.00 Hr	1.543 Rem
% savings				
1993 vs 1991	59.6%	N/A	51.41%	75.35

References: Bishop, D., "Personnel Radiation Exposure Reduction During Remote Stud Handling at Indian Point 2," *1993 Radiation Exposure Management Seminar*, Westinghouse, Pittsburgh, Pennsylvania, 1993.

Duration: from: 1993 to: 1993

Funding: N/A

Status: Completed

Last Update: December 15, 1993

REPLACEMENT OF RWCU PIPING WITH STATE-OF-THE-ART MATERIALS

Keywords: CONTAMINATION REMOVAL; REACTOR WATER CLEANUP SYSTEM; SUB-SYSTEM DECONTAMINATION; PERRY

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Project Manager:

Objectives: Replace the reactor water cleanup system (RWCU) piping at the Perry Nuclear Power Plant and record the dose savings resulted from it.

Comments: Perry Nuclear Power Plant is a 1250 mW BWR. During the Second Refueling Outage in 1990, about 300 person-rem was expended for work in areas of the drywell adjacent to the RWCU cross-tie line. This 140 feet long, 4 inch diameter carbon steel line connected the two loops of the reactor recirculation system and the RPV bottom head drain line.

To reduce the radiation field in this area, a decision was made to:

1) chemically decontaminate the piping using LOMI
2) cut the pipe into sections that would easily fit into the B25 shipping container
3) replace the pipe along the same routing
4) replace the carbon steel with stainless steel
5) process the replacement pipe to mitigate recontamination, ie, electropolishing and passivation

347 NG stainless steel was selected as the replacement piping material because of:

1) IGSCC resistant per NUREG 0313
2) low cobalt
3) no increase in ISI requirements
4) good weldability
5) good availability
6) corrosion resistant

Remarks/Potential for dose limitation:
ALARA actions to minimize personnel dose:

* mock-up training
* chemical decon
* catch containments
* HEPA ventilation
* wireless communications
* closed circuit TVs
* The chemical decontamination skid was located in the fuel handling building away from traffic areas.
* shielded demineralizers and filters
* remotely operated tools
* low dose waiting areas

General area dose-rates were reduced by a factor of 5.2 (80.8%). A better general area DF would have been realized if the reactor recirculation system pumps and piping could have been included in the LOMI flowpath.

During the second refueling outage average shielded general area dose-rates in the drywell on the 583' and 599' elevation were 0.093 and 0.078 mR/hr. During the 1993 mid-cycle outage the average unshielded general area dose-rates were 0.064 and 0.034 mR/hr.

References: Kindred, G. W., "Replacement of RWCU Piping With State-of-the-Art Materials," *Radiation Field Control Seminar*, Electric Power Research Institute, Seattle, Washington, 1993.

Duration: from: 1990 to: 1993

Funding: N/A

Status: Completed

Last Update: December 21, 1993

EVALUATION OF ZIRCALOY FUEL CLAD OXIDATION AT MILLSTONE 3 PWR

Keywords: COMPONENT RELIABILITY; ZIRCALOY; MILLSTONE; FUEL CLADDING

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Objectives: Analyze plant data on Zircaloy-4 clad oxide thickness taken from Millstone 3 in comparison with data from North Anna 1.

Comments:
COOLANT LITHIUM EFFECTS ON ZIRCALOY-4 CLAD OXIDE THICKNESS:
- Summary of laboratory tests
  - No voidage, no B, 3.5 ppm Li --> 2% to 5% increase compared with 2.2 ppm Li
  - No voidage, >50 ppm B, 3.5 ppm Li --> negligible increase
  - High voidage, any B, any Li --> large increase
- Plants do not operate with high voidage: effect may be slight
- Need to check actual plant data, especially for thick oxides
- Hence EPRI/Westinghouse program at Millstone 3 and comparison plant, North Anna 1. EPRI contracted Nuclear Electric to analyse results.

SUMMARY OF OXIDE THICKNESS RESULTS:
- Millstone (Elev) 13% or 14%* higher than N Anna (Co-ord)
- Millstone One Cycle exposures (few rods only): D, E Assemblies (Elev) 29% or 42%* lower than A assemblies (Co-ord)
- Millstone Two Cycle exposures: D Assemblies (Elev) 33% or 36%* lower than B, C assemblies (Co-ord and Elev)
- All above results significant at >99% confidence level
- Hence inconsistent results on Li effect
- Inconsistency probably due to batch-to-batch differences in standard Zr-4 clad

* One value from slope of fit to measured vs predicted, the other from mean measured/predicted ratios

Remarks/Potential for dose limitation:
- Inconsistent apparent effect of Elevated Li Chemistry:
  - Millstone/N Anna, higher oxidation
  - Millstone/Millstone, lower oxidation

- Likely reason for inconsistency is batch-to-batch variation in cladding:
  - Composition
  - Annealing
- Cannot conclude from this study that Elevated Li enhanced Millstone oxide thicknesses
- Future work: measure oxide thicknesses after Millstone cycle 4 exposed to Co-ordinated pH(308) 6.9 chemistry

References:

Swan, T. and Polley, M.V., "Zircaloy Fuel Clad Oxidation at Millstone 3 PWR," *Radiation Field Control Seminar*, Electric Power Research Institute, Seattle, Washington, 1993.

Polley, M.V. and Evans, H.E., "A Comparison of Zircaloy Oxide Thicknesses on Millstone 3 and North Anna 1 PWR Fuel Cladding," *EPRI TR-102826 Report*, Electric Power Research Institute, Palo Alto, CA, 1993.

Duration: from: 1992 to: 1993 **Funding:** N/A

Status: Completed **Last Update:** January 4, 1994

AN UPDATE ON CHEMISTRY RELATED DOSE REDUCTION EFFORTS AT MILLSTONE NUCLEAR GENERATING STATIONS

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; CONTAMINATION PREVENTION; WATER CHEMISTRY; ZINC INJECTION; COBALT REDUCTION; MILLSTONE

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Project Manager:

Objectives: Describe the ALARA efforts at Northeast Utilities and the results from elevated pH, zinc injection, decontaminations, electropolishing, etc.

Comments: An aggressive chemistry and materials ALARA program was initiated at NE Utilities in 1986. It included:
- Elevated pH coolant chemistry control at MP3 (PWR)
- Zinc injection (GEZIP) at MP1 (BWR)
- Decontaminations, SG channel head electropolishing, and cobalt source removal

Millstone 3 End-Of-Cycle-3 Results (3.7 EFPY):
- Radiation fields remain low
- No evidence of adverse effects on I-600
- Continuing uncertainty about adverse effects of Li on high burnup zircaloy oxidation
- Standard coordinated pH 6.9 for cycle 4 and the future until
  - more information on cladding limitations
  - And/or fuel load with more corrosion resistant cladding

Millstone 1 Results After Decontamination and 2 Cycles of GEZIP:
- Recontamination dose rates leveled out at 50% less than without GEZIP
- Manageable side effects due to Zn-65 Production
  - dissolution during shutdown
  - food chain incorporation
  - personnel monitoring and detectability
  - extra waste curie load
  - zinc depleted in Zn-64 (GEZINC) is now being tested to reduce Zn-65 production

Remarks/Potential for dose limitation: Other Chemistry ALARA Efforts at NU:
- Subsystem and SG channel head decons
- Early boration during shutdowns
- Acid reducing nickel clean up during start up after SG replacement at Millstone 2
- Modified elevated pH coolant chemistry control
- Participation in PWR Zinc Injection Program

Summary

- Two cycles of operation with zinc injection at Millstone 1 proved successful in reducing worker radiation exposure.
- Two cycles of operation with elevated pH at Millstone 3 proved successful in controlling the increase of SG channel head dose rates. However, fuel cladding oxidation concerns have led to a temporary respite in this program.
- Activities such as decontamination, surface treatment, and cobalt source removal programs, in conjunction with the coolant chemistry programs, will continue to be evaluated as means to reduce occupational radiation exposure at NU nuclear plants.

References: Hudson, M.J., "An Update On Chemistry Related Dose Reduction Efforts at Millstone Nuclear Generating Stations," *Radiation Field Control Seminar*, Electric Power Research Institute, Seattle, Washington, 1993.

Duration: from: 1986 to: 1994

Funding: N/A

Status: In progress

Last Update: January 5, 1994

SURFACE CHARACTERIZATION OF THE STEAM GENERATOR CHANNEL HEAD FOLLOWING MECHANICAL/ ELECTROPOLISHING AT MILLSTONE POINT UNIT 2

Keywords: CONTAMINATION PREVENTION; SURFACE CONDITIONING; MECHANICAL POLISHING; ELECTROPOLISHING; MILLSTONE

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Objectives: The Surface Roughness Measurement Program for Millstone Point Unit 2 was developed to investigate the effects of mechanical polishing versus electropolishing with respect to surface roughness.

Comments: Under the auspices of EPRI, Westinghouse performed nondestructive surface roughness measurements on portions of the Millstone Point Unit 2 replacement steam generator channel heads. In preparation for service, Northeast Utilities performed pre-service surface conditioning of the replacement steam generator channel heads, namely, mechanical polishing/electropolishing. The results of this program will aid in determining whether a relationship exists between the surface roughness and activated corrosion product deposition.

Remarks/Potential for dose limitation: The data analysis of the surface roughness measurements performed at Millstone Point Unit 2 showed that the effectiveness of the electropolishing process on the divider plate was less than that on the channel head bowl and stay cylinder, which are weld overlay surfaces. This was due to the difference in the metallurgical surface structure of the weld overlay and the wrought 304L stainless steel divider plate. During electropolishing of the weld overlay surfaces, the ferrite stringers were preferentially removed as compared to the surrounding surface structure. In addition, the electropolishing process removed or reduced the larger asperities, i.e., metal folds and scratches, caused by the mechanical polishing. This resulted in a smoother surface with respect to the average roughness parameter, R_a. An evaluation of the roughness data showed that the electropolished weld overlay surfaces were smoother than the mechanically polished surfaces with a statistically significant confidence level (>= 99%).

References: Perock, J.D., "Surface Characterization of the Steam Generator Channel Head Following Mechanical/Electropolishing at Millstone Point Unit 2," *Radiation Field Control Seminar*, Electric Power Research Institute, Seattle, Washington, 1993.

Duration: from: 1992 to: 1993

Funding: N/A

Status: Completed

Last Update: January 12, 1994

CHROMIUM TREATMENT OF RHR PIPING

Keywords: CONTAMINATION PREVENTION; SURFACE PRETREATMENT; SURFACE CONDITIONING; RESIDUAL HEAT REMOVAL SYSTEM; DIABLO CANYON PLANT; CHROMIUM

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Objectives:

1) Investigate feasibility of using phosphoric acid based chromium plating bath to improve process application
2) Develop a range of acceptable chromium plating parameters for treating plant components
3) Evaluate effectiveness of surface pretreatments to mitigate activity buildup on out-of-core components
4) Scale-up stabilized chromium pretreatment process to full size plant components

Comments: With the positive results obtained in the Doel-2 coupon testing for surface pretreatment using chromium films, a full-scale application program was initiated. This program involved pretreating surfaces of RHR replacement piping at the Diablo Canyon Plant. This new stainless steel piping is 8-inch nominal and was installed during the 1993 refueling outage.

Both chromium and stabilized chromium films were applied to two different sections of replacement piping. A third section of the piping was left in the electropolished only condition while all the remaining pieces were electropolished and pre-oxidized. The electropolished section will serve as the "reference" surface. The pre-oxidation process is the RCT process as is typically applied to boiling water reactor primary system piping. Thus, there are four distinctly different surface treatments in the plant which will be monitored for activity buildup over the next two fuel cycles. Both gross gamma and gamma spectrographic measurements will be made at each opportunity to access the piping.

Remarks/Potential for dose limitation:

- Piping exposed to reactor coolant for only about 36 days at low temperature (~140°F)
- Gamma spectroscopy measurements made approximately 46 days after system was removed from service
- Co-58 was by far the predominant nuclide (>95%), with only trace levels of Co-60, Mn-54, and Fe-59
- Only small differences in activity buildup were observed due to the short exposure period
- General trend of activity buildup correlates with the various surface pretreatments (i.e., stabilized chromium pipe had the lowest buildup and the as-received pipe had the highest buildup)

References:

Asay, R.H., "Chromium Treatment of RHR Piping," *Radiation Field Control Seminar*, Electric Power Research Institute, Seattle, Washington, 1993.

Asay, R.H., "Activity Pickup by Coated Coupons Exposed in the Doel Reactor " *Radiation Field Control Seminar*, Electric Power Research Institute, Seattle, Washington, 1993.

Duration: from: 1993 to: 1994

Funding: N/A

Status: In progress

Last Update: January 12, 1994

ON-LINE MONITORING OF DOSE RATES AND SURFACE ACTIVITY DURING THE CYCLE 17 SHUTDOWN OF RINGHALS 2

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; SHUTDOWN CHEMISTRY; DOSE RATES; RINGHALS 2

Principal Investigator:

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Project Manager:

Objectives: Monitor in detail and collect data for the 1993 outage in Ringhals 2.

Comments: The shutdown of Ringhals 2 after cycle 17 in May 1993 was conducted according to the EPRI guidelines for PWR shutdown chemistry. The plant was kept at acidic reducing conditions for some 48 hours. Half of this time was spent at 86°C (187°F). After addition of hydrogen peroxide, an oxidizing clean-up with RCP-operation was run for 25 hours. The shutdown was monitored in great detail by frequent analyses of chemical and radiochemical parameters in the reactor coolant water. When the shutdown was completed, the routine gamma scan of various components in the plant was performed. In addition, crud sampling from four fuel assemblies for chemical and radiochemical analysis also was performed. The reduction in dose rates and surface activity on the excore surfaces is small, within 5-10%. These findings are in agreement with the amount of activity released during the reducing phase. From the data collected, it can be concluded that some 50% of the gamma source strength in the reactor coolant system reside on the stainless steel surfaces. The steam generator tubing holds 10% and the fuel some 20%. The remaining 20% were released and removed from the system during shutdown.

Remarks/Potential for dose limitation: The 1993 shutdown removed some 20% of the total gamma source strength from the reactor coolant system, but we removed very little of the gamma source strength on the RCS excore surfaces. The basis for this statement is the dose rate and gamma scan measurements plus the release during the reducing phase. Extending the time with reducing conditions would probably improve the result, since the release rate of Co-58 was stable at the end of the reducing phase. Zion 2 recently reported substantial dose rate reductions, but they used several weeks at reducing conditions to achieve this. With the dose rates and occupational doses we have had in Ringhals 2, it has not been judged necessary to spend that time to reduce the dose rates. Possible future repair operations and new, stricter rules for occupational exposure may force us to reconsider the implementation of an extended reducing phase. It is most likely that the majority of Co-58 released (some 80%) in the shutdown originates from the fuel cladding and so does at least 50% of the Co-60. The decrease of the Co-60 release through the years indicates that we have decreased the supply of cobalt to the reactor system.

SWEDEN H-204

References: Aronsson, P.O., Bengtsson, B., Bjornkvist, L., and Granath, G., "On-Line Monitoring of Dose Rates and Surface Activity During the Cycle 17 Shutdown of Ringhals 2 1993," *Radiation Field Control Seminar,* Electric Power Research Institute, Seattle, Washington, 1993.

Duration: **from:** 1993 **to:** 1993

Funding: N/A

Status: Completed

Last Update: January 12, 1994

CORROSION PRODUCTS BEHAVIOR IN FRENCH PRESSURISED WATER REACTOR DURING SHUTDOWN OPERATION

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; SHUTDOWN CHEMISTRY; CORROSION PRODUCTS; FRENCH PWRS

Principal Investigator:

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Project Manager:

Objectives: The objectives of the experimental program are:
1) Primary water measurements
 - volumetric activity of corrosion products by gamma spectrometry
 - concentration of chemical elements by X-ray fluorescence
2) Deposited activity measurements with a portable gamma spectrometer
 - on steam generator tubing
 - on primary pipes
3) Measurements were carried out during 20 shutdowns for refueling (900 MWe PWRs)

Comments: During shutdown operation, cooling and injection of boric acid modify pH of the primary coolant leading to a significant increase in the activity of corrosion products. The modification from reducing to oxidizing conditions of the primary water produces a significant increase in cobalt isotope activity. The separate effects of cooling and oxygenation are quantified and studied. Analysis of concentrations and activities of corrosion products in the primary coolant enabled us to identify their origin.

Remarks/Potential for dose limitation: The evolution of contamination of the steam generator tubes and primary pipes using gamma spectrometry measurements during more than 20 shutdowns lead to the following conclusions:
- No significant reduction of out-of-core deposited activities occur during cooldown.
- The majority of the Co-58 and Co-60 activity released into the primary water come from the dissolution of in-core deposits (Ni or NiO).
- Aeration at temperatures >90°C increases the Co-58 deposited on the out-of-core surfaces.
- The use of H_2O_2 reduces the outage critical path.

References: Anthoni, S., Ridoux, P., and Caramel, C, "Corrosion Products Behavior in French Pressurised Water Reactor During Shutdown Operation," *Radiation Field Control Seminar*, Electric Power Research Institute, Seattle, Washington, 1993.

Duration: from: 1992 to: 1994 **Funding:** N/A

Status: In progress **Last Update:** January 12, 1994

PWR STARTUP AND SHUTDOWN CHEMISTRY GUIDELINES

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; COOLANT CHEMISTRY; STARTUP CHEMISTRY; SHUTDOWN CHEMISTRY; CHEMISTRY GUIDELINES

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Project Manager:

Objectives: Develop PWR startup and shutdown chemistry guidelines.

Comments: The following guidelines originated from the study of industry data and experience by an ad-hoc committee formed by several utilities and EPRI.

Technical Basis for Shutdown Chemistry Control
- Field data did not strongly defend intensive or impacting chemistry controls during shutdown to reduce exposure rates
- Laboratory research indicated mildly acidic conditions would not likely affect long-term, Co-60 rich deposits that control plant exposure rates
- Some field data suggested that lack of chemistry control during shutdown may lead to release of particulate corrosion products and increased plant exposure rates
- The committee finally concluded that guidelines could identify the most appropriate shutdown chemistry scheme, though the technical basis was problem-avoidance, not radiation field reduction

Remarks/Potential for dose limitation:
Principles of Refueling Cold Shutdown Chemistry Control
1) Control reactor coolant pH during cooldown: Chemistry personnel should take measures to prevent increasing the alkalinity of the reactor coolant during cooldown. This is done by:
2) Maximizing the time with at least one reactor coolant pump operating.
3) Monitor coolant chemistry and maintain adequate cleanup capability.
4) Maximize letdown purification flow.
5) Maintain reducing conditions.
6) Ensure conditions in decay heat removal system do not adversely impact the chemistry of the RCS when mixed.
7) Create acid-oxidizing conditions to provide controlled solubilization of radiocobalt.
8) Reduce dissolved hydrogen to <5 cc/kg prior to opening the RCS.

References: Brobst, G., "PWR Startup and Shutdown Chemistry Guidelines," *Radiation Field Control Seminar*, Electric Power Research Institute, Seattle, Washington, 1993.

Duration: from: 1992 to: 1994

Funding: N/A

Status: In progress

Last Update: January 12, 1994

GERMANY H-207

UPDATE ON DOSE RATES IN SIEMENS-DESIGNED PWRS

Keywords: CONTAMINATION PREVENTION; OPERATIONAL AND MAINTENANCE TECHNIQUES; DOSE RATES; SIEMENS; STELLITE REDUCTION; WATER CHEMISTRY

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Objectives: Investigate
1) Radiation fields in Siemens/KWU PWRs
2) Influence of the cobalt replacements on Co-60 activity concentration in the coolant and on the system surface
3) Comparison of Co-60 and Co-58 concentration levels
4) Overview on the occupational radiation exposures

Comments:
- An improvement was shown going from pH 6.9 (at 300°C) to pH 7.2-7.4 and further at plants with low cobalt inventory
- 1991 conclusions were: Radiation field will be reduced by using cobalt-free hardfacing materials and by operating with the "modified" B/Li chemistry

Results Drawn From Dose Rate Development at 7 Siemens PWRs
- The plant starting to operate at pH 6.9 (300°C) (coordinated B/Li chemistry) continues to see a slight increase in radiation field
- The plant starting to operate with "modified" B/Li chemistry seems to have reached an equilibrium situation after 4 years
- All plants with cobalt replacements are operating with dose rates of <= 0.5 mSv/h

Remarks/Potential for dose limitation: All data available at Siemens/KWU confirm the Stellite replacement concept. To summarize:
- Radiation fields reduce with Stellite reduction
- Co-60 on surfaces reduces with Stellite reduction
- Coolant Co-60 reduces with Stellite reduction
- Coolant Co-60 correlates with Co-60 on surfaces
- Co-58 on surface: no correlation with Stellite reduction
- Coolant Co-58 does not correlate with Stellite reduction

References: Marchl, T. and Riess, R., "Update on Dose Rates in Siemens-Designed PWRs," *Radiation Field Control Seminar*, Electric Power Research Institute, Seattle, Washington, 1993.

Duration: from: 1991 to: 1994

Funding: N/A

Status: In progress

Last Update: January 13, 1994

DOSE RATE TRENDS AND CHEMISTRY AT SIEMENS-DESIGNED BWRS

Keywords: CONTAMINATION PREVENTION; OPERATIONAL AND MAINTENANCE TECHNIQUES; DOSE RATES; SIEMENS; STELLITE; HEATER DRAIN

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Objectives: Investigate the dose rate trends and chemistry at Siemens-designed BWRs.

Comments: Current Status:
- All plants were operating with Normal Water Chemistry (NWC)
- None of the plants has considered Hydrogen Water Chemistry or Zinc Injection
- As far as there are data available the conclusion cannot be drawn that there is a remarkable influence of water chemistry on the Co-58 and Co-60 activity concentrations. However, the material concept seems to have a major influence on these values.

Remarks/Potential for dose limitation:
Conclusions from Occupational Radiation Exposures and from the Dose Rate Measurements:
- The plant with external recirculation piping has the highest radiation fields and the highest exposure
- Forward pumped units show higher radiation levels
- However, cascading or forward pumping of heater drains has no influence on personnel exposures
- Stellite replacement is most effective in reducing radiation fields and exposure rates

Two major steps can be identified in order to reduce radiation fields and occupational radiation exposure in Siemens/KWU BWRs:
- Replacement of the external recirculation piping
- Partial replacement of cobalt-base alloys in the RPV

References: Reitzner, U., Marchl, T., and Riess, R., "Dose Rate Trends and Chemistry at Siemens-designed BWRs," *Radiation Field Control Seminar*, Electric Power Research Institute, Seattle, Washington, 1993.

Duration: from: 1992 to: 1994

Funding: N/A

Status: In progress

Last Update: January 13, 1994

APPENDIX

Introduction to ACE and ACEFAX

In addition to being periodically published in NUREG/CR-4409 reports, the information on dose reduction research and health physics technology projects gathered by the ALARA Center are available through two computerized systems, ACE and ACEFAX. ACE stands for ALARA Center Exchange. It is an on-line database system accessible via computer, modem, and the appropriate communications software. ACEFAX denotes our ALARA Center Exchange interactive fax retrieval system. By means of a fax machine with a handset (as opposed to computer fax boards), one can instruct ACEFAX to fax the information of interest to oneself. We will proceed to describe these two systems in detail.

ACE

Any new, pertinent information from published reports, conference proceedings, personal contacts, and other sources arriving at the ALARA Center is first summarized and then entered into our ACE electronic database system (hereby referred to simply as ACE). Thus, this medium contains the most up to date information available from the ALARA Center at any time. ACE is actually composed of a total of nine distinct databases, each focusing on different ALARA issues. The databases and their contents are:

BIB	1500 abstracts of published articles related to dose reduction and ALARA
DOSES	current radiation dose exposure levels for workers at US nuclear power plants
HPTECH	health physics technology - applied efforts at plants to reduce dose levels
JOBS	lessons learned from high dose jobs at plants
NEWS	current events from journals, newletters, etc. in dose reduction and ALARA
PLANTS	responses by plants on the usefulness and efficacy of new processes and practices
PRACTICE	current administrative and management techniques to reduce dose levels
PROCESS	current engineering and scientific techniques to reduce dose levels
RESEARCH	latest research findings on ways of reducing occupational doses in plants

ACE is organized on an IBM PS/2 with the commercial database software Q&A, produced by the Symantec Corporation. It connects to the outside world by modem and the software CLOSE-UP. In order to access ACE, one needs a PC compatible computer with a modem and the communications software CLOSE-UP 5.0, a commercial product available from the Norton-Lambert Corp. (For details on obtaining CLOSE-UP 5.0, see p. A-21.) CLOSE-UP is a remote control program that allows a person to use ACE as if one is sitting in front of the ALARA Center IBM PS/2 computer. It sets up the user's computer as a terminal (remote) and the ALARA Center computer as the controlling entity (host).

Once the communication link between the host and remote is established, a menu will be displayed on the user's computer screen. (See the ACE Reference Guide on p. A-23.) For the sake of simplicity and ease of use, only four core commands have been defined for ACE. The commands allow the user to: 1) look sequentially at the records (often referred to as data forms, documents, or information sheets) of any of the 9 databases in ACE, 2) select a topic of interest through the use of keywords, 3) print the record on the screen, and 4) return to the menu from anywhere in the system.

ACEFAX

After the new information to the ALARA Center has been entered electronically into ACE, the next step is to produce a hard copy of the information. Records from the HPTECH, JOBS, NEWS, and RESEARCH databases are put into bite size documents of 1 or 2 pages designed to capture the main points and conclusions of the original report and provide a source for further information in an easy to read and pleasant looking format. The documents are then put on our ACEFAX system (hereby referred to simply as ACEFAX).

ACEFAX complements the ACE system in the distribution of ALARA Center information. It allows a person to access the four aforementioned databases plus some general documents by means of a regular fax machine with a handset (not from computer fax boards, however). ACEFAX was set up with the belief that more people have (or prefer to use) fax machines than computers. The main advantage being the ease of use, while sacrificing some of the flexibility of ACE.

(continued on next page)

When a person calls from a fax machine, ACEFAX responds with a voice message and brief instructions on how to obtain the documents. Basically, the user enters the number of the document of interest and ACEFAX will send that document to the user's fax machine. Document "12" contains the list of all documents available on ACEFAX. The document numbering system is as follows:

1	to	100	general
101	to	500	health physics technology database
501	to	1000	jobs database
1001	to	2000	research database
2001	to	3000	reserved for future use
3001	to	4000	news database

Note that the numbers assigned to the health physics technology and research database documents are related to the numbering of the projects published in the NUREG/CR-4409 (vol. 4 & 5) report. For example, projects H-195 and R-385 from NUREG/CR-4409 correspond to documents 195 and 1385 in ACEFAX.

Phone Numbers for ACE AND ACEFAX

ACE 516-282-3481
(call from computer with modem and CLOSE-UP 5.0)

ACEFAX 516-282-7361
(call from the handset of a fax machine)

BROOKHAVEN NATIONAL LABORATORY
ALARA CENTER
ACEFAX SYSTEM

Requesting ALARA Documents Using Your Fax Machine

For the fastest service, you don't even need a computer. The ACEFAX system will fax information to you immediately. The service is available 24 hours a day. Here's how to use it:

1. Pick up the handset on your facsimile machine and dial (516) 282-7361. **Listen with the handset -- do not hang up after dialing.**

2. A voice will answer and give you instructions. Using the touchtone pad of your fax machine, you will enter the number of the document you want from the attached list. You may order up to five documents per session. To correct an error, press the astertisk key (*) and enter the number again.

3. When you are prompted by the voice, press the "START/COPY" or "RECEIVE" button on your fax machine (usually the green button). You may then hang up the phone. You will begin to receive the documents within a minute.

   Note: If you receive errors, your fax machine and ours may be incompatible. Fill in the form below and mail it to the **ALARA Center, Building 703M, Brookhaven National Laboratory, Upton, NY 11973**. We will mail the documents to you.

If You Don't Have a Fax Machine

Name ______________________________

Company ______________________________

Address ______________________________

____

City, State, Zip ______________________________

Please send the following documents:

____

Please check one: ____ I don't have a fax machine.

____ I was not able to make a successful connection to ACEFAX.

BNL ALARA Center

List Of Documents On ACEFAX

516/282-7361

General Documents

10 ALARA Notes No. 8 (22 pages)
12 LIST OF DOCUMENTS ON ACEFAX (15 pages)
13 ACE Manual (Revision 5) (4 pages)
14 ALARA Notes No. 9 (19 pages)
15 Dollar Worth Of A Person-Rem For US Utilities: Updated 1994 Values (1 page)
16 A Survey Of Doses To Worker Groups In The Nuclear Industry (13 pages)
17 Collective Dose Per Reactor For Selected Countries (1 page)
18 Registration Form For ACE (1 page)
19 Bournemouth Meeting On Water Chemistry Of Nuclear Reactor Systems #6 (10 pages)
20 More Details On Documents On ACEFAX (20 pages)
21 The Program Of The ALARA Center At Brookhaven National Laboratory (15 pages)
22 BWR and PWR Collective Radiation Exposure 3-Year Rolling Average Tables For 1992 (2 pages)
23 Third International Workshop On Implementation Of ALARA At Nuclear Power Plants - May 1994 (Information and Registration Form) (4 pages)
24 Third International Workshop On Implementation Of ALARA : Agenda (13 pages)
25 Subject Index For Jobs Database (1 page)
26 Subject Index For Health Physics Technology Database (2 pages)
27 Subject Index For Research Database (4 pages)
28 Subject Index For News Database (6 pages)
30 Collective Dose And Electricity Produced: U.S. Nuclear Power Plants (1 page)

Documents From The Health Physics Technology Database

138 Innovative Approaches At TMI-2
139 Identify All Cobalt Contributors In PNPS
140 Evaluate Hot Spots Associated With Spent Fuel Pool System
141 Surrogate Laser Disc Plant Tour System
142 Maintain Radiological Evaluation Factors
143 Replace Feedwater Control Valve Trim With Non-Cobalt Design
144 Recirculation Pump Cobalt Elimination
145 Fuel Improvements To Reduce Cobalt Source
146 Establish Chemical Decontamination Strategy

147 Evaluate Zinc Addition To Reactor Feedwater (GEZIP)

148 Reactor Control Blade Management Considering ALARA

149 Evaluation, Possible Reduction In Operation And Testing Of CRDs To Reduce Cobalt Input

150 Project MINDOS

151 Study On The ALARA Policy In Korea

152 Reduction Of Time, Exposure, And Cost Through Plant Decontamination

153 Reactor Cavity Decontamination At V. C. Summer

154 Use Of Respirators And Dose Expansion

155 Optimizing Worker Protection: A Practical Application Of Risk Analysis

156 Advanced Radiation Worker Training Program And Laboratory

157 ALARA Aspects Of The Calvert Cliffs Pressurizer Repair Project

158 ACE - ALARA Center's Dose-Reduction Information System

159 An Effective ALARA Awareness Program

160 An ALARA Training Program For Design Engineers

161 System Decontamination Of RWCU System

162 Resistance Temperature Detector Bypass System Elimination

163 400 R/hr Hot Spot Removal At Cooper Nuclear Station

164 Innovative Shielding

165 Removal Of Control Rod Drive Through Robotics

166 Data Acquisition On PWR Contamination

167 Panthere RP: A Tool For Evaluating Dose Rates

168 The Ingredients Of A Utility's Dose Reduction Program

169 Methods Used To Achieve Outage Goals At Diablo Canyon

170 Radiation Exposure Reduction Program At Mitsubishi Heavy Industries

171 Clamshell Nozzle/Pipe Shielding

172 Feedwater Nozzle Thermal Sleeve Hydrolyzing

173 Snubber Positioning Fixture

174 Removal Of Fine Chrome Particulate From Spent Fuel Pools By Means Of A Radial Lamella

175 A Method For Optimizing The Use Of Respiratory Protection In Radiation Areas

176 Indian Point 2 Sub-System Decontaminations

177 National Demonstration Of A Full RCS Chemical Decontamination

178 Plant E.I. Hatch Chemical Decon 1991

179 Chemical Decontamination Of The Residual Heat Removal System

180 Ontario Hydro Decontamination Experience

181 Resource Management As An ALARA Tool

182 PWR Upper/Lower Internals Shield

183 Internal Dose, Respiratory Protection And Revised 10CFR20 At Davis-Besse Nuclear Power Station

184 Zion Unit 2 Cycle 12 Shutdown And Early Boration Results

185 Health Physics Services On The Platform At Salem Using ROMMRS

186 Chemical Decon Of Systems: Results And Problems

187 An Automated Program Implementing New 10CFR20 Requirements At Southern Nuclear Plants

188 Steam Generator Replacement Project At North Anna Power Station
189 ALARA Programme Management And Organization In EDF Nuclear Power Stations
190 Enhanced Radiation Worker Training At James A. Fitzpatrick Nuclear Plant
191 S/G Replacement At Beznau 1: Experience And Results In Radiological Protection
192 Partners In Performance: An ALARA Perspective
193 Steam Generator Snubber Elimination
194 Future Power Stations In The United Kingdom: Designing For Low Doses
195 A Team Approach For The Management Of Radioactive Liquid Effluents
196 ALARA And Work Management
197 Radiological Assessment Of Decommissioning At Fort St. Vrain
198 Personnel Radiation Exposure Reduction During Remote Stud Handling At Indian Point 2
199 Replacement Of RWCU Piping With State-Of-The-Art Materials
200 Evaluation Of Zircaloy Fuel Clad Oxidation At Millstone 3 PWR
201 An Update On Chemistry Related Dose Reduction Efforts At Millstone Nuclear Generating Stations
202 Surface Characterization Of The Steam Generator Channel Head Following Mechanical/ Electropolishing At Millstone Point Unit 2
203 Chromium Treatment Of RHR Piping
204 On-Line Monitoring Of Dose Rates And Surface Activity During The Cycle 17 Shutdown Of Ringhals 2
205 Corrosion Products Behavior In French Pressurised Water Reactor During Shutdown Operation
206 PWR Startup And Shutdown Chemistry Guidelines
207 Update On Dose Rates In Siemens-Designed PWRs
208 Dose Rate Trends And Chemistry At Siemens-Designed BWRs

Documents From The Job Database

501 Replacement Of Waste Collector Filter Septa
502 Welding And Inspection Of Pressurizer
503 Repair Of Steam Dryer
504 Decontamination Of Reactor Cavity
505 Replacement Of Filter Septums
506 Remote Control Rod Drive Handling System
507 Core Grid Support Repair
508 Heat Exchange Decontamination
509 Modification Of Reactor Temperature Detector System
510 Turning Vanes Replacement
511 Decontamination Of RWCU Using The Cord Process
512 Feedwater Sparger Replacement
513 Work Inside Reactor Vessel
514 Vacuum Cleaning Of Steam Generators
515 Decontamination Of Primary Coolant Pumps

516 Replacement Of Incore Monitors
517 Desludging And Decontamination Of Radwaste Aisles
518 Installation Of Insulation Cartridges
519 Shot Peening Of The Hot Leg Tubes Of A Steam Generator
520 Shipment Of Spent Fuel Assemblies
521 Radiation Protection Surveys And Job Coverage
522 Replacement Of a Pump In a High Radiation Area
523 Insulation Replacement On Recirculation Piping And Valves
524 Replacement Of Insulation On Main Steam Piping
525 Removing Scaffolding From Drywell
526 Replacement Of Reactor Recirculation Pump Cooler
527 Replacement Of a Flange Gasket
528 Repair Of Main Steam Sensing Line Supports
529 Replacement Of Steam Generators
530 Replacement Of Primary Pipes Inside Containment
531 Shot Peening Of The Cold Leg Tube Of Steam Generator
532 Up-Flow Conversion
533 Additional Experience From Shot Peening
534 Replacement Of Reactor Coolant System Valve Internals
535 Decontamination Of The Drywell
536 Valve Repair In Reactor Water Cleanup System
537 Repair Of Tube Bellows Flange
538 RWCS Pump Reassembly

Documents From The Research Database

1229 VERALIGHT - A New Light Manipulator For Steam Generator Inspection
1250 Development, Fabrication, And Test Of The ODEX-3 Maintenance Vehicle
1251 Source Book For Chemical Decontamination Of Nuclear Power Plants
1252 The Nature And Behavior Of Particulates In PWR Primary Coolant
1253 PWR Radiation Control Demonstration
1254 Field Tests Of Radiation Control Techniques - 1
1255 Effect Of Surface Treatments On Radiation Buildup In Steam Generators
1256 Millstone 1 Zinc Injection Evaluation
1257 PWR Steam Generator Preconditioning Studies
1258 The Treatment Of Radioactive Ion-Exchange Resins
1259 PWR Corrosion Tests Using LOMI
1260 Crud Transport Chemistry
1261 Qualification Of Cobalt-Free Hardfacing Alloys For LWR
1262 Production Of NOREM Hardfacing Alloys
1263 Cobalt Replacement Guidelines
1264 BWR Cobalt Deposition Studies

1265 Research Reactor Loop Water Chemistry Study

1266 Radiation Field And Dose Data Assessment

1267 Passivation And Surface Conditioning

1268 Feedwater Flow Element Improvement

1269 Coolant Chemistry And Radiolysis In Boiling Reactor Coolant

1270 On-Line Monitoring Techniques For Redox Potential, Hydrogen Concentration, and pH In Nuclear Reactor Coolant Circuits

1271 In-Plant System For Continuous Low-Level Ion Measurement In Steam-Producing Water

1272 Resin Separability To Improve Polishing Under Morpholine AVT

1274 Oxygen Transport In BWR Cycles

1275 Remote Repair Technique For MSIVs

1276 Intellitorque : A System For Monitoring Root Cause Of MOV Malfunctions

1277 Using Ultrasonics To Avoid Check Valve Disassembly

1278 A "Wet Motor" Sealless Pump For Reactor Water Clean Up System In BWRs

1279 A Rotating UT System For Inspection Of Steam Generator Tubes

1280 The ALOK 3 Ultrasonic Inspection System

1281 Acoustic Leak Monitoring In Japan

1282 Use Of Vibration Monitoring To Assess Reactor Coolant Pump Integrity

1283 Improved Test Methods For Plant Protective Coating

1284 Automated Control Rod Drive Bolting Wrench System To Support Boiling Water Reactor Maintenance

1285 Measurement Of Oxide Film Released As Particles During The CAN-DEREM Decontamination Process

1286 "Wet Motor" Sealless Pump For Reactor Water Clean Up System In BWRs

1287 Reactor Water Cleanup (RWCU) Sealless Pump

1288 Exposure Reduction Measures In The Design Of Siemens/KWU PWR Plants

1289 Full System Decontamination Of The BR-3 PWR Plant

1290 Mitigation Of The Impact Of Reduced Radiation Exposure Limits On Nuclear Power Plant Operations

1291 Sources Of Cobalt-60 In The Primary Systems Of Pressurized Water Reactors

1292 Performance Of Iron Base Hardfacing Alloys Under Pressurized Water Reactor Conditions

1293 U.K. Program To Qualify Cobalt-Free Hardfacing Alloys

1294 Supplying Cobalt-Free Nuclear Valves

1295 An Examination Of Foreign Approaches To Controlling Radiation-Field Buildup In Boiling Water Reactors

1296 Guidelines For The Reduction Of Cobalt From Reactor Systems

1297 BWR Radiation Field Trends

1298 Status Of Zinc Injection In Boiling Water Reactors

1299 Experience With Zinc Injection At Millstone 1

1300 Control Of Radiation Fields At Boiling Water Reactors By Reducing Iron Input

1301 Effect Of Preconditioning On Cobalt Corrosion Release Rates

1302 Radiation Field Issues In Switching To Hydrogen Water Chemistry

1303 Qualification Of Electropolishing For Replacement Steam Generators
1304 French Experience With Electropolishing Steam Generator Channel Heads
1305 Surface Pretreatment Of Primary System Components To Reduce Radiation Buildup
1306 Reducing Radiation Buildup By Surface Coating Of Primary System Components
1307 PWR Primary Water Chemistry Guidelines - Revision 2
1308 Reduction Of Radiation Fields By Elevated pH Control At Millstone-3
1309 Loop Experiments On Zinc Injection Under PWR Conditions
1310 Corrosion Control And Dose Rate Reduction
1311 Effects Of pH And Li On PWSCC Initiation And Growth
1312 Radioactivity Pick-Up By Carbon Steel And Stainless Steel In Slightly Oxidizing Lithiated Coolant
1313 Lessons Learned From Recent BWR Chemical Decontamination Applications
1314 Decontamination Of Beaver Valley Steam Generators Using The CAN-DEREM Process
1315 Full RCS Chemical Decontamination
1316 BWR Full System Decontamination
1317 PWR Coolant Chemistry Studies In Support Of Dose Reduction Using In-Pile Loops At MIT
1318 Solubility Measurement Of Crud And Evaluation Of Optimum pH
1319 Full Reactor Coolant System (RCS) Decontamination National Demonstration Plan
1320 Full System Decontamination Of The BR-3 Plant
1321 Future Developments In Processing Decontamination Waste
1322 Reduction Of Critical Path Time For BWR Recirculation System Decontaminations
1323 Improvements In The LOMI Decontamination Process
1324 Radiation Fields Trends And Control At French PWRs
1325 Weldability Of NOREM For In-Situ Repair & Replacement
1326 High pH Operation At Swedish PWRS
1327 Radiation Field Control By Early Boration During Shutdown At Beaver Valley Power Station
1328 High pH Operation In ABB Combustion Engineering Plants
1329 Reactor Coolant System Shutdown Chemistry And Nickel Management At H.B. Robinson Nuclear Project
1330 Zinc Injection At Millstone 1
1331 The Effect Of Zinc On Corrosion And Dose Rate Control
1332 TRACKER: An Absolute Tube-Position Detection And Tube Marking System
1333 BWR Underwater Disassembly/Assembly - Wetlift 2000
1334 PWR Primary System Chemistry: Exprience With Elevated pH at Millstone Point Unit 3
1335 Steam Generator Dose Rates At Babcock & Wilcox Reactors
1336 Preconditioning Of PWR Steam Generators To Reduce Radiation Buildup
1337 Welding Of NOREM Iron-Base Hardfacing Alloy Wire Products - Procedures For Gas Tungsten Arc Welding
1338 Endurance Tests Of Valves With Cobalt-Free Hardfacing Alloys
1339 Replacement Of Pins And Rollers In Irradiated BWR Control Blades
1340 Secondary Hydriding Of Defected Zircaloy-Clad Fuel Rods
1341 Testing Of An Organic Removal Process In BWR Radwaste Systems
1342 Evaluation Of Reactor Pressure Vessel Head Cracking In Two Domestic BWRs

1343 Relationship Of Radiation-Induced Segregation Phenomena To Irradiation-Assisted Stress Corrosion Cracking (IASCC)

1344 Light Water Reactor Materials And Water Chemistry Studies At Halden

1345 PWR In-Pile Loop Studies In Support Of Coolant Chemistry Optimization

1346 The Effect Of Dissolved Oxygen In Lithiated Coolant

1347 Chemistry Parameters Influencing The Dose Rate Build-Up In BWR Plants

1348 Overview Of Activities For The Reduction Of Dose Rates In Swiss Boiling Water Reactors

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3003 Enriched Boric Acid Promises Greater Flexibility For PWRs

3004 Control Rod Drive Removal And Installation Mechanism Saves Dose

3005 Steam Generator Pipe End Decon Saves Approximately 100 Person-Rem

3006 More On Strippable Coatings

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3010 Protective Coatings For Radiation Control In Boiling Water Nuclear Power Reactors

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3147 World Nuclear Industry Handbook - 1992
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3173 Remotec's All-Terrain Robots
3174 New Remote Radiation Surveillance System
3175 Instant Monitoring On Hand

3176 Improved Plant Designs Reduce Dose By A Factor Of 20

3177 Dose Control At China's First Nuclear Power Plants

3178 Candu Large Scale Fuel Channel Replacement Project: Individual & Collective Dose Reduction By ALARA Integration

3179 Radiation-Field Control Manual -1991 Revision

3180 Nuclear Power Plant Resource Book - Vol. 2: BWR

3181 A Brief History Of Robots In The United States

3182 Trod Cleans Up At Nine Mile Point 1

3183 More Radiation Monitoring Backfits For The Future

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3191 BWR Radiation Fields With Hydrogen Water Chemistry

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3197 Looking For Links Between BWR Hydrogen Water Chemistry And Increased Shutdown Dose Rates

3198 New PWR Guidelines Target Intergranular Attack And Stress Corrosion Cracking In Steam Generators

3199 Water Chemistry And Dose Reduction: New PWRs

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3201 PWR Primary Coolant Chemistry And Dose Reduction

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3224 BWR Chemistry Guidelines Revision Committee Being Formed
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3226 GPU Cuts Critical Path Time By Using EPRI Radwaste Desk Reference
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3228 10 CFR 20'S Impact On Computerized Dose Tracking
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3238 Piping System Inspection And Testing: Managing The Massive Results, Records, And Reports
3239 Reducing Radiation Dose By Effective ALARA Engineering
3240 Service Worker Dose Reduction: Whose Job Is It?
3241 Control Rod Drive Mechanism Nozzle Inspection
3242 Reducing Exposure At U.S. Nuclear Plants
3243 PWR Primary Shutdown And Startup Chemistry Guidelines Complete
3244 1992 Year-To-Date Dose At U.S. BWRs
3245 Iron-Based Alloy Testing In BWR Chemistries
3246 Second Meeting Of The ISOE Steering Group
3247 Some Successful Techniques For Exposure Control
3248 Radiation Exposure Trends
3249 Chromium Coating To Reduce Activity Pickup
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3256 UK Goes For 20 msv Limit
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3261 Listening To Reactor Pressure At The Boundaries For The Sounds Of Cracks And Leaks
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3270 More Circumferential Cracking Indications Found In Ringhals 2 Upper Head
3271 A Record-Breaking SG Replacement
3272 Neutron Detectors
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UPDATE: ACE ON-LINE INFORMATION SYSTEM TO SWITCH TO CLOSE-UP 5.0

Access to the databases of the BNL ALARA Center is available on-line, 24 hours a day to the ALARA community using the ALARA Center Exchange (ACE). Formerly, ACE access was possible only with the communications software, PC Anywhere III. However, new ACE users are finding this software very difficult to obtain, making the switch to CLOSE-UP 5.0 necessary. ACE uses only four basic commands, and the new four-page user manual (attached) includes all the instructions required to access and search the ALARA Center databases. **AS OF MARCH 1, 1994 THE ACE SYSTEM WILL BE ACCESSIBLE ONLY WITH CLOSE-UP 5.0.**

If you own any other communications softw.re, you can upgrade to CLOSE-UP 5.0 at a special low price of $69.95. (The list price is $199) This limited-time offer is being made by the maker of CLOSE-UP, Norton-Lambert Corp., P.O. Box 4085, Santa Barbara, CA 93140, USA (Phone 805/964-6767; Fax: 805/683-5679). When ordering, you will need to supply Norton-Lambert with a xerox of the first page of the commications software manual, or a xerox of the program disk, that you now own. This special offer will expire on June 30, 1994.

Once you receive CLOSE-UP, please complete the form below, and mail or fax it to us. This will ensure that you receive notification of any changes to the system, on- and off-line assistance, and updated material on ACE.

Name: ______________________________

Job Title: ______________________________

Company: ______________________________

Address: ______________________________

____

City, State, Zip, Country: ______________________________

Telephone: ______________________ Fax: ______________________

Close-Up Registration Number: ______________________________

Please return to:

Maria Beckman
BNL ALARA Center
Building 703M
P.O. Box 5000
Brookhaven National Laboratory
Upton, NY 11973-5000

Phone: 516/282-3228
Fax: 516/282-7091 or 282-5810

Please follow these instructions carefully and do not give commands that are not on this card.

In order to connect to ACE you must install the CLOSE-UP program and run CREMOTE *(the part of CLOSE-UP which is designed for linking from a remote computer).*

1. General Comments on Using ACE.

(1) The purpose of ACE is to provide a vehicle for rapid information search and retrieval.

(2) There are just four items on the ACE Quick Menu. **You MUST follow the ACE commands on this card to retrieve information.** *You do NOT need any other commands that may be shown on the ACE screen.*

(3) Each ACE menu command is activated by pressing the ALT key and a letter. *You can not activate these commands by highlighting them and pressing ENTER.* Each of these commands is a MACRO program. **Please complete each command as described in this card, so that the macro program is not interrupted.** *(If you must interrupt the macro, type ESCAPE to stop the macro before you use the next ALT menu command).*

(4) After typing a command, pause and let the modem and computers complete the command. Give the next command only after the screen has come to a stand still.

(5) The program CREMOTE is used to link to ACE. CREMOTE has its own menu and instructions. You will see this menu before you even connect to ACE. You MAY follow the CREMOTE instructions from the screen. Type ALT-R to call up the CREMOTE menu at any time. To remove the CREMOTE menu from the screen type ESCAPE once or twice.

2. Installing CLOSE-UP on your PC (for details, see Close-up Manual, page 3-1).

(1) Put the CLOSE-UP disk into the A: drive. At the DOS prompt type A:CINSTALL. Follow the directions on screen.

(2) The installation is essentially automatic *(at the end of the installation process the program may propose a baud rate which is several times the transmission rate of your modem. This will be due to the data compression capabilities of modern modems. We recommend however, that you select a baud rate no higher than 19,200 for 9,600 and 14,400 baud modems and 2,400 for 2,400 baud modems).*

(3) *Optional:* Add the directory C:\CLOSEUP to the PATH statement in your computer's Autoexec.bat file *(e.g. PATH=C:\CLOSEUP;).* This way you can start CREMOTE from any directory.

3. Connecting to ACE (for details, see Close-up Manual, page 3-4).

(1) Type CREMOTE at the DOS prompt *(this will start the CREMOTE program and CREMOTE menu items will be displayed on the screen).*

(2) The menu item *phone* and its drop-down menu item *dial* will

already be selected for you. Press ENTER twice to select *Manual Dial.*

(3) Type the ALARA Center number and press ENTER. *(Include any prefix digits for outside lines and long distance that you may require e.g. from inside the U.S. 1, 516 282 3481).*

(4) You should be connected to ACE and see the **ACE Quick Menu**.

(5) After a short while you will see a message ***Connection established. Press ENTER to begin session.*** When you press ENTER you will start using ACE.

4. Browsing through a Database

(1) Press ALT-B from the Quick Menu.

(2) Highlight a database with UP / DOWN arrow keys and press ENTER. *(As each database is highlighted its contents will be described by an enhanced line at the bottom of the screen).*

(3) To see other screens of a form, press PAGEUP or PAGEDOWN keys *(not UP / DOWN arrow keys).*

(4) To see the next form, press F10; to see previous form, press F'

5. Searching through Keywo..

(1) Press ALT-K from the Quick Menu.

(2) Highlight a database with UP / DOWN arrow keys and press ENTER. *(A partial query will be started at the top of the form. Below that, in a window, an alphabetically sorted list of all the KEYWORDS will be displayed, starting with the letter 'A').*

(3) If the keyword you require starts with another letter start typing your keyword. It will appear near the top of the list. *(e.g. start typing "zinc" to see the keyword "zinc injection" near the top of the list).*

(4) Highlight the keyword and press ENTER. All forms with your keyword will be displayed. *(please note item 8 below).*

(5) Press PAGEUP or PAGEDOWN keys to see other screens of a form.

(6) Press F10 to see the next form on the subject; press F9 to see the previous form.

6. Printing a Form

(1) Make sure in the CREMOTE *Print* menu you have the selection *Print at Remote only* (for details, see Close-Up manual, page 5-23).

(2) Turn your printer ON.

(3) Press ALT-P **while looking at the form you want to print**. The form will be printed on your printer. *(Note: Do not install any new printers. An all purpose printer has already been setup for you).*

7. Printing to a File on your Fixed Disk

(1) Select *Print at Remote Spooler* in the CREMOTE *Print* menu (for details, see Close-up manual, page 5-24).

(2) Press ALT-P while looking at the form you wish to print *(Printing to the spooler file and then printing later off-line will save long distance charges).*

8. To see the 'Quick Menu'

(1) Press ALT-Q from any screen. (*If the ALT-B or ALT-K command has not*

been completed, e.g. after these commands you do not wish to select a database or Keyword but to see the Quick Menu again, then stop the macro program by pressing ESCAPE before pressing ALT-Q).

9. Exiting ACE (Close-up manual, p. 3-5)

(1) Press ALT-Q

(2) Press ALT-R from any screen. *This will bring back the CREMOTE menu.*

(2) *Phone* and *Hang-up* will already be selected for you from the CREMOTE menu. Press ENTER twice to hang-up the line.

10. Setting up ACE in the Phone Book (for details, see Close-up Manual, page 3-5).

(1) Press ALT-R to bring up the CREMOTE menu.

(2) Select *Phone* and then *Phone Book Editor. (This brings up the editor menu).*

(3) Select *Phone Book*

(4) Follow the instructions on the screen.

(5) For *baud* rate, we recommend you use *Default.*

11. Capturing Screens during an ACE session

(1) Whenever you wish to capture and save information in a screen, press ALT-R to bring up the CREMOTE menu.

(2) Select *Record* and then *Save Current Screen.* Follow CREMOTE prompts to save screen.

(3) Use the ESCAPE key to remove the CREMOTE menu from the screen

(4) Saving, and then playing back saved screens is much quicker than printing on-line. You may then print from saved screens off-line to save long distance charges.

12. Playing Back Screens

(1) Exit ACE (see item 9 above).

(2) In the CREMOTE menu select *Playback* and then *Display Saved Screens*

(3) Follow screen instructions.

13. Printing captured screens

(1) Once the captured screens are displayed you may print them one-by-one using the *Print Screen* key on your personal computer.

14. Recording your ACE session

(1) You may record the whole or any part of your session while connected to ACE. You may then play it back off-line in fast or slow motion.

(2) While connected to ACE, type ALT-R to bring up the CREMOTE menu.

(3) Select *Record* and say *Yes* to *Record Session*

(4) Accept CREMOTE's suggestion to delete old recording file and start a new one.

(5) Your session is now being recorded. Type ESCAPE to remove CREMOTE menu from screen.

(6) To stop recording any time, follow steps (1) to (5), but select *No* in step (3) above.

15. **Playing back the ACE Session**
    (1) Exit ACE by hanging up (see item 9 above).
    (2) Select from CREMOTE *Playback* and *Playback Saved Session*.
    (3) Follow screen instructions for fast / slow motion playback and stopping playback.
16. **Exiting CLOSE-UP REMOTE**
    (1) Press ESCAPE once or twice, until a message appears *Press ENTER to exit Close-up* (see manual, p. 3-5).
    (2) Press ENTER .

Note 1: The elapsed long distance time is always shown on the top right of the screen while you are connected to ACE.

Note 2: You may ask us for advice by opening the Chat window with the command ALT-C. If we are available near the host computer we will respond.

In case of problems, you may contact us by phone at either (516)282-3228 or (516)282-4012.

BNL ALARA CENTER TELEPHONE NUMBER REFERENCE GUIDE

To access ACE by computer and modem	(516)282-3481
To fax a document to yourself through a fax machine (pick up the hand set of your fax machine and dial this number from your fax machine only. Follow the voice prompts. Password is not required)	(516)282-7361
To apply for access to ACE	(516)282-3228
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Information Request Form

BNL ALARA Center Database

1. Title:

2. Investigator(s): **3. Project Manager:**
[*name, organization, address, phone, and fax number*]

4. Objectives:

5. Comments: [*Describe the background, perspective, progress, or significant findings* .]

(continued on next page)

Return to: Dr. Tasneem A. Khan, Brookhaven National Laboratory, ALARA Center, Building 703M, P.O. Box 5000, Upton, NY 11973-5000. Phone: 516-282-4012. Fax: 516-282-7091.

6. Remarks/Potential for dose limitation: [*Indicate any potential beneficial or detrimental dose impacts. Note any important conclusions.*]

7. Duration - From: 19 **To:** 19

8. Funding: [*Monetary amount or person-years*]

9. Status: [*Indicate the degree of completion, i.e., PROPOSED, INITIATED, IN PROGRESS, or COMPLETED.*]

10. References: [*Recent reports, articles, or publications*]

11. Key Words: [*Indicate the broad areas related to the project.*]

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(2-89)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

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(See instructions on the reverse)

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Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This is the fifth 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 two of several databases maintained by Brookhaven National Laboratory's ALARA Center for the U.S. Nuclear Regulatory Commission.

The research section of the report covers dose reduction projects that are in the experimental or developmental phase. It includes topics such as steam generator degradation, decontamination, robotics, improvements in reactor materials, and inspection techniques. The section on health physics technology discusses dose-reduction efforts that are in place or in the process of being implemented at nuclear power plants. A total of 105 new or updated projects are described.

All project abstracts from this report are made available to nuclear industry professionals with access to a fax machine through our ACEFAX system or a computer with a modem and the proper communications software through our ACE system. Detailed descriptions of how to access all of our databases electronically are in the appendices of the report.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

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