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Data Base on Dose Reduction Research Projects for Nuclear Power Plants

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Washington, DC 20555
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ABSTRACT

This is the third volume in a series of reports that provide information on dose-reduction research and health physics technology for nuclear power plants. The information is taken from data base maintained by Brookhaven National Laboratory's ALARA Center for the Nuclear Regulatory Commission.

This report presents information on 80 new projects, covering a wide area of activities. Projects on steam generator degradation, decontamination, robotics, improvement in reactor materials, and inspection techniques, among others, are described in the research section. The section on health physics technology includes some simple and very cost-effective projects to reduce radiation exposures.

Collective dose data from the United States and other countries are also presented. In the conclusion, we suggest that although new advanced reactor design technology will eventually reduce radiation exposures at nuclear power plants to levels below serious concern, in the interim an aggressive approach to dose reduction remains necessary.

CONTENTS

PREFACE

The staff at the ALARA Center, Brookhaven National Laboratory, maintain a data base of information on international projects on dose-reduction research and health physics technology. The data base, concerned primarily with nuclear power generation, is part of a project (FIN A-3259) sponsored by the Nuclear Regulatory Commission. The series of reports NUREG/CR-4409 describe the information in the data base.

The first two reports in this series (Ref.1,2) described approximately 300 projects; this report presents 80 more. Several steps were taken to facilitate access to the information. First, as in the previous reports, the division into two main groups has been retained: the **E** projects are on engineering research, the **T** projects describe efforts related to health physics technology at nuclear power plants. In addition, the projects have been divided into six categories to make it easier to access areas of interest.

The indices list all the projects in the data base, including those in previous volumes. The projects in the present volume are shown in bold face. The indices start by listing all the projects sequentially by identification number, followed by indices for the six major subject categories, for the project manager and principal investigator, and for sponsoring and contracting organizations. The final index has the various subjects and is based on the keywords on each sheet. The addresses, telephone numbers, and where available, telefax numbers of the project managers and the investigators should facilitate exchange of more detailed information.

Since the data is continuously updated and new information sheets added, users of the data base are invited to call (516) 282-4012, or write for the latest information, or to access the data base electronically, as described in section 7.2. Moreover, information on new projects is always welcome. Projects listed in the data base are disseminated internationally, and may lead to exchange of information in areas of interest. A blank form is included in section 8 for sending of information to the ALARA Center. Our telefax number is (516) 282-5810.

ACKNOWLEDGMENTS

We thank all project managers and principal investigators for providing information on their projects. We acknowledge the contribution of several people who have discussed with us related research activities and health physics programs. We thank Alan Roecklein, Robert Alexander, Charles Hinson, and Carl Feldman of the NRC for their support in this work. We would like to acknowledge the advice and support of Charles Meinhold of the Brookhaven National Laboratory. We also express our thanks to Maria Beckman 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. Crandal	North East Utilities
B. Dionne	Boston Edison Company
C. Hinson	Nuclear Regulatory Commission
G. Hudson	Tennessee Valley Authority
T. Kovac	Commonwealth Edison Company
T. Murphy	GPU Nuclear Corporation
G. Powers	General Electric Company
F. Roddy	Bechtel Power Corporation
L. Smith	Institute of Nuclear Power Operations
K. Travis	Edison Electric Institute
C. Wood	Electric Power Research Institute

1. INTRODUCTION

On behalf of the Nuclear Regulatory Commission, the staff at Brookhaven National Laboratory's ALARA Center have been carrying out research in dose reduction and monitoring worldwide activities to reduce occupational radiation exposure. A many facetted program was developed to comply with the requirements of the Commission. Previous activities included a comparative assessment of foreign and U.S. nuclear power plant dose experience (Ref.3), an investigation on the cost-effectiveness of engineering modifications at nuclear power plants (Ref.4), an analysis of high-dose jobs and related dose-reduction techniques (Ref.5), an evaluation of worldwide activities on dose reduction (Ref.6), and a study which explored various options to reduce contamination at nuclear power plants (Ref.7). The ALARA Center also hosted an international workshop on historic dose experience and ALARA (Ref.8).

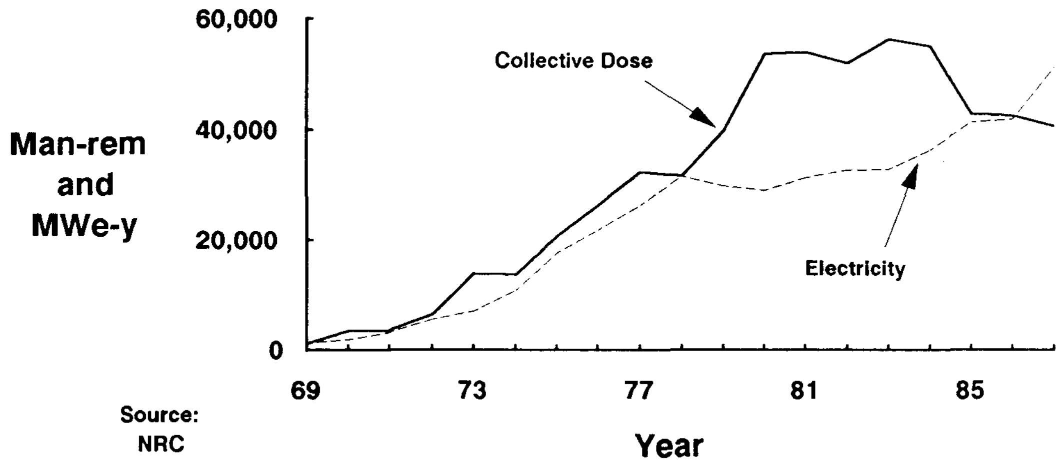

Fig.1: Collective Dose and Gross Electricity vs Year

A computerized, international data-base of information on dose-reduction research and health physics technology was been developed as a part of one project (NRC FIN A-3259). The information on the data base is continually updated and monthly summaries are provided to the NRC. Contributors to the data base receive information upon phone request, by means of electronic interrogation of the data base, or by periodic (approximately annual) mailings. Presently, there are 250 research projects and 135 projects on health physics technology in the data base, that are described in two annual reports, NUREG/CR-4409 Volumes 1 and 2 (Ref.1,2). In addition, bibliographies of selected readings in radiation protection and ALARA are periodically published, NUREG/CR-3469 Volumes 1, 2 and 3 (Ref.9,10,11). Volume 4 of the bibliography is scheduled for publication this year. This report is the third volume in the NUREG/CR-4409 series, which describe new projects in the data base.

2. TREND IN COLLECTIVE DOSE EQUIVALENT

Although occupational radiation exposures (ORE) to individuals have been kept well below the regulatory limits in the United States, the collective occupational dose equivalents at US nuclear power plants have shown large increases over time (Ref.12). Between 1969 and 1978 the annual collective dose increased at roughly the same rate as the total amount of electricity produced by the plants. However, after 1978 the electricity generated was nearly constant for several years but the collective occupational dose rose steeply (Fig.1).

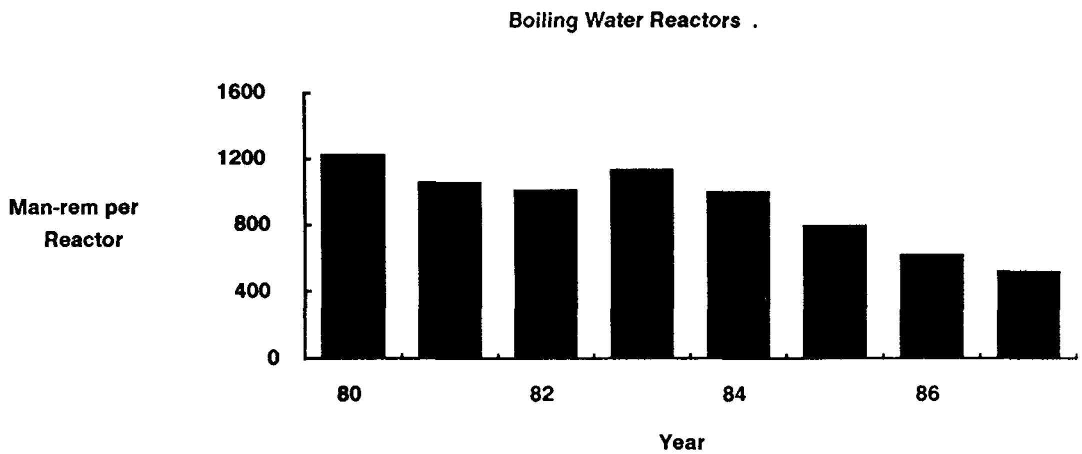

Fig.2: Collective Dose per Reactor

A part of the increase in ORE could be attributed to the multi-plant actions that were mandated after the Three Mile Island 2 accident (Ref.13). Nevertheless, it was necessary to ascertain that appropriate efforts were being made to reduce ORE in accordance with the ALARA principle.

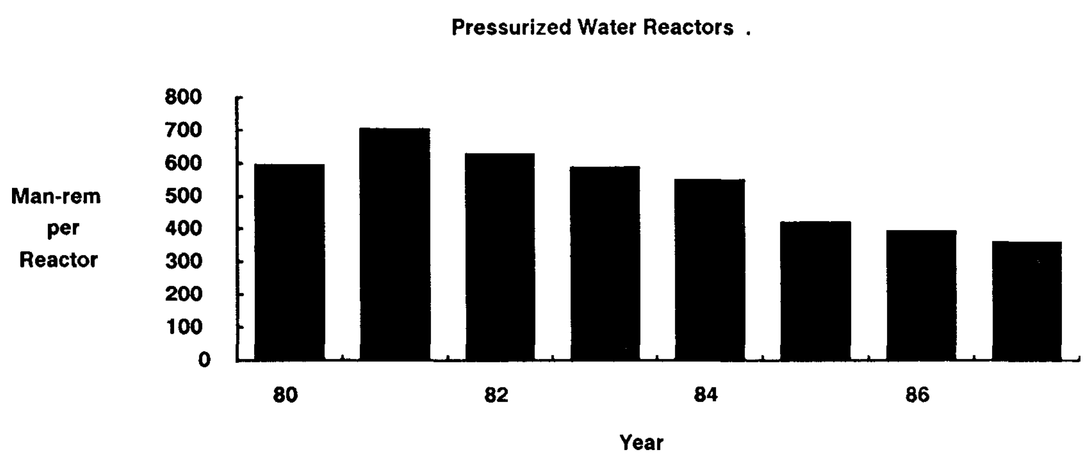

Fig.3: Collective Dose per Reactor

In recent years there has been a monotonic decrease in collective dose per reactor in the United States. The doses for BWRs have declined every year since 1983, and for PWRs since 1981 (see Figures 2 and 3). Although the dose per reactor has been consistently higher for BWRs, the decrease in collective dose has been more definite for this type of plant in the last three years.

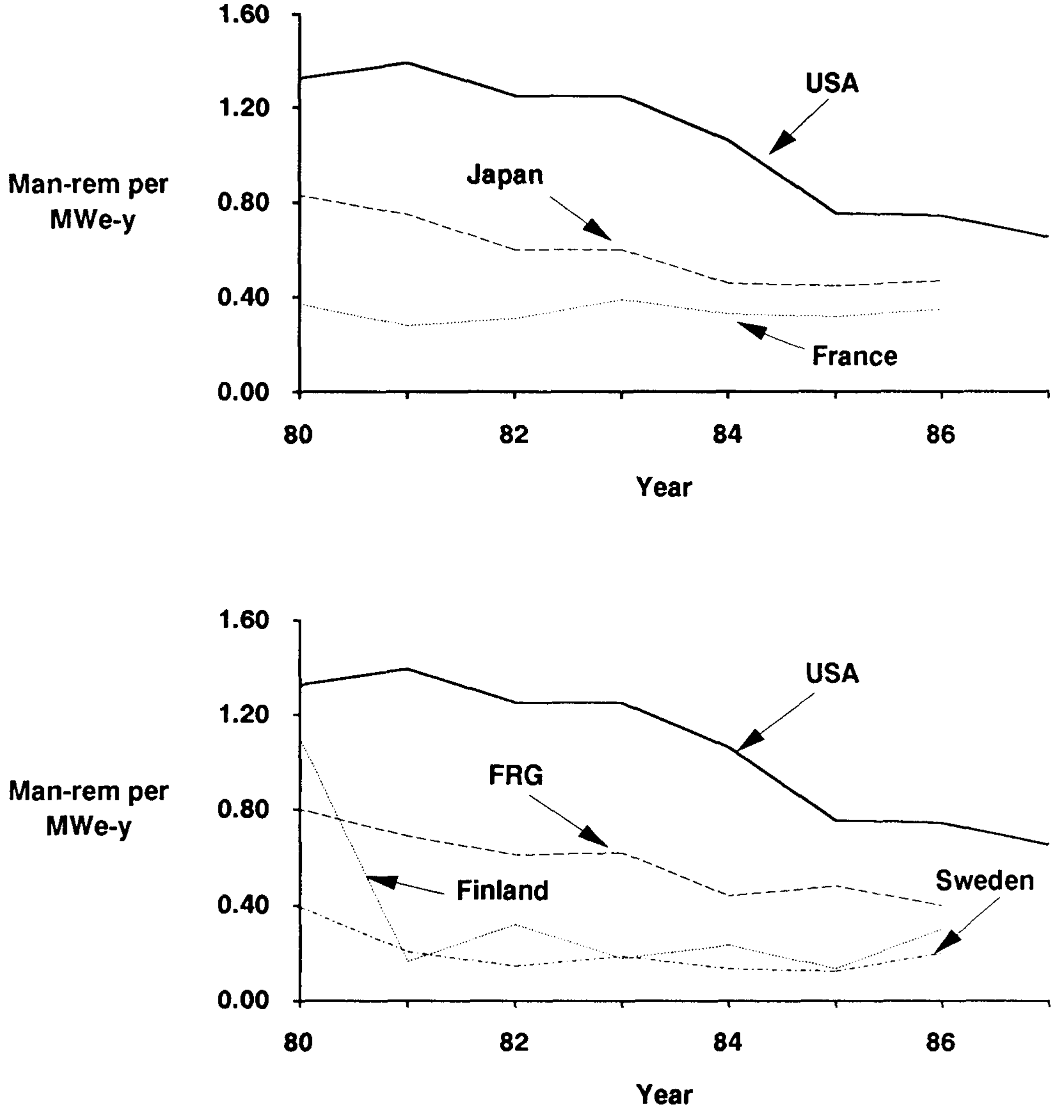

Fig. 4: Comparison of Annual Collective Dose at PWRs

In Figures 4 and 5 the collective exposures in the United States are compared with those in other countries with significant nuclear power generation. For this comparison, the data were normalized to units of collective dose per unit of electricity generated. Although the U.S. doses are higher, their general downward trend is encouraging. Part of the reason for the higher U.S. doses may be because many U.S. plants are older and were designed when much less information was available on reducing the radiation dose at the design stage. A recent study on PWRs showed that the older plants had the relatively higher doses (Ref.14).

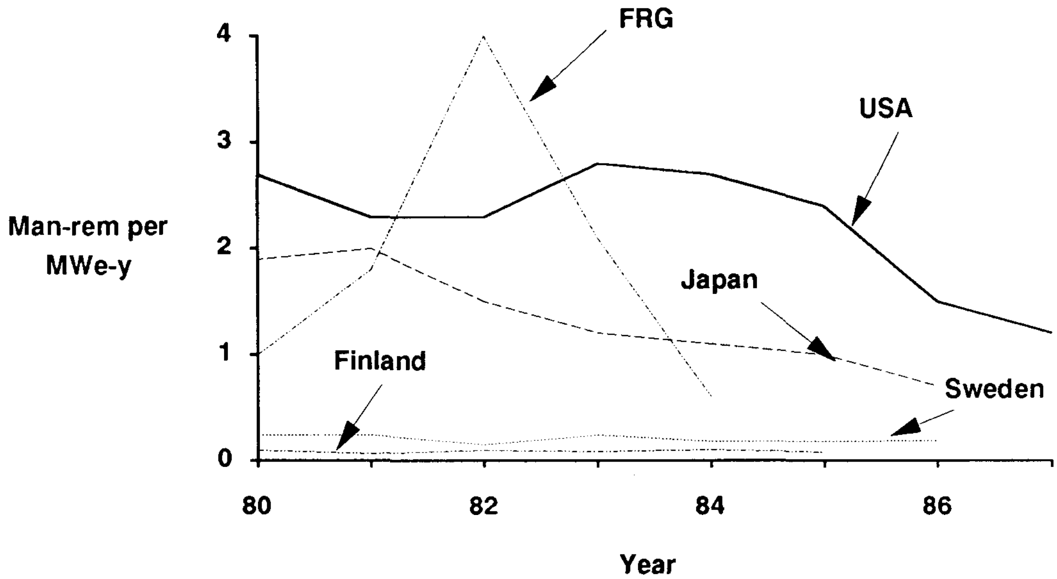

Fig. 5: Comparison of Annual Collective Dose at BWRs

3. RECENT DEVELOPMENTS IN DOSE REDUCTION

The information sheets in this report show an assortment of activities to reduce exposures at nuclear power plants, ranging from studies on steam generator tube degradation (E-194, 197, 199, 202, 203, 204, 205, 211, 225, 243), to materials improvements in steam generators (E-194, 198, 201, 243,) and other plant components (E-193, 207, 213,). There are several sheets on full-system decontamination (E-232, 234, 241, T-130) which may be the technology providing the largest reductions in collective dose once it is qualified. Another area of technology showing promise is that of robotics (E-218, 219, 220, 221, 222, 223, 224, 229, 246, 247, 248, 249). The capabilities of robots are rapidly increasing in functionality, sophistication and durability, and they are likely to play an important role in dose reduction in the future. Other areas worth attention are the possibility of replacing natural boron with isotopically enriched boron to control the reactivity of PWRs (E-236, T-127), and the use of advanced inspection technology (E-209, 213, 214, 224, 229, 230).

There are some interesting new international projects: For example, there is information on steam generator tube performance (E-228) and replacement strategy (E-226) from France, on fission product transport (E-195) and on steam generator degradation (E-197, 198) from Canada, on robotics (E-218, 229) from Belgium and the Netherlands. The Kansai Electric Power Company of

Japan has provided information on operating experience (T-128), and there is a project on robotics from Hitachi. Sweden provided information on the Ozone Decontamination Process (E-237). There are several projects from F.R. Germany, on antimony removal (E-241), on robotics (E-219, 220, 223), on reducing radiation fields (E-242), and on decontamination (E245). The Dutch/German effort on full-system decontamination is described in sheet (E-241)

The health physics technology section of the data base has some very simple and yet cost-effective techniques, such as a valve identification program (T-117), the use of closed circuit television for remote visual inspections (T-118), the utilization of extended-life light bulbs (T-121), and effective and innovative uses of shielding (T-114, 122, 135). Sheets T-115 and 116 describe the optimization of radiation exposure in Japanese and in Finnish plants, respectively. The optimization of contamination control is described in T-119. There are also projects on ultra filtration (T-124), snubber reduction (T-125, 132) and the chemical decontamination of components (T-126). A Japanese and a U.S. utility described their plant-wide ALARA plans (T-128, 129).

There are two particularly noteworthy projects in this section of the data sheets. T-133 describes a British project in which workers at high-dose jobs are watched by TV cameras, and information about their dose and dose-rate is superimposed on the picture, so identifying any procedures where the radiation hazard is high. By recording the combined display, the exact details of the exposure can be seen and protection measures targeted to the maximum effect. One can see important uses of this system in work planning and in training. Moreover, the health physicists and other operations persons not directly involved in the task, such as advisory personnel, can remotely monitor high-dose tasks without incurring unnecessary occupational exposures.

The surrogate tour interactive system, described in sheet T-134, is also noteworthy. It combines the technology of the laser video disc and personal computers to simulate motion and provide detailed visual information of contaminated and restricted plant areas. The system can also use the latest radiation-survey information to superimpose radiation levels on floor plans and may be used to generate radiation survey maps. Again, there is considerable potential for dose reduction if the system is effectively utilized in work planning, in training, and in maintaining complete visual records of the radiation areas.

4. RELATED PROJECTS

This section of the report provides information on other related work of the ALARA Center to ensure that users of the data base are aware of some relevant services and facilities provided by the Center.

4.1 International Workshops

We plan to hold periodic international workshops (about every four years) like those described in Reference 8. Apart from being an excellent vehicle for exchanging ALARA-related information, the material from these workshops will augment the information in the data base. The next workshop, jointly sponsored by the NRC, the DOE, and the ALARA Center, in co-operation with the Nuclear Energy Agency of the OECD, is scheduled for September 1989. Additional information about the 1989 workshop is provided in section 7.1.

4.2 Electronic Access to the Data Base

Recently, the data base was made available to users through an on-line link, using a personal computer and a modem. There is also an electronic bulletin board service where users can deposit and retrieve information and exchange ideas and experiences. Section 7.2 gives additional information on the bulletin board.

4.3 Collaboration with the Nuclear Energy Agency

On a wider scale, the need for the kinds of activities that the ALARA center is carrying out, led the Nuclear Energy Agency of the Organization for Economic Cooperation and Development to propose an extension of this program encompassing Western Europe, Japan, Canada, and the United States. They are considering having regional centers to collect and exchange dose-related information from member countries, analyze the information, and make it available to interested parties including the nuclear industry and national regulatory organizations. The ALARA Center is collaborating with the NEA in this program.

5. Conclusion

The status of occupational radiation exposure at nuclear power plants is changing very rapidly. The National Council on Radiation Protection and Measurement (NCRP) recently recommended an annual dose limit of 5 rem for occupational workers (Ref.15). It also recommended that the numerical value of the individual worker's lifetime effective dose equivalent in units of rem should not exceed their age in years. The National Radiological Protection Board of the U.K. already recommended a lower dose limit of 1.5 rem per year for occupational workers (Ref.16). The International Commission on Radiological Protection is examining the implications of the revised data on the Japanese atomic bomb survivors, and will make a new recommendation at the end of 1988 (Ref.16). This trend towards lower dose limits will require an even greater emphasis on ALARA practices and procedures at nuclear power plants. New techniques will have to be developed to optimize dose reduction practices and to better match the number of personnel to task requirements.

The designs of the new generation of nuclear plants, however, are likely to meet the new requirements, since they are being designed for extremely low annual collective doses. The 1200 MWe Sizewell 'B' PWR, designed by Westinghouse Electric Corporation for the U.K., is expected to have annual collective dose expenditures of 240 person-rem per year (Ref.17). The advanced boiling-water reactor described in sheet E-196 is projected to have average annual plant collective dose of just 40 person-rem. Advanced pressurized-water plants with low collective dose characteristics were described previously (Ref.6).

Looking farther into the future, the 300 MWe Thorium High Temperature Gas Cooled Reactor, which was designed to demonstrate HTR technology and has been producing electricity in F.R. Germany for the last two years, required annual collective doses of 5 and 10 person rem in its first two years of operation, including inspections and maintenance (Ref.18). Similar and other advanced technologies are also under development in the United States and in other countries with major nuclear power programs (Ref.19,20). Thus, the future looks promising for dose reduction and ALARA, as new advanced designs drastically reduce occupational exposures.

However, the bulk of today's collective dose is produced at standard light-water plants and this is likely to remain so for a large number of years. Thus, until a new generation of nuclear plants have superseded the standard light-water reactors, it will be necessary to continue to master the current technology in order to make a sizeable dent in collective dose expenditures. Thus, to reduce occupational dose in the foreseeable future will require perseverance with research and development, and an aggressive approach to health physics technology.

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7. INFORMATION ON RELATED ACTIVITIES

7.1 International Workshop Information

SECOND ANNOUNCEMENT AND CALL FOR ABSTRACTS

INTERNATIONAL WORKSHOP ON NEW DEVELOPMENTS IN OCCUPATIONAL DOSE CONTROL AND ALARA IMPLEMENTATION AT NUCLEAR POWER PLANTS AND SIMILAR FACILITIES

BROOKHAVEN NATIONAL LABORATORY
UPTON, NEW YORK 11973
SEPTEMBER 18-21, 1989

Brookhaven National Laboratory will be the site for the *International Workshop on New Developments in Occupational Dose Control and ALARA Implementation at Nuclear Power Plants and Similar Facilities* planned for September 18-21, 1989. The workshop is sponsored jointly by the U.S. Nuclear Regulatory Commission (NRC) and the Department of Energy (DOE) in cooperation with the Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency.

Invited and contributed papers and workshop discussions will be held on the following topics.

ALARA organization in design and operation, including design reviews and operational planning, studies of overall plant optimization, results from the Nordic study, mock-up training, and training workers for self-monitoring.

ALARA engineering in design and modifications, including cobalt reduction in nuclear plants, special shielding, cost-benefit analyses and other decision-aiding techniques, remote tooling and robotics, and rebuild facilities for control-rod drive.

System chemistry and water purification, including zinc injection, pH control, hydrogen water chemistry, ultra-fine filtration, and passivation of pipes and components.

ALARA in operation, including maintenance work, system (and component) decontamination, remote inspections and surveillance, and start-up and shutdown procedures.

Recommendations and regulations of groups such as ICRP/NCRP, IAEA, INPO, CEC, DOE, and NRC.

Abstracts and Summaries: Participants are expected to submit either an abstract (300 words or less) of a formal presentation, or a one-page summary of information on a topic intended for discussion by March 1, 1989. The title should be followed by the names of the authors, their mailing addresses, and the telephone number of the senior author. Participants will be limited to 100 and will be notified of acceptance by May 15, 1989.

Manuscripts: Formal, full-length manuscripts and final versions of summaries and abstracts will be due August 15, 1989. Complete instructions for their preparation will accompany the notice of acceptance.

Fees and Housing: Registration fees, including four nights of on-site housing in residences (single room, shared bath) will be $250, payable 30 days in advance. Additional information on transportation, other housing available at additional cost, and other matters will be mailed at a later date.

Abstracts and summaries should be addressed to:

Dr. John W. Baum
BNL ALARA Center
Building 703M
Upton, NY 11973

Telephone:	(516) 282-4214
Telex:	6852516 BNL DOE
Fax:	(516) 282-5810

10/10/88

Tentative Agenda

INTERNATIONAL WORKSHOP ON NEW DEVELOPMENTS IN OCCUPATIONAL DOSE CONTROL AND ALARA IMPLEMENTATION AT NUCLEAR POWER PLANTS AND SIMILAR FACILITIES

BROOKHAVEN NATIONAL LABORATORY
UPTON, NEW YORK 11973
SEPTEMBER 18-21, 1989

Monday Evening, September 18

7:00-9:00 p.m. **Registration/Mixer**

Tuesday, September 19

Session 1: **Invited and Contributed Papers on ALARA Status, Studies, and Organization**

- o ALARA - Past, Present, and Future
- o NEA/OECD Overview Report on Occupational Exposure
- o Results from Major Studies, e.g., the Nordic Study

BREAK

Session 2: **Workshop Discussions on Session 1 Topics and:**

- o ALARA Policies and Procedures
- o ALARA Committees
- o Photo and Video Documentation

LUNCH

Session 3: **Invited and Contributed Papers on ALARA Engineering in Design and Modifications**

- o Cobalt Reduction in Nuclear Power Plants
- o Special Shielding
- o Remote Tooling and Robotics
- o Control Rod Drive Rebuild Facilities
- o Cost/Benefit Analyses and Other Decision-Aiding Techniques
- o Overall Plant Optimization Studies

BREAK

Session 4: **Workshop Discussions on Session 3 Topics and:**

- Control Rod Drive Improvements
- Snubber Reductions
- Platforms, Ladders, Rigging, Scaffolding
- Multi-stud Tensioners
- QA of Cobalt Reductions

Tuesday Evening, September 19 - Mixer/Dinner

* *

Wednesday, September 20

Session 5: **Invited and Contributed Papers on System Chemistry and Water Purification**

- Zn Injection
- pH Control
- Hydrogen Water Chemistry
- Ultra-fine Filtration
- Passivation of Pipes and Components

BREAK

Session 6: **Workshop Discussions on Session 5 Topics and:**

- Reactor Water Cleanup System
- Magnetic and High Temperature Filters
- Condensate Prefilters
- O_2, Fe, Cl, and Conductivity Controls

LUNCH

Session 7: **Invited and Contributed Papers on ALARA in Operation**

- In-Service and other Inspections, and Surveillance
- Maintenance Work
- Startup and Shutdown Procedures
- System (and Component) Decontamination
- Remote Inspections and Surveillance
- Radioactive Waste Management
- Training

BREAK

Session 8: **Workshop Discussions on Session 7 Topics and:**

- o Remote Communications
- o Mock-Up Training
- o ALARA Suggestion Systems
- o Training Workers for Self Monitoring

Wednesday Evening, September 20 - Mixer/Chamber Music

* *

Thursday, September 21

Session 9: **Invited and Contributed Papers on Recommendations and Regulations**

- o ICRP/NCRP-Related Work
- o IAEA-Related Work
- o CEC Programs in Dose Control (ALARA)
- o INPO Activities in Dose Control (ALARA)
- o DOE ALARA Studies
- o NRC Regulatory Plans

Session 10: **Workshop Discussions on Session 9 Topics and:**

- o Impacts of New Recommendations and Regulations
- o Hot Particle Problems

BREAK

Session 11: **Presentation and Discussion on the Plans for the OECD/NEA Occupational Exposure Information System**

LUNCH

Thursday, September 21, 1:30 to 4:30 p.m.

- o Post-Workshop Meetings to Continue Discussions and Plans for the NEA Information System

- o BNL Lab Tours if Desired

For information contact BNL ALARA Center, (516) 282-4425

7.2 Electronic Access Information

ALARA CENTER DOSE-REDUCTION RESEARCH DATA BASE ACCESSIBLE THROUGH COMPUTERS

The Brookhaven National Laboratory's ALARA Center is pleased to announce that the information in its data base on dose reduction research and health physics technology programs is now accessible to personal computer users on-line by means of a modem. The information may be accessed by calling (516) 282-3481 or FTS 666-3481. Access will be available continuously morning and night from 11 a.m. on Monday to 5 p.m. on Friday. The system will not be available on weekends. For assistance on linkup and advice on rapid communication procedures please call Dr. Tas Khan at (516) 282-4012 or FTS 666-4012.

Description of features

Technical advice on how to access the bulletin board is provided on separate sheets and also on the welcome screen when connection is made to the ALARA Center's computer. The information below describes the main features available as of this writing.

Basic Service:

Users of the service will have access to the directory H:\ALARA, which will contain all pertinent information. It will be the current directory when a link is established. All information has been split into three groups: The first group contains information from the ALARA Center's data base on dose reduction. All the files describing this information have extensions .TXT or .DOC. The second group contains information from the ALARA Center's BIBLIOGRAPHY. These files have the extension .BIB. The third group contains other information and have the extension .ALA.

Group 1 Files:

New information, which becomes available for the data base, will be maintained as information sheets analogous to those published periodically in the NUREG/CR-4409 series of reports. The sheets beginning with the letter E describe research projects, those starting with T describe health physics technology projects. The sheets will be identified by file names which use their BNL Identification Number and the DOS extension TXT. For example, the sheet E-235 has the file name E235.TXT. As new information comes in, new sheets will be added and the existing ones updated. The information sheet files on dose-reduction projects are typically 2000 Bytes. Sheets with the extension .TXT are in ASCII format. You may wish to download the file TITLES.TXT which is the key to the data and describes the information sheets. This 18,000 Bytes long file lists the currently available sheets by title and number and outlines their contents.

Group 2 Files:

The files with the extension .BIB provide basic information on papers and reports that will be published in volume 4 of the ALARA Center's bibliography. The files TITLES1.BIB and TITLES2.BIB list the titles of the abstracts in the bibliography. The first file lists the various articles which have been abstracted; the second file lists all the reports. The files AUTHORS.BIB and SUBJECTS.BIB are the author and suject indices. All these files are in ASCII format. The bibliography's abstract numbers are given in all the indices so that the relevant abstracts may be conveniently downloaded from the bulletin board.

*Material since last update is shown in italics

The abstracts themselves have been grouped into a number of small files each containing 25 abstracts. Thus the file A826.BIB contains 25 abstracts starting with abstract number 826.

Users are advised to download and examine the various index files to decide which abstracts are of interest before they download the appropriate file containing those abstracts.

Group 3 Files

Files with the extension .ALA are short reports, papers, notes, dose data, etc. The files in this group are contributed by the ALARA Center, by the users, and other sources available to the ALARA Center. The files named ALARA-1.ALA, ALARA-2.ALA etc. will be provided by the ALARA Center. Files with other names are from users.The file KEY.ALA provides information on what is available in this series of files.

Some software is also available in this group. The programs NUCOST.BAS and RCOST.BAS are in BASIC . There are also some templates for SYMPHONY and LOTUS 1-2-3. These programs help in controlling contamination and optimizing radiation exposure. They are described in NUREG report NUREG/CR-5038.

Contributing Information for Other Users

Users may contribute information on dose reduction, for example a technique, a procedure, notes, short papers, reports, dose information, software, etc. Files containing such information may be uploaded to the system. Depositors should include their name, address and telephone number at the beginning of the file and leave a message in the electronic mail to indicate that a file on a topic has been deposited. The file extension .ALA should be used for these files. Any appropriate file name may be used except ALARA-n, (where n is a number), since these file names are reserved for the ALARA Center. For example, a contribution from the Beaver Valley plant could have the file name BV-1.ALA. The file KEY.ALA will provide a table listing the title of each such contribution and the name and affiliation of the contributor.

Exchange of Information by Electronic Mail

A simple electronic-mail service will also be available to users of this facility. This may be used for short notes to all users, to specific users or to the ALARA Center. Through this means advice may be sought or short comments made on radiation protection and other appropriate subjects. When a file is deposited, a brief mail message to users, describing the file and giving the name, affiliation and phone number of the contributor, may be appropriate.

Information by FAX

Information may also be transmitted to the ALARA Center by FAX. The ALARA Center FAX number is (516) 282-5810. For further information, please contact Dr. Tas Khan.

Technical Aspects

Every user is presented the following menu after successfully logging on and viewing the news file.

```
F)iles U)pload D)ownload
H)elp T)ime C)hat G)oodbye
R)ead mail L)eave mail
```

An available function is invoked by pressing the first letter of the name. No carriage return is needed so to download, for example, just type the letter **D** .

Downloading Files

To download the sheets (i.e. retrieve the information in the sheets by computer and modem) set your modem and communications software to 2400, 1200 or 300 baud, 8 data bits, 1 stop bit, no parity. Dial 516-282-3481 or FTS 666-3481. Depending on your communication software, it may be necessary to hit CR twice after the connection is made. The following message should follow: "Welcome to the BNL ALARA Center". When prompted, give your first and then last name, and the password "ALARA". An explanatory screen will follow which will describe the latest information on accessing the bulletin board and using its features. When you are presented with the menu, select "Download" by typing the letter "D". At the next menu select the mode of file transfer (**X** for XMODEM, **K** for KERMIT, etc.). Give the file spec, e.g., E225.TXT etc., and complete the download by carrying out the appropriate download procedure at your computer. Each sheet file is comparatively short (about 2,000 bytes). To select which information sheets to download, it may be desirable to first download the file TITLES.TXT (18,000 bytes). An analogous procedure to deposit (*Upload*) information is described below and on the screen when help is called for by typing the letter **H**.

For Users of WORD and WRITE

The information sheets with the extension TXT are in ASCII format. Persons that have access to the Microsoft word processors WORD or WINDOWS WRITE (or to programs that translate from these word processors to another) may download elegantly formatted information sheets and produce hard copies with the word processors. The sheets were formatted for an HP Laserjet II printer and the Z cartridge using the proportional Helvetica font. However, other printers should also produce appealing results. The formatted files have the extension .DOC. To transmit the sheet E-225 in WORD format, download file E225.DOC. The downloaded files may also be readily converted to the WINDOWS WRITE format.

Function Description of Menu Commands

CHAT: Sounds the speaker on the host machine and waits for F1 to be hit on the host keyboard. If the host operator presses F1, the host machine is placed into chat mode. If any other key is pressed on the host keyboard, paging is stopped and a message is sent to the user. A user can abort a page request by pressing Ctrl-C.

DOWNLOAD: The user is prompted for a protocol to use and then for the specification of the file(s) to download from the host. Only files in the Host download directory *(which will be the current directory, H:\ALARA*) are allowed. When a valid file specification is entered, the message "Begin your ??????? transfer procedure" is issued and the host waits for the user to download the file(s) using the specified protocol.

FILES: Prompts for a file specification (like DOS' DIR command) and displays a list of matching downloadable files. A user can cancel the file display by pressing Ctrl-C. *The files displayed will be those from the current directory, H:\ALARA.* Specifications such as *.* or *.TXT or TITLES.TXT may be used. For example, to view Group 1 files, use specification *.TXT or *.DOC; to view Group 2 files use specification *.BIB; to view Group 3 files use *.ALA. To view the available software, use *.BAS and *.WK1.

GOODBYE: Terminates user and recycles the host to answer the next call.

HELP: Displays the help file:PCPLUS.HHP. A user can cancel the help display by pressing Ctrl-C.

LEAVE MAIL: The user is prompted for the following:

To: (the intended recipient of the mail)
Re: (the subject of the mail)
Private Mail (Y/N) (Y to limit viewing, N for public)

The user is then placed in a line-at-a-time input mode which continues until an empty line is entered. When an empty line is entered, the user is prompted with:

S)ave
A)bort
D)isplay
C)ontinue

Save: Append the message to the mailbase and return to the main menu.
Abort: Erase the message and return to the main menu (after confirmation).
Display: Show any text which has been entered (using the same format as the Read Mail facility)and display this prompt again.
Continue: Return to input mode.

READ MAIL: The user is prompted to choose one of the following:

F)orward read
S)earch mail
I)ndividual read
Q)uit

Forward Read: Sequential multiple read. Prompts for the message number to begin displaying and begins to display all accessible messages starting with that number.

Search mail: Selective sequential multiple read. Asks for a field to search, for example "To" or "Re" of LEAVE MAIL, then a string (a set of letters of the alphabet e.g. "All Users") to search for,and finally what message number to begin the search with. A display of all accessible messages which match the search criteria is then begun.
Individual read: Single message read. Asks which message to read and displays it if accessible.
Quit: Return to the main menu.

TIME: The time the user came on-line is displayed, followed by the current time.

UPLOAD: The user is prompted for a protocol to use for the transfer. A file specification is then asked for, followed by a file description. The host then waits for the user to begin using the specified protocol to upload the file(s). Uploaded files are placed in the Host mode upload directory, *H:\ALARA which will be the current directory.*

8. INFORMATION REQUEST FORM

BNL ALARA Center Data Base

COUNTRY

1. Title:

2. Investigator(s):

(Name)
(Organization)
(Mailing Address)
(Phone)

} Contracting Organization

3. Project Manager

(Name)
(Organization)
(Mailing Address)
(Phone)

} Sponsoring Organization

4. Objectives:

(Outline main objectives of the project.)

5. Comments:

(Give any comments which would shed light on project, e.g., regarding background, perspective, progress, or significant findings.)

6. Potential for dose limitation:

(Even though the principal objective of the project may not be exposure reduction, indicate any anticipated beneficial or detrimental dose impacts or potentials.)

7. Duration: From: 19 To: 19

8. Funding: (Amount or person-years)

9. Status: (Proposed, initiated, in progress, terminated)

10. References:

(Recent reports, articles, or publications related to the project.)

11. Key Words:

(Broad heading followed by descriptors or key words.)

Return to: Dr. John W. Baum, Brookhaven National Laboratory, BNL ALARA Center, Building 703M, Upton, NY 11973. Phone: 516-282-4214

BNL ALARA Center Data Base

1. Title:	
2. Investigator(s):	3. Project Manager
4. Objectives:	
5. Comments:	
6. Potential for dose limitation:	
7. Duration: From: 19 To: 19	8. Funding:
9. Status:	
10. References:	
11. Key Words:	

Return to: Dr. John W. Baum, Brookhaven National Laboratory, BNL ALARA Center, Building 703M, Upton, NY 11973. Phone: 516-282-4214

9. LIST OF PROJECTS

9. LIST OF PROJECTS

The projects in this series of reports have been divided into two main groups. The **E** projects are **E**ngineering Research type projects, whereas the **T** projects are more closely related to Health Physics **T**echnology efforts at nuclear power plants. In addition to this division, the projects have been indexed under six main categories so that they may be more easily accessed.

E/T	ID	TITLE
E	1	WATER CHEMISTRY AND RADIATION BUILD-UP AT THE LA SALLE-1 BWR
E	2	RADIATION CONTROL THE IN PRIMARY COOLANT LOOP OF PWR PLANTS
E	3	ON-LINE MONITORING TECHNIQUES FOR REDOX POTENTIAL, HYDROGEN CONCENTRATION, AND PH IN COOLANT CIRCUITS OF NUCLEAR COOLANT REACTORS
E	4	WATER REACTOR DECONTAMINATION
E	5	HIGH TEMPERATURE FILTRATION (HTF)
E	6	CONSTRAINTS ON THE INSPECTION, MITIGATION, REPAIR AND REPLACEMENT OF THE BWR PIPING SYSTEM
E	7	THE EFFECT OF SURFACE TREATMENTS ON THE IGSCC OF BWR MATERIALS
E	8	ANALYSIS OF BWR RADIATION BUILD-UP
E	9	DRESDEN UNIT 1 CHEMICAL CLEANING PROJECT
E	10	EVALUATION OF ALTERNATIVE TECHNIQUES FOR MONITORING MACHINERY VIBRATION
E	11	DEVELOPMENT OF A MONITORING AND DIAGNOSTIC SYSTEM FOR THE REACTOR COOLANT PUMP
E	12	ACOUSTIC LEAK DETECTION IN LIGHT-WATER REACTORS
E	13	AN EXPERIMENTAL STUDY OF ADVANCED ACOUSTIC EMISSION CONTINUOUS MONITORING OF LIGHT WATER REACTORS
E	14	OXYGEN SUPPRESSION IN BWRs - PHASE II
E	15	CHEMICAL CLEANING OF STEAM GENERATOR
E	16	EVALUATION OF TECHNIQUES FOR DECONTAMINATION OF REACTOR COOLANT
E	17	NOBLE GASES AND HALOGENS DISSOLVED IN HOT WELLS
E	18	CARBON DIFFUSION IN DISSIMILAR WELDS
E	19	EFFECTS OF IMPURITIES ON BWR WATER CHEMISTRY
E	20	RADIOISOTOPIC TRANSFER FROM CORE TO BALANCE OF PLANT
E	21	INVESTIGATION OF BWR INTERGRANULAR STRESS CORROSION CRACKING
E	22	SOLUBILITY AND PARTITION OF HALOGENS

E/T	ID	TITLE
E	23	EFFECTIVENESS AND SAFETY ASPECTS OF SELECTED DECONTAMINATION METHODS FOR LWRs
E	24	WESTINGHOUSE ALARA PROGRAM
E	25	PWR pH/RADIATION CONTROL TEST IN WATTS BAR UNIT 1
E	26	FLOW-INDUCED VIBRATION FOR LIGHT WATER REACTORS
E	27	PWR STEAM GENERATOR OWNERS GROUP - 1
E	28	EVALUATION OF HIGH TEMPERATURE FILTRATION IN PWRs
E	29	DECONTAMINATION OF DILUTE SOLVENT PWR
E	30	ANNULAR PELLET HIGH BURN-UP FUEL FOR PWR
E	31	ACOUSTIC MONITORING IN POWER PLANTS
E	32	IMPROVEMENT IN CONDENSER RELIABILITY
E	33	DEVELOPMENT OF CHEMICAL DECONTAMINATION REAGENTS
E	34	WATER CHEMISTRY OF BWRs
E	35	STRESS CORROSION OF BWRs
E	36	IMPROVEMENT IN RECIRCULATION PUMP SEAL RELIABILITY
E	37	HIGH TEMPERATURE FILTRATION - 2
E	38	COMPONENT DECONTAMINATION
E	39	OPTIMIZATION OF RADWASTE SYSTEM
E	40	PROTECTION OF PIPING INTERNAL SURFACE
E	41	IMPROVEMENT IN SNUBBER RELIABILITY
E	42	ROBOTIC MAINTENANCE
E	43	REPLACEMENT OF CONTROL ROD STUB TUBE
E	44	RUBBER GOODS MANAGEMENT
E	45	IN-CORE INSTRUMENTATION LIFE
E	46	ADVANCED FUEL MANAGEMENT
E	47	DEMONSTRATION OF ADVANCED IN-SERVICE INSPECTION

E/T	ID	TITLE
E	48	MONITORING OF ROTATING EQUIPMENT VIBRATION
E	49	IMPROVEMENT IN FUEL CHANNEL LIFE
E	50	BWR PIPECRACKING - PIPELOCKS
E	51	BWR PIPECRACKING - MECHANICAL STRESS IMPROVEMENT
E	52	IN-SERVICE INSPECTION - MINAC RADIOGRAPHY
E	53	REMOTELY OPERATED GENERATOR EXAMINATION AND REPAIR (ROGER)
E	54	REMOTELY OPERATED SERVICE ARM (ROSA)
E	55	SRL WALKING ROBOT
E	56	REMOTE MANIPULATION AND CONTROL SYSTEM FOR CANDU RETUBING
E	57	DEVELOPMENT OF DECONTAMINATION TECHNIQUES FOR PWRs
E	58	HYDRAULIC SNUBBER IN-SERVICE OPERABILITY CRITERIA FOR SAFETY-RELATED PIPING SYSTEMS
E	59	HIGH TEMPERATURE FILTERS FOR SECONDARY SIDE OF PWR STEAM GENERATORS
E	60	PWR STEAM GENERATOR OWNERS GROUP - 2
E	61	DILUTE DECONTAMINATION OF PRESSURIZED WATER REACTORS
E	62	PWR STEAM GENERATORS OWNERS GROUP - PHASE I
E	63	PWR STEAM GENERATOR OWNERS GROUP - 3
E	64	IMPROVEMENTS IN MAIN COOLANT PUMP SHAFT SEAL
E	65	OPTIMIZATION OF SECONDARY SYSTEM WATER TREATMENT
E	67	OPTIMIZATION OF CORPORATE RADIATION DOSE EXPENDITURES
E	68	BWR HYDROGEN WATER CHEMISTRY-CHEMICAL MONITORING
E	69	BWR ALTERNATE WATER CHEMISTRY - RADIOLOGICAL MONITORING
E	70	HYDROGEN WATER CHEMISTRY - FUEL MATERIALS
E	71	BWR RADIATION CONTROL - PLANT DEMONSTRATION
E	72	DEVELOPMENT OF COBALT-FREE HARDFACING ALLOYS FOR NUCLEAR APPLICATIONS

E/T	ID	TITLE
E	73	FIELD WEAR MEASUREMENTS OF COBALT ALLOYS
E	74	FIELD TESTING OF WEAR-RESISTANT COBALT-FREE ALLOYS/NUCLEAR VALVES
E	75	BWR RADIATION CONTROL-PLANT DEMONSTRATION
E	76	WEAR OF HIGH COBALT ALLOYS IN VALVES
E	78	EVALUATION OF LOW COBALT ALLOYS FOR HARD-FACING APPLICATIONS IN NUCLEAR COMPONENTS
E	80	OPTIMIZATION OF PWR FUEL DESIGN
E	81	HYDROGEN WATER CHEMISTRY-BOILING WATER REACTORS
E	82	WATER CHEMISTRY MEASUREMENTS AT RINGHALS-1
E	83	ANALYSIS OF FAILURE OF MAIN COOLANT PUMP SHAFT SEAL
E	84	ROBOT APPLICATIONS FOR NUCLEAR POWER PLANTS
E	85	DEVELOPMENT AND TESTING OF HIGH-BURN-UP FUEL
E	86	EXPERIENCE WITH COBALT-FREE ALLOYS IN VALVES
E	87	ALARA AWARENESS SEMINAR
E	88	RELEASE RATES OF COBALT CORROSION PRODUCT
E	89	IN-PLANT SYSTEM FOR CONTINUOUS LOW-LEVEL ION MEASUREMENT IN STEAM PRODUCING WATER
E	90	DEVELOPMENT OF NONCHEMICAL DECONTAMINATION TECHNIQUES
E	91	LOMI PROCESS
E	92	REMOTE HANDLING EQUIPMENT FOR GAS-COOLED REACTOR INSPECTION, MAINTENANCE, AND DECOMMISSIONING
E	93	REFINEMENT OF LIQUID ABRASIVE DECONTAMINATION FOR NUCLEAR APPLICATIONS
E	94	THE C/V ROBOT PROJECT
E	95	APPLIED OPTIMIZATION OF RADIATION PROTECTION AT NUCLEAR POWER PLANTS
E	96	STRESS IMPROVEMENT REMEDIES FOR IGSCC

E/T	ID	TITLE
E	97	DEVELOPMENT OF COUNTERMEASURES AGAINST IGSCC IN LARGE DIAMETER PIPE
E	98	DEVELOPMENT OF ROBOTICS FOR NUCLEAR POWER PLANTS
E	99	DEVELOPMENT OF IMPROVED TECHNIQUES TO INSPECT GUIDE TUBE SUPPORT PINS IN PWRs
E	100	DEVELOPMENT OF TECHNIQUES FOR UNMANNED INSPECTION AT HINKLEY POINT AGR
E	101	DEVELOPMENT OF STEAM GENERATORS FOR THE ITALIAN PWR
E	102	SELF-STRIPPING DECONTAMINATION POLYMERS
E	103	REMOTELY OPERATED CONCRETE FLOOR DECONTAMINATION MACHINE
E	104	DEMONSTRATION OF ROBOTIC APPLICATIONS AT THE WEST VALLEY PROJECT
E	105	ROBOTIC MANIPULATOR SYSTEM: RM-10
E	106	DEMONSTRATION PROJECT FOR ROBOTIC INSPECTION SYSTEM AT NUCLEAR POWER PLANTS
E	107	EVALUATION OF ROBOTIC INSPECTION SYSTEMS AT NUCLEAR POWER PLANTS
E	108	REACTOR CAVITY INSPECTION ROBOT
E	109	REDUCTIVE DECONTAMINATION TECHNIQUE
E	110	DEVELOPMENT OF RADIATION BUILD-UP MODELING FOR BWRs
E	111	DEVELOPMENT OF LIVE LOAD VALVE STEM PACKING
E	112	EVALUATION OF TECHNIQUES FOR OCCUPATIONAL EXPOSURE REDUCTION IN PWRs
E	113	ANALYSIS OF PAST EXPERIENCE IN RADIOLOGICAL PROTECTION AT PWRs
E	114	IMPROVEMENT PROGRAM FOR STEAM GENERATOR RELIABILITY
E	115	FURTHER DEVELOPMENT OF THE CONVOY DESIGN FOR PWRs
E	116	DEVELOPMENT PROGRAM FOR STEAM GENERATOR FOR PWRs
E	117	DEVELOPMENT OF REMOTE AND AUTOMATIC TOOLING FOR NUCLEAR POWER PLANTS

E/T	ID	TITLE
E	118	DEVELOPMENT OF THE PLANISOL-M DETERGENT
E	119	DEVELOPMENT OF TECHNIQUES TO OVERCOME STRESS CORROSION CRACKING IN BWRs
E	120	SELECTION AND TESTING OF ALTERNATE MATERIALS TO MINIMIZE Co-60 IN KWU PWRs
E	121	STUDIES OF THE EFFECT OF IMPROVEMENTS IN PRIMARY COOLANT PURIFICATION ON RADIOACTIVITY IN PWR COOLANT
E	122	KRAFTWERK UNION'S OCCUPATIONAL EXPOSURE REDUCTION PROGRAM
E	123	MODELING THE TRANSPORT OF CORROSION PRODUCT IN BWR PRIMARY SYSTEM
E	124	STUDIES OF BWR COBALT DEPOSITION
E	125	GENERAL ELECTRIC'S PORGRAM OF ZINC INJECTION PASSIVATION
E	126	DEVELOPMENT OF TECHNIQUES FOR REPLACING CRACKED TUBE SECTIONS OF STEAM GENERATOR'S WITH CORROSION-RESISTANT TUBING
E	127	DEVELOPMENT OF AUTOMATED TECHNIQUES FOR WELDING STEAM GENERATOR TUBES
E	128	GUIDELINES FOR PWR SECONDARY WATER CHEMISTRY
E	129	EXAMINATION OF MILLSTONE-2 TUBES
E	130	TUBE FRETTING AND FATIGUE IN STEAM GENERATORS
E	131	SUPPLEMENTARY EXAMINATION AND ALTERNATIVE MATERIALS OF MODEL STEAM GENERATOR INTERNALS
E	132	ROTO-PEENING OF TUBES TO PROLONG THE LIFE OF STEAM GENERATORS
E	133	STUDIES OF PWR WATER CHEMISTRY LOOP
E	134	FIELD TESTS FOR RADIATION CONTROL
E	135	ASSESSMENT OF RADIATION FIELD AND DOSE DATA
E	136	DEMONSTRATION OF ELECTROPOLISHING
E	137	VALIDATION OF DECONTAMINATION CORROSION
E	138	CHEMISTRY OF CRUD TRANSPORT
E	139	A REMOTE MAINTENANCE ROBOT FOR A PULSED NUCLEAR REACTOR

E/T	ID	TITLE
E	140	DEMONSTRATION OF PWR RADIATION CONTROL
E	141	DEVELOPMENT OF POST-ACCIDENT CHEMICAL DECONTAMINATION METHOD
E	142	REACTOR CHANNEL REHABILITATION AT CANDU REACTORS
E	143	BREAKDOWN HANDLING VIA ROBOTICS
E	144	IMPROVEMENT OF FUEL HANDLING SYSTEM EQUIPMENT FOR CANDU REACTORS
E	145	REPLACEMENT FOR COBALT BALLS IN CANDU FUEL HANDLING SYSTEMS
E	146	FUELING MACHINE FLOW INJECTION FOR CANDU REACTORS
E	147	PWR pH/RADIATION CONTROL PLANT TEST
E	148	RELEASE RATES OF COBALT CORROSION PRODUCTS
E	149	COBALT REPLACEMENT AND RADIATION FIELD MEASUREMENTS IN CANDU REACTORS
E	150	PASSIVATION OF SPECIMENS FOR VNC TEST FACILITY
E	151	PASSIVATION/PRECONDITIONING OF STAINLESS STEEL: CHEMICAL METHODS
E	152	PASSIVATION/PRECONDITIONING OF STAINLESS STEEL: OXIDIZING TREATMENTS
E	153	PASSIVATION/PRECONDITIONING OF STAINLESS STEEL: STEEL ZIRCONIDING
E	154	EVALUATION OF PLANT DECONTAMINATION
E	155	DECONTAMINATION WASTE MANAGEMENT
E	156	FIELD TEST FOR FUEL DECONTAMINATION
E	157	DECONTAMINATION OF BWR PRIMARY SYSTEM
E	158	EVALUATION OF DECONTAMINATION REAGENT CORROSION
E	159	DECOMMISSIONING OF THE PROTOTYPE CANDU-BLW 250 MW(e) GENTILLY #1
E	160	DRY CANISTER STORAGE FOR IRRADIATED FUEL: DECOMMISSIONING OF GENTILLY #1 CANDU-BLW 250 MW(e)

E/T	ID	TITLE
E	161	DEVELOPMENT OF SPECIAL EQUIPMENT: DECOMMISSIONING OF GENTILLY #1 CANDU-BLW
E	162	DEVELOPMENT OF ULTRASONIC TESTING TECHNIQUES ON AUSTENITIC WELDS FOR IN-SERVICE INSPECTION
E	163	SURFACE MODIFICATION OF PWR STEAM GENERATOR CHANNEL HEADS
E	164	SLUDGE REMOVAL IN PWR STEAM GENERATORS
E	165	ROTATING EDDY CURRENT AND ULTRASONIC PROBES
E	166	PRE-OPERATIONAL CHEMICAL CLEANING
E	167	DETECTION AND SIZING OF IGSCC IN BWR PIPING
E	168	STUDIES OF ULTRASONIC ENERGY PROPAGATION
E	169	IN-SERVICE INSPECTION OF ADVANCED STEAM GENERATOR
E	170	INSPECTION TECHNOLOGY FOR NOZZLES AND PIPES
E	171	DETECTION OF NEAR-SURFACE CRACKS
E	172	EPRI'S CENTER NONDESTRUCTIVE EVALUATION
E	173	ANALYSIS OF IN-SERVICE INSPECTION DATA
E	174	OPERATION OF BWR HYDROGEN WATER CHEMISTRY PLANT
E	175	STEAM GENERATOR REPLACEMENT PROGRAM AT RINGHALS #2
E	176	REPLACEMENT PROGRAM FOR THE FRENCH STEAM GENERATOR
E	177	THE WESTINGHOUSE/MITSUBISHI ADVANCED PWR DESIGN
E	178	COMBUSTION ENGINEERING'S PROGRAM FOR MONITORING EPRI-PWR STANDARD RADIATION
E	179	DECONTAMINATION DEVELOPMENT
E	180	ENGINEERING IMPACTS AND INNOVATIONS: THE CLEANUP OF TMI-2
E	181	RADIATION MAPPING AND ALARA PLANNING SYSTEM-RADMAPS
E	182	MICROSHIELD-A MICROCOMPUTER SHIELDING PROGRAM
E	183	THE OAK RIDGE ADVANCED SERVOMANIPULATOR (ASM)
E	184	THE GEC ADVANCED SLAVE MANIPULATOR

E/T	ID	TITLE
E	185	ROBOT-ASSISTED REPAIR OF CHINON-A3 GAS-COOLED REACTOR BY ISIS
E	186	HYDROGEN WATER CHEMISTRY FOR BWRS-STATUS OF THE U.S. PROGRAM
E	187	TECHNIQUES FOR DETERMINING THE RADIOLOGICAL IMPACTS OF HWC
E	188	EVALUATION PROGRAM FOR BWR FUEL ROD PERFORMANCE
E	189	PSE&G ROBOT TASK FORCE
E	190	DEVELOPMENT OF DECONTAMINATION TECHNOLOGY FOR PWRs AND BWRs
E	191	REMEDIAL METHODS FOR INTERGRANULAR ATTACK OF ALLOY 600 TUBING
E	192	SAMPLING STEAM GENERATOR TUBE
E	193	EVALUATION OF THE TOUGHNESS OF AUSTENITIC STAINLESS STEEL PIPE WELDMENTS
E	194	STRESS CORROSION CRACKING OF PWR STEAM GENERATOR TUBING
E	195	TRANSPORT OF THE FISSION PRODUCT
E	196	THE ADVANCED BOILING WATER REACTOR
E	197	DEGRADATION STUDIES ON STEAM GENERATOR TUBE
E	198	ADVANCED RECIRCULATING STEAM GENERATOR FOR CANDUS AND LARGE PWRs
E	199	DEVELOP AND IMPLEMENT A SHOT-PEENING PROCESS TO REDUCE STRESS CORROSION CRACKING IN STEAM GENERATOR TUBES
E	200	ELECTROMAGNETIC FILTERS FOR NUCLEAR PLANTS
E	201	WORKSHOP ON THERMALLY TREATED ALLOY 690 TUBES FOR NUCLEAR STEAM GENERATORS
E	202	LABORATORY EXAMINATIONS OF SELECTED TUBES FROM TEST FACILITIES OF THE STEAM GENERATOR OWNERS GROUP
E	203	TESTING OF CREVICE HIDEOUT RETURN
E	204	REMOVAL OF A TUBESHEET SAMPLE FROM A RETIRED POINT BEACH UNIT 1 STEAM GENERATOR

E/T	ID	TITLE
E	205	EXAMINATION OF CALVERT CLIFFS UNIT 1 TUBE
E	206	EVALUATION OF INTERGRANULAR ATTACK ON ALLOY 600
E	207	DEVELOPMENT OF COBALT-FREE HARD-FACING ALLOYS FOR NUCLEAR APPLICATIONS
E	208	CORROSION-PRODUCT RELEASE IN LWRs
E	209	ULTRASONIC INSPECTION OF RADIOACTIVE FUEL
E	210	PROPERTIES OF COLLOIDAL CORROSION PRODUCTS AND THEIR EFFECTS ON NUCLEAR PLANTS
E	211	STRESS CORROSION CRACKING TEST OF EXPANDED STEAM GENERATOR TUBES
E	212	PWR RADIATION FIELDS AT COMBUSTION ENGINEERING PLANTS THROUGH MID-1985
E	213	INVESTIGATION OF ADVANCED ACOUSTIC AND OPTICAL NONDESTRUCTIVE EVALUATION TECHNIQUES
E	214	IMPROVEMENT OF A PORTABLE HIGH-ENERGY RADIOGRAPHIC INSPECTION SYSTEM
E	215	STUDIES OF BWR COBALT DEPOSITION
E	216	MATHEMATICAL MODELING OF ELECTROCHEMICAL CONDITIONS WITHIN A STRESS CORROSION CRACK
E	217	COMPARISON OF FACTORS IMPACTING ON RADIATION BUILD-UP AT TWO BWRs
E	218	TELEMAC VAMPIRE ROBOT
E	219	THE MF3 ROBOT
E	220	THE MF4 ROBOT
E	221	THE K2A ROBOT
E	222	THE GCA-PAR-1 ROBOTICS DEVICE
E	223	THE MF2 ROBOT
E	224	THE REMOTE INSPECTION SYSTEMS (RIS) DEVICE
E	225	EFFECT OF STEAM GENERATOR TUBE TEMPERATURE ON THE STRESS CORROSION CRACKING OF ALLOY 600
E	226	STRATEGY FOR STEAM GENERATOR REPLACEMENT

E/T	ID	TITLE
E	227	BORIC ACID RECLAMATION BY MEMBRANE TECHNOLOGY
E	228	MEASURES TO IMPROVE THE PERFORMANCE OF STEAM GENERATOR TUBES AT EDF's 900 MWE PWRs
E	229	VERALIGHT - A NEW LIGHT MANIPULATOR FOR STEAM GENERATOR INSPECTION
E	230	IN-PLANT APPLICATIONS OF NEW EDDY CURRENT INSPECTION TECHNIQUES
E	231	OPTIMIZATION PROGRAM FOR INDUCTION HEATING STRESS IMPROVEMENT (IHSI)
E	232	STUDIES ON THE FEASIBILITY OF FULL-SYSTEM DECONTAMINATION
E	233	CHEMICAL DECONTAMINATION OF BWR FUEL AND CORE MATERIALS
E	234	WESTINGHOUSE QUALIFICATION PROGRAM FOR FULL-SYSTEM DECONTAMINATION
E	235	INVESTIGATION OF THE EFFECT OF LITHIUM HYDROXIDE CONCENTRATIONS ON PRIMARY WATER STRESS CORROSION CRACKING IN PWR PLANTS
E	236	USE OF ENRICHED BORIC ACID IN PRESSURIZED WATER REACTOR PLANTS
E	237	THE OZONE DECONTAMINATION PROCESS (ODP)
E	238	EVALUATION OF THE STUDSVIK/ASEA-ATOM OZONE DECONTAMINATION PROCESS (ODP)
E	239	DEVELOPMENTS IN DECONTAMINATION WASTE MANAGEMENT
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11. ENGINEERING RESEARCH PROJECTS

BNL ALARA Center Data Base

Country: U.S.A. **ID**: E-193

EVALUATION OF THE TOUGHNESS OF AUSTENITIC STAINLESS STEEL PIPE WELDMENTS

Keywords: COMPONENT RELIABILITY; AUSTENITIC STAINLESS STEEL; CRACKING; WELDS; PIPE FRACTURES

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Project Manager:

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Objectives: To evaluate the elastic-plastic fracture properties of submerged-arc and shielded metal-arc weldments made with austenitic stainless steel weldments, in both as-deposited and solution annealed conditions, at LWR operating temperatures.

Comments: The study involved a microstructural evaluation of four types of weldments. Tests included fracture behavior and tensile properties. It was concluded solution annealing does not change toughness of weldment.

Potential for dose limitation:

References: EPRI Report NP-4668, June 1986, available from Research Report Center, Box 50490, Palo Alto, CA 94303.

Duration: **from**: 1984 **to**: 1988 **Funding**: N/A

Status: In progress. **Last Update**: February 20, 1987

BNL ALARA Center Data Base

Country: U.S.A. **ID**: E-194

STRESS CORROSION CRACKING OF PWR STEAM GENERATOR TUBING

Keywords: COMPONENT RELIABILITY; STEAM GENERATORS; STRESS CORROSION CRACKING; INCONEL-600; PWR

Principal Investigator:

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Project Manager:

Nuclear Regulatory Commission
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Phone:

Objectives: To develop quantitative data as a basis for determining the useful life of in-service Alloy 600 tubing from accelerated test data.

Comments: Two conditions being studied are: 1) residual stress where deformation is no longer active, and 2) where deformation continues. "Known" steam generator tubes will be obtained for tests to verify the model.

Potential for dose limitation: Through potential use for predicting SG tubing service performance. Plugging criteria and maintenance will also be affected.

References: "Safety Research Programs Sponsored by the Office of Nuclear Regulatory Research," NUREG/CR-2331, December 1985.

Duration: **from**: 1978 **to**: 1988 **Funding**: 2 man-years

Status: In progress. **Last Update**: September 11, 1986

Country: CANADA **ID:** E-195

FISSION PRODUCT TRANSPORT

Keywords: CONTAMINATION PREVENTION; FISSION PRODUCT TRANSPORT; DEFECTIVE FUEL; SHUTDOWN FIELDS; CANDU; PWR

Principal Investigator:

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Project Manager:

COG-CANDEV

Phone:

Objectives: To quantitatively determine the mechanism of fission product transport and deposition in the primary system of water-cooled reactors.

Comments:

Potential for dose limitation: Using gaseous and dissolved fission product data, it is possible to diagnose incipient unstable defects in fuel and have these removed before significant contamination of the primary heat transport system occurs.

References: AECL Report AECL-8705 (available from CRNL, Chalk River, ONT, Canada KOJ 1JO).

Duration: **from**: 1985 **to**: 1990

Funding: CN$ 1 Million

Status: In progress.

Last Update: September 11, 1986

BNL ALARA Center Data Base

Country: U.S.A. **ID:** E-196

THE ADVANCED BOILING WATER REACTOR

Keywords: COMPONENT RELIABILITY; BWR; PUMPS; CONTAINMENT; CONTROL ROD DRIVE; FUEL

Principal Investigator:

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Objectives: (1) Enhance plant operability, (2) Increase plant safety, (3) Improve plant availability, (4) reduce plant operator's radiation exposure, (5) Reduce plant capital and operating costs.

Comments: Significant features include: (1)internal recirculation pumps, (2) fine motion CR drives, (3) improved reinforced concrete containment vessel integrated with building, (4) improved core and fuel, (5) optimized safety systems, and (6) advanced controls and instrumentation systems.

Potential for dose limitation: The ABWR occupational radiation exposure has been estimated to be 40-man-rem/year. This value was estimated based on the latest technology to limit radiation buildup in the plant and was also based on the latest plant exposure data in Japan.

References: D.R. Wilkins, T.Seko, S.Sugino, H.Hashimoto, "Advanced BWR" Nuc. Eng. Intl., Vol. 31, No. 383 (June 1986), pp. 36-45.

Duration: **from**: 1978 **to**: 1988 **Funding**: N/A

Status: In progress. **Last Update:** September 24, 1986

BNL ALARA Center Data Base

Country: JAPAN **ID**: E-196B

THE ADVANCED BOILING WATER REACTOR

Keywords: COMPONENT RELIABILITY; BWR; PUMPS, CONTAINMENT, CONTROL ROD DRIVE, FUEL

Principal Investigator:

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Project Manager:

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Objectives: (1) Enhance plant operability, (2) increase plant safety, (3) improve plant availability, (4) reduce plant operator radiation exposure, (5) reduce plant capital and operating costs.

Comments: Significant features include: (1) internal recirculation pumps, (2) fine-motion CR drives, (3) improved reinforced concrete containment vessel integrated with building, (4) improved core and fuel, (5) optimized safety systems, and (6) advanced controls and instrumentation systems.

Potential for dose limitation: The ABWR occupational radiation exposure has been estimated to be 40 man-rem/year. This was estimated based on the latest technology to limit radiation buildup in the plant and was also based on the latest plant exposure in Japan.

References: D.R. Wilkins, T. Seko, S. Sugino, H. Hashimoto, "Advanced BWR" Nuc. Eng. Intl., Vol. 31, No. 383 (June 1986), pp. 36-45.

Duration: **from**: 1978 **to**: 1988 **Funding**: N/A

Status: In progress. **Last Update:** September 24, 1986

BNL ALARA Center Data Base

Country: CANADA **ID**: E-197

STUDIES OF STEAM GENERATOR TUBE DEGRADATION

Keywords: COMPONENT RELIABILITY; STEAM GENERATORS; STRESS CORROSION CRACKING (SCC); INTER-GRANULAR ATTACK (IGA); DENTING; PITTING; TUBE DEGRADATION; PWR

Principal Investigator:

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Project Manager:

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Objectives: A survey of defects in steam generator tubes is being carried out, using data from utilities. Analysis of the results will provide information on the reasons for tube degradation.

Comments: 84 tubes were plugged at 63 reactors (43% of survey). The main cause of failure was stress corrosion cracking or intergranular attack. Previously pitting, denting, and phosphate wastage were the primary causes. Remedies, e.g., titanium tubing, were evaluated.

Potential for dose limitation:

References: "Worldwide SG Tube Performance: Analysis of 1983-84 Statistics," Nucl. Eng. Intl. Vol. 31, (June 1986), pp. 81-83.

Duration: **from**: 1971 **to**: 1988 **Funding**: N/A

Status: In progress. **Last Update:** September 24, 1987

Country: CANADA **ID**: E-198

ADVANCED RECIRCULATING STEAM GENERATOR FOR CANDUS AND LARGE PWRS

Keywords: COMPONENT RELIABILITY; STEAM GENERATORS; CANDUS; PWRS; ALLOY 600; ALLOY 800; TUBE TREATMENT

Principal Investigator:

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Objectives: To further develop the design of the B&W Advanced Steam Generator.

Comments: After a state-of-the-art review of Steam Generator (SG) designs and operating experience, an advanced SG design was developed for CANDU 600, CANDU 950, and PWRs. Previous experience with CANDU SGs has been extremely good. Outages related to SG failure for CANDUs are 0.01% compared to 25% for PWRs. Dose reduction features, such as manway manipulators which permit rapid handling, are part of the design.

Potential for dose limitation:

References: Smith, J.C. and Akeroyd, J.K., Nucl. Eng. Intl., Vol. 31 (June 86), 83.

Duration: **from**: 1984 **to**: 1988 **Funding**: N/A

Status: In progress. **Last Update:** September 25, 1987

Country: U.S.A. **ID**: E-199

DEVELOP AND IMPLEMENT A SHOT-PEENING PROCESS TO REDUCE STRESS CORROSION CRACKING IN STEAM GENERATOR TUBES

Keywords: COMPONENT RELIABILITY; STEAM GENERATOR; SHOT PEENING; STRESS CORROSION CRACKING (SCC); PWSCC; IGSCC; PWR

Principal Investigator:

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Objectives: Develop and implement a shot-peening process that reduces the likelihood of primary side stress corrosion cracking within the tubes at the tubesheet roll transition region of steam generators.

Comments: This process is being developed and implemented in cooperation with the Metal Improvement Company.

Potential for dose limitation: Should reduce long-term radiation exposure by improving the reliability of steam generators.

References:

Duration: **from**: 1985 **to**: 1987

Funding: N/A

Status: 18 channel heads completed

Last Update: September 26, 1986

BNL ALARA Center Data Base

Country: U.S.A. **ID:** E-200

ELECTROMAGNETIC FILTERS FOR NUCLEAR PLANTS

Keywords: COMPONENT RELIABILITY; ELECTROMAGNETIC FILTERS; FEEDWATER; CONDENSATE; PWR; CLEANUP

Principal Investigator:

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Project Manager:

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Objectives: To optimize electromagnetic filters that remove magnetic and non-magnetic materials from feedwater and condensate systems at nuclear power stations.

Comments: The B&W filter is designed for low power consumption. It offers permanent filter media, efficient backflush cleaning, and a high capacity for particulate loading.

Potential for dose limitation: These filters will not affect the radiation levels in the primary system. However, they should reduce radiation exposure by improving system reliability.

References: "High Temperature Filtration of PWR Feedwater," Report NWT-228, February 1985.

Duration: **from:** 1984 **to:** 1986 **Funding:** N/A

Status: Results published. **Last Update:** September 30, 1986

BNL ALARA Center Data Base

Country: U.S.A. **ID**: E-201

WORKSHOP ON THERMALLY TREATED ALLOY 690 TUBES FOR NUCLEAR STEAM GENERATORS

Keywords: COMPONENT RELIABILITY; STEAM GENERATORS; INCONEL ALLOYS; TUBES; CORROSION; PWR'S

Principal Investigator:

Project Manager:

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Objectives: To collect available metallurgical and corrosion information on TT alloy 690 and compare it with data on other tube alloys.

Comments: Data presented at this workshop confirmed the superior resistance to corrosion of thermally treated alloy 690. Pending further tests and optimization procedures, this material appears to be the best choice for manufacturers of steam generator tubes.

Potential for dose limitation: The increased reliability of the steam generators, due to the adoption of thermally treated alloy 690 tubes, should substantially reduce the requirement for steam generator related work at PWRs.

References: EPRI NP-4665M-SR, July 1986, 40 pages. EPRI NP-4665S-SR, July 1986, 572 pages.

Duration: **from**: 1985 **to**: 1986 **Funding**: N/A

Status: The work of the workshop has been completed. **Last Update**: February 12, 1987

Country: U.S.A. **ID:** E-202

LABORATORY EXAMINATIONS OF SELECTED TUBES FROM TEST FACILITIES OF THE STEAM GENERATOR OWNERS GROUP

Keywords: COMPONENT RELIABILITY; STEAM GENERATORS; CREVICE CORROSION; INCONEL ALLOYS; TUBES; PITTING; PWR'S

Principal Investigator:

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Objectives: As part of its multiphased investigation into model steam generators, the Steam Generator Owners Group destructively examined tubes exposed to aggressive chemical environments (EPRI reports NP-3044 and NP-3275). Previous examinations had provided preliminary data on tube corrosion and denting. Additional examinations were necessary to investigate tube degradation.

* To determine whether denting in model boilers contributed to corrosion damage of alloy 600 tubes.
* To determine why a TT alloy 600 tube cracked in a sulfate environment during a model steam generator test.

Comments: In tests with single-tube model boilers, short-term exposure to a mild denting environment caused only slight corrosion of alloy 600 tubes. In a year-long test of a model steam generator, a thermally treated tube exposed to a sulfate environment and elongation developed circumferential stress corrosion cracking. Microstructural examinations found inadequate carbides at grain boundaries.

Potential for dose limitation:

References: EPRI NP-4625, 1986, 284 pages.

Duration: **from:** 1984 **to:** 1987

Funding: N/A

Status: Completed

Last Update: February 12, 1987

BNL ALARA Center Data Base

Country: U.S.A. **ID**: E-203

TESTING OF CREVICE HIDEOUT RETURN

Keywords: COMPONENT RELIABILITY; STEAM GENERATORS; CREVICE CORROSION; RESIDUES; PWR

Principal Investigator:

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Objectives: To evaluate procedures for removing contaminants from crevices in support plates to minimize corrosion.

Comments: Shutdowns represent a convenient procedure for removing hideout salts from crevices in slightly dented support plates in steam generators. The tests demonstrate that on-line additions of boric acid and calcium hydroxide--besides their neutralizing and inhibitory functions--increase the return of sodium from crevices upon shutdown.

Potential for dose limitation:

References: EPRI NP-4678, July 1986, 188 pages.

Duration: **from**: 1984 **to**: 1986 **Funding**:N/A

Status: Completed. **Last Update**: February 12, 1987

Country: U.S.A. **ID:** E-204

REMOVAL OF A TUBESHEET SAMPLE FROM A RETIRED POINT BEACH UNIT 1 STEAM GENERATOR

Keywords: COMPONENT RELIABILITY; STEAM GENERATORS; INCONEL ALLOYS; TUBES; CREVICE CORROSION; PWR

Principal Investigator:

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Objectives: To obtain a sample of the tubesheet from a retired nuclear plant steam generator.

Comments: Intergranular attack (IGA) of steam generator tubes forced Wisconsin Electric Power Company to plug a large number of tubes and to eventually replace the steam generators at its Point Beach unit 1 plant. The retired generators provided an opportunity to confirm laboratory data indicating that certain chemical environments cause IGA in the crevices in the steam generator tubesheet. Engineers used electric discharge machining to remove a tubesheet sample containing 15 undisturbed tubes from a nuclear steam generator retired from service because of corrosion damage. In follow-up work, several laboratories will analyze the sample to determine the cause of intergranular attack on the tubes in the tubesheet.

Potential for dose limitation:

References: EPRI NP-4901, 1986, 144 pages.

Duration: **from**: 1984 **to**: 1987 **Funding**:N/A

Status: In progress **Last Update**: February 12, 1987

BNL ALARA Center Data Base

Country: U.S.A. **ID**: E-205

EXAMINATION OF CALVERT CLIFFS UNIT 1 TUBE

Keywords: COMPONENT RELIABILITY; STEAM GENERATORS; INCONEL ALLOYS; PITTING; DEPOSITS

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Objectives: To determine--by nondestructive and destructive examination of tube segments from Calvert Cliffs Unit 1 steam generators--the causes and the forms of tube damage that had been indicated by the in-service inspection tests.

Comments: Annual in-service inspection of a Calvert Cliffs steam generator possible tube degradation, prompting an investigation of root causes and extent of damage. Nondestructive and destructive examinations of the tubing revealed that surface deposits and shallow wall thinning, rather than pitting, were the sources of the degradation.

Potential for dose limitation: This project should lead to improvements in the design of steam generators. Better interpretation of in-service inspection results is also likely. The improved reliability and reduced inspection requirements is likely to have a favorable impact on radiation dose.

References: EPRI Report NP-4904, 1986, 92 pages

Duration: from: 1984 to: 1987 **Funding**: N/A

Status: Completed. **Last Update**: January 30, 1987

Country: U.S.A. **ID**: E-206

EVALUATION OF INTERGRANULAR ATTACK ON ALLOY 600

Keywords: COMPONENT RELIABILITY; STEAM GENERATORS; INCONEL ALLOYS; IGSCC; CORROSION TESTING

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Objectives:

* To assess the effects of alkaline species on IGA of alloy 600 tubes in tubesheet and tube support plate crevices.

* To select a reference environment for the evaluation of remedial measures.

Comments: Exposure to several common contaminants of the steam generator in solution produced intergranular attack in the crevices of model boiler tubesheets and tube support plates. The results identified a sodium hydroxide-sodium sulfate solution as the most realistic environment for testing the effectiveness of boric acid as a remedy.

Potential for dose limitation: Leaks resulting from intergranular attack (IGA) on alloy 600 tubes in nuclear steam generators have required costly repairs and, in some cases, the replacement of steam generators. This study is part of a larger EPRI-Steam Generator Owners Group (SGOG) II program to understand the causes of IGA and to develop on-site preventive measures (EPRI reports NP-4458 and NP-4478). The project is likely to lead to increased reliability of steam generators and hence reduce radiation dose due to reduced maintenance requirements.

References: EPRI Report NP-4978, 1986, 212 pages.

Duration: **from**: 1984 **to**: 1987 **Funding**: N/A

Status: In progress. **Last Update**: January 30, 1987

Country: U.S.A. **ID**: E-207

DEVELOPMENT OF COBALT-FREE HARD-FACING ALLOYS FOR NUCLEAR APPLICATIONS

Keywords: CONTAMINATION PREVENTION; WEAR; HARD-FACING ALLOYS; ALLOY DEVELOPMENT; MECHANICAL PROPERTIES

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Objectives: Earlier EPRI studies (reports NP-3444 and NP-3888) showed that, over time, wear and corrosion of cobalt-based hard facings used on nuclear reactor components release significant amounts of cobalt into primary water circuits. The objectives of the project are to develop cobalt-free hard-facing alloys for use on nuclear components, identifying those with adequate resistance to wear and corrosion and with acceptable weldability and mechanical properties.

Comments: The present study showed that several iron-based alloys have the weldability, mechanical proprieties, and wear resistance to replace the cobalt-based alloys used as hard facings in both nuclear and nonnuclear industries. In future (1987), the best alloys will be deposited on valves and subjected to loop tests.

Potential for dose limitation: Cobalt-based alloys used for hard-facing reactor valves are a major source of radiation fields in nuclear power plants. The new, iron-based alloys exhibit the mechanical properties, weldability, and wear resistance needed for service in nuclear power plants. Used as replacement hard facings, these alloys could reduce radiation-field buildup on out-of-core components.

References: EPRI Report NP-4775, 1986, 96 pages.

Duration: **from**: 1984 **to**: 1988

Funding: N/A

Status: In progress.

Last Update: January 30, 1987

BNL ALARA Center Data Base

Country: U.S.A. **ID**: E-208

RELEASE OF CORROSION-PRODUCT IN LWRS

Keywords: CONTAMINATION PREVENTION; CORROSION; RADIATION SOURCES; REACTOR MATERIALS; RADIATION BUILDUP

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Objectives:

* To investigate release of corrosion-product from Inconel 600 and Type 304 stainless steel in coolant containing lithium hydroxide and boric acid.
* To characterize corrosion films that form after exposure to coolant either saturated or unsaturated with corrosion products.
* To develop a model to relate the kinetics of film formation to the release of corrosion product.

Comments: Corrosion products released from construction materials are a major source of radiation buildup in LWRs. Recent findings in this three-year study show releases from Inconel and stainless steel in lithiated-borated coolants to be similar to those measured earlier under lithiated conditions. The results of corrosion-film characterization studies suggest a method of reducing radiation fields in new PWRs.

Potential for dose limitation: Accurate information about the release of cobalt due to the corrosion and wear of structural alloys will help nuclear power plant designers identify materials in which the cobalt content must be reduced to minimize radiation-field buildup. The release data obtained from experiments with lithiated and borated coolant are consistent with earlier cobalt inventory calculations (EPRI reports NP-2681 and NP-2685), indicating that corrosion release from Inconel 600 steam generator tubing is the primary source of cobalt in PWRs. These results have motivated several utilities to specify low-cobalt (0.015%) steam generator tubing for replacement steam generators, rather than the 0.04% value found in most operating plants. This work complements studies of cobalt release caused by wear of nuclear components (EPRI reports N-2684, NP-3220, and NP-3444).

References: EPRI Report NP-4741, 1986, 64 pages.

Duration: **from**: 1984 **to**: 1988 **Funding**: N/A

Status: In progress. **Last Update**: January 30, 1987

BNL ALARA Center Data Base

Country: U.S.A. **ID**: E-209

ULTRASONIC INSPECTION OF RADIOACTIVE FUEL

Keywords: REMOTE SYSTEMS; FUEL INSPECTION; FAILED FUEL; FUEL ROD; ULTRASONIC INSPECTION OF FUEL

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Project Manager:

Services of all U.S. Nuclear Power Utilities

Phone:

Objectives: The Brown Boveri Failed Fuel Rod Detection System (FFRDS) uses a unique ultrasonic probe operated by a remote underwater manipulator. Individual fuel rods are rapidly inspected, and defective rods positively identified without disassembling the fuel. Over 400,000 rods in more than 2,000 fuel assemblies (PWR and BWR) were successfully inspected by Brown Boveri by the end of 1986. Elimination of leaking fuel from operating reactor cores has reduced worker exposures.

Comments: Brown Boveri identified nearly 800 leaking fuel rods (about 0.2% of those inspected) in 41 different inspection campaigns in the U.S. and overseas. Of the 2,022 fuel assemblies inspected, nearly 20% were found to have at least one defective fuel rod. The accuracy and reliability of the FFRDS was demonstrated in direct comparison with time-consuming eddy current measurements (which require disassembly of the fuel), and agreement was seen for 99.95% of the more than 4,000 rods.

Potential for dose limitation: Significant reductions in worker exposures can result from routinely inspecting all partially burned fuel assemblies that are reinserted. Elimination of leaking fuel (up to 0.125% of the total fuel) has been estimated to reduce exposures by 540% (ref: NUREG/CR-4485, 1/86). Leaking fuel also increases handling requirements for solid and liquid radwaste, may require use of respirators, and increase long-term plant contamination due to tramp uranium. Routine FFRDS testing of discharged fuel will reduce exposures for handling, storage, and eventual disposal of spent fuel.

References: 1. ANS Meeting, Reno, Nevada, June 1986. 2. NUCLEAR PLANT SAFETY, Nov-Dec, 1986.

Duration: **from**: 1979 **to**: 1988

Funding: Inspection costs are <2% of residual fuel value of each assembly.

Status: In progress.

Last Update: February 5, 1987

BNL ALARA Center Data Base

Country: U.S.A. **ID:** E-210

PROPERTIES OF COLLOIDAL CORROSION PRODUCTS AND THEIR EFFECTS ON NUCLEAR PLANTS

Keywords: CONTAMINATION PREVENTION; LWR; COBALT-60; RADIOACTIVITY TRANSPORT; CORROSION; WATER CHEMISTRY

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Objectives: To develop laboratory methods of determining the role that each common oxide of the corrosion-product plays in cobalt-60 transport and sludge buildup in LWR coolant loops and to determine the chemical conditions under which each oxide dissolves.

Comments: Using laboratory-produced particulate oxides to simulate the surfaces of nuclear reactor piping, researchers showed that high pH tends to immobilize ionic and particulate forms of cobalt in LWR coolant. The study should also develop new methods of producing the particulates that aid in understanding the transport of corrosion-product.

Potential for dose limitation: The study will enhance the understanding of the transport of corrosion products, including cobalt-60, in the piping surfaces. It has already shown that sudden drops in pH will cause desorption of cobalt-60 from in-core surfaces and in its redistribution to out-of-core surfaces. This understanding will help to reduce radiation fields and thus decrease radiation exposure. A second, less important reason for exposure reduction will be because the fact that the project has developed laboratory methods of producing well-characterized corrosion particles in place of real and active piping surfaces.

References: EPRI Report NP-4817, December 1986, 72 pages.

Duration: **from:** 1986 **to:** 1987 **Funding:**N/A

Status: Completed. **Last Update:** March 4, 1987

BNL ALARA Center Data Base

Country: U.S.A. **ID**: E-211

STRESS CORROSION CRACKING TEST OF EXPANDED STEAM GENERATOR TUBES

Keywords: COMPONENT RELIABILITY; STEAM GENERATOR; STRESS CORROSION CRACKING; MAINTENANCE

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Objectives: To demonstrate that explosive expansion lessens the susceptibility of mechanically expanded tubes to stress corrosion cracking.

Comments: In tests on samples having representative mechanical rolling faults, the explosive reexpansion technique proved effective in increasing the crack resistance of steam generator tubing. In comparison to rerolling, the technique offers utilities faster and less expensive repairs.

Potential for dose limitation: This work demonstrates that the technique of explosive reexpansion improves the resistance of tubing to stress corrosion cracking caused by roller-expanded imperfections. Moreover, the fact that the technique can operate under remote control offers the important advantage of reducing personnel exposure in operating plants.

References: EPRI Report NP-5012, January 1987, 112 pages.

Duration: **from**: 1986 **to**: 1987

Funding: N/A

Status: In progress.

Last Update: March 4, 1987

BNL ALARA Center Data Base

Country: U.S.A. **ID**: E-212

PWR RADIATION FIELDS AT COMBUSTION ENGINEERING PLANTS THROUGH MID-1985

Keywords: CONTAMINATION PREVENTION; PWR; RADIATION MONITORING; RADIATION DOSE

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Objectives: To apply the standard radiation monitoring program started by EPRI to Combustion Engineering plants and to collect a first round of data.

Comments: Surveys have provided data on radiation fields at seven Combustion Engineering plants. Measurements at well-defined locations on steam generator and reactor coolant piping will aid in planning changes in PWR operation and design.

Potential for dose limitation: The following information obtained should be beneficial in reducing occupational exposure:

* Radiation fields on the hot leg side were higher than on the cold leg side, suggesting the influence of temperature differences.
* The highest dose rates inside the channel head were found to be at the tubesheet, suggesting decontamination or shielding effort should be concentrated there.
* Reactor coolant pH influences the rates of transport and deposition of corrosion products within the primary coolant system. The activation of this crud within the core and its subsequent transport and deposition throughout the reactor coolant system are largely responsible for out-of-core radiation fields.

References: EPRI Report NP-4998, January 1987, 108 pages.

Duration: **from**: 1986 **to**: 1987 **Funding**: N/A

Status: In progress. **Last Update**: March 4, 1987

BNL ALARA Center Data Base

Country: U.S.A. **ID**: E-213

ADVANCED ACOUSTIC AND OPTICAL TECHNIQUES FOR NONDESTRUCTIVE EVALUATION

Keywords: COMPONENT RELIABILITY; NONDESTRUCTIVE TESTING; ACOUSTIC IMAGING; HOLOGRAPHY; ULTRASONIC TOMOGRAPHY; IN-SERVICE INSPECTION

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Objectives: (1) To assess advanced acoustic and optical techniques for nondestructive evaluation of nuclear power plant components. (2) To produce a real-time holographic imaging system.

Comments: A holographic imaging system being developed in this study could increase the accuracy of nondestructive evaluations of nuclear plant components in the field. The systems ability to create optical or computational images of defects in metal components such as pipes, nozzles, and pressure vessels should greatly improve the characterization of flaws.

Potential for dose limitation: In the past, the major obstacle to using holography in the field was the length of time needed to gather the data and interpret the images. Recent technologic advancements removed most of those limitations, providing a fast, accurate approach to complex inspection tasks. At the same time, the potential for significant doses to the inspections personnel has been significantly reduced.

References: EPRI NP-4897, October 1986, 108 pages.

Duration: **from**: 1985 **to**: 1987

Funding: N/A

Status: In progress.

Last Update: March 30, 1987

Country: U.S.A. **ID**: E-214

IMPROVEMENT OF A PORTABLE HIGH-ENERGY RADIOGRAPHIC INSPECTION SYSTEM

Keywords: COMPONENT RELIABILITY; NONDESTRUCTIVE TESTING; CRACK DETECTION; CRACK

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Objectives: To expand the MINAC capability for radiographic nondestructive evaluations in the field through design improvements, an increased energy output, and a capability for motion radiography.

Comments: Using portable high-energy X-ray equipment, utilities can perform nondestructive evaluations of thick-section components in the field. The upgraded MINAC system, which operates over a range of 4 to 6 MeV, has a small new X-ray head that can be operated up to 20 ft from the radio-frequency power source without signal loss of output. Auxiliary radiographic equipment, including an image enhancement unit, permits real- time filmless inspection. Image enhancement proved to be independently valuable for reinterpreting current and archival film radiographs.

Potential for dose limitation: The small size and ease of handling of the new X-ray head which make it possible to aim it at areas of the pipe otherwise difficult to access, is just one of the reasons why this inspections device saves occupational exposure.

References: EPRI NP-4848, February 1987, 128 pages.

Duration: **from**: 1986 **to**: 1987 **Funding**: N/A

Status: In progress. **Last Update**: March 30, 1987

Country: U.S.A. **ID**: E-215

BWR COBALT DEPOSITION STUDIES

Keywords: COMPONENT RELIABILITY, RADIATION BUILDUP; BWR; WATER CHEMISTRY; COBALT; STAINLESS STEEL

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Objectives: To determine, under simulated BWR conditions, the effects of water chemistry and of the preconditioning of stainless steel surfaces on cobalt-60 deposition in primary coolant systems.

Comments: Buildup of radiation fields on BWR circulation piping is strongly influenced by water chemistry during initial power raising and subsequent operation. Laboratory tests show that improving water quality reduces cobalt-60 deposition in both normal water chemistry and hydrogen water chemistry.

Potential for dose limitation: Occupational radiation exposure in BWRs results primarily from the deposition of cobalt-60 on out-of-core piping surfaces. This project is part of a larger coordinated study involving plant measurements and laboratory loop experiments to identify operating conditions that minimize radiation buildup.

References: EPRI NP-4725, Interim Report, August 1986, 140 pages.

Duration: **from**: 1985 **to**: 1988

Funding: N/A

Status: In progress.

Last Update: March 30, 1987

BNL ALARA Center Data Base

Country: U.S.A. **ID**: E-216

MATHEMATICAL MODELING OF ELECTROCHEMICAL CONDITIONS WITHIN A STRESS CORROSION CRACK

Keywords: COMPONENT RELIABILITY; STRESS CORROSION CRACKING; CRACKING; CREVICE CORROSION; MATHEMATICAL MODELS

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Objectives: To develop mathematical models that describe the spatial and temporal variations of the concentration of chemical species and electrostatic potential within a stress corrosion crack.

Comments: Stress corrosion cracking and corrosion fatigue are responsible for the cracking and eventual failure of many power plant components each year. The study seeks an understanding of the mechanisms of cracking through mathematical models. Related EPRI-sponsored experimental studies use physical models of stress corrosion cracks in investigating details of those mechanisms.

Potential for dose limitation: A better understanding of the mechanisms which cause cracking of nuclear components will result in improvements in their reliability.This will reduce maintenance requirements and have a beneficial impact on occupational exposure.

References: EPRI RD-4877, November 1986, 88 pages.

Duration: **from**: 1985 **to**: 1987 **Funding**: N/A

Status: In progress. **Last Update**: March 30, 1987

Country: U.S.A. **ID**: E-217

COMPARISON OF FACTORS EFFECTING RADIATION BUILDUP AT TWO BWR'S

Keywords: CONTAMINATION PREVENTION; RADIATION BUILDUP; RADIATION TRANSPORT; WATER CHEMISTRY; REACTOR MATERIALS; BWR

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Objectives: To document differences in system design, operation, and chemistry at the Monticello and Vermont Yankee BWRs and to identify factors responsible for differences in radiation buildup trends at the two plants.

Comments: Although the Monticello and Vermont Yankee plants were similar in many respects there was a large difference in the radiation buildup in the primary system. While Monticello had one of the lowest radiation buildup levels, Vermont Yankee had one of the highest. There were some differences in the design, operation and chemistry areas that could have effected radiation buildup. The differences occurred in core power density, jet pump design, volume of radwaste recycle, amount of stellite-bearing materials in the feedwater system, and composition and morphology of corrosion films. For example, the corrosion film on the Vermont Yankee recirculation line decontamination flange differed from that at Monticello. Data also suggested that the coolant chemistry at Vermont Yankee was less oxidizing. Other reasons for the higher buildup at Vermont Yankee might have been the higher copper concentration which could have affected the coolant and possibly the corrosion film, and the higher organic input at Vermont Yankee, associated with increased radwaste recycle volume.

Potential for dose limitation: This detailed investigation has not yet identified a key factor causing differences in radiation buildup trends at the Vermont Yankee and Monticello BWRs although several items of significance have emerged. Further work is in progress.

References: EPRI NP-5103, Interim Report, March 1987.

Duration: **from**: 1986 **to**: 1988 **Funding**: N/A

Status: In progress. **Last Update**: June 17, 1987

Country: BELGIUM **ID:** E-218

TELEMAC VAMPIRE ROBOT

Keywords: REMOTE SYSTEMS; ROBOTICS; OPERATIONS; SURVEILLANCE; TELEOPERATED DEVICES; AUTONOMOUSLY OPERATED DEVICES

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Objectives: Vampire is designed for operational and surveillance functions in nuclear power plants.

Comments: The tetherless, aluminum and stainless steel vehicle has 6 tracks for locomotion. Four of the tracks are articulated to allow for a variable geometry and greater stability in stair climbing. The vehicle can surmount 18-inch-high obstacles. It is designed to operate for 2.5 hours using two batteries. The device can be teleoperated or made to follow a prelaid path autonomously. The microwave communication link was tested inside reactor buildings. Its two manipulator arms can extend 82 inches vertically and 56 inches horizontally and each can lift 88 lbs. Vampire can be mounted with video cameras, radiation source monitors, temperature, pressure, and humidity sensors as well as vibration monitors and fault-indicating devices.

Potential for dose limitation:

References:

Duration: **from:** 1984 **to:** 1987 **Funding:** N/A

Status: In progress. **Last Update:** September 4, 1987

Country: F.R.GERMANY **ID:** E-219

THE MF3 ROBOT

Keywords: REMOTE SYSTEMS; ROBOTICS; OPERATIONS; EXPLOSIVES HANDLING; TELEOPERATED DEVICES;

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Project Manager:

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Objectives: To develop a flexible robotic vehicle capable of operating in hazardous environments and carrying out an assortment of tasks.

Comments: The tethered, four-track vehicle weighs 900 lbs. It can operate for unlimited time. It is normally equipped with one manipulator arm but an optional second arm may be installed. The vehicle can climb stairs, move on a 45-degree slope, surmount 24-inch-high obstacles and traverse 3-ft.-wide crevices. The teleoperator-controlled device can be fitted with a video camera and headlights. It has an on-board power socket for power tools.

Potential for dose limitation:

References:

Duration: **from**: 1985 **to**: 1988 **Funding**: N/A

Status: In progress. **Last Update**: September 4, 1987

BNL ALARA Center Data Base

Country: F.R. GERMANY **ID:** E-220

THE MF4 ROBOT

Keywords: REMOTE SYSTEMS; ROBOTICS; OPERATIONS; EXPLOSIVES HANDLING; TELEOPERATED DEVICES;

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Objectives: To develop a rugged device capable of carrying out several operational tasks in the environment of a nuclear power plant.

Comments: The two-tracked, tethered vehicle weighs 364 lbs., and can operate for 1 to 2 hours. It is equipped with one manipulator arm which can lift 44 lbs. The robot vehicle can climb stairs, move on a 32-degree slope, surmount small obstacles, and turn on a 58-inch radius. The device is teleoperator-controlled and may be fitted with a video camera, spotlight, force-measuring device for gripper arm, sound detector, and a directional microphone. It has an on board power socket for power tools.

Potential for dose limitation:

References:

Duration: **from**: 1986 **to**: 1988

Funding: N/A

Status: In progress.

Last Update: September 4, 1987

BNL ALARA Center Data Base

Country: U.S.A. **ID**: E-221

THE K2A ROBOT

Keywords: REMOTE SYSTEMS; ROBOTICS; OPERATIONS; SURVEILLANCE; AUTOMATION; AUTONOMOUSLY OPERATED DEVICES; TELEOPERATED DEVICES

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Objectives: To develop a robotic device suitable for operations and surveillance in hazardous areas of nuclear power plants and conventional factories.

Comments: The 3-wheeled omnidirectional device weighs 265 lbs. It has a cast aluminum chassis, hardened steel drive shafts and an aluminum skin. It has a zero turning radius, 0-120 degree/second turn-rate, and can climb a 20-degree slope. It can be operated either tethered or tetherless for up to 8 hours autonomously or by teleoperation. Moving at 180 ft/min, the device may be operated in either manual or automatic modes. The navigation programs can contain up to 255 instructions. The program destinations can be given in cartesian coordinates entered manually, or by driving the vehicle in the "manual/teach" mode. The color graphic console computer displays real time data from the vehicle. A map display is provided to track the vehicle. Cybermation is supplying equipment for projects at Dupont Savannah River, Oak Ridge National Laboratories/University of Michigan, and Sandia National Laboratories.

Potential for dose limitation:

References: Savannah River, Joe Byrd; Oak Ridge, Rasey Fezzell.

Duration: from: 1983 to: 1987 **Funding**: Internal.

Status: Demonstration systems operational. **Last Update**: September 30, 1987

BNL ALARA Center Data Base

Country: U.S.A. **ID**: E-222

THE GCA-PAR-1 ROBOTICS DEVICE

Keywords: REMOTE SYSTEMS; ROBOTICS; OPERATIONS; TELEOPERATED DEVICES

Principal Investigator:

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Project Manager:

Phone:

Objectives: To design a vehicle capable of remote operations in nuclear power plants.
Comments: The tethered manipulator vehicle has two tracks and teleoperator control via cable. It is equipped with a video camera and lights. It has a heavy-duty manipulator arm on a mast with a total reach of 12 feet. Other options are available. It can be operated continuously for unlimited periods.

Potential for dose limitation:

References: For references contact Pat Canada, Martin Marrietta Energy Systemsm Oak Ridge National Laboratory, Oak Ridge, Tenn.

Duration: **from**: 1985 **to**: 1989

Funding: Corporate funds and personnel used for design.

Status: Eleven units are in operation on a periodic basis.

Last Update: January 4, 1988

Country: F.R.GERMANY **ID**: E-223

THE MF2 ROBOT

Keywords: REMOTE SYSTEMS; ROBOTICS; OPERATIONS; TELEOPERATED DEVICES; ALL TERRAIN VEHICLE

Principal Investigator:

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Project Manager:

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Phone:

Objectives: To develop a vehicle for remotely controlled operations in nuclear power plants.

Comments: The tetherless, 2 track vehicle is teleoperator controlled via RF. Its manipulator arm can lift up to 880 lbs. The device can be equipped with parallel jaw, grip hook, pipe tong, and other manipulator end-effectors.

The all terrain vehicle is equipped with 4 search lights and 3 video cameras, 2 of which are used for 3-D stereo viewing. It also has as standard equipment 2 stereo microphones, radiation detectors, temperature sensors, and a host of other devices. Its pan/tilt mechanism for all sensors, including video and its swivel turret, give it additional capability.

Potential for dose limitation: The MF2 robot was used at the Chernobyl nuclear power plant.

References:

Duration: **from**: 1985 **to**: 1988 **Funding**:N/A

Status: In progress. **Last Update**: May 12, 1988

BNL ALARA Center Data Base

Country: JAPAN **ID**: E-224

THE RIS (REMOTE INSPECTION SYSTEMS) DEVICE

Keywords: REMOTE SYSTEMS; ROBOTICS; INSPECTIONS; SURVEILLANCE; TELEOPERATED DEVICES

Principal Investigator: **Project Manager:**

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Objectives: To develop a robotics vehicle suitable for nuclear power plant inspections and surveillance.

Comments: The four-wheel, 550 lb. vehicle has a robotic arm with a lifting capacity of 22 lbs. The arm is equipped with a vibration detector and a thermometer. The tether is played out by a board cable reel drum. The vehicle has a turning radius of 3.3 ft. Its navigation is controlled by means of guide aluminium tapes, pasted to the floor of the facility, which are read by an on-board optical reflective sensor. The device is teleoperated. It is computer controlled, with transmission of information to the control console through a coaxial cable. Its lead acid battery provides power for up to two hours.

Potential for dose limitation:

References:

Duration: **from**: 1986 **to**: 1989 **Funding**:N/A

Status: In progress. **Last Update**: September 8, 1987

Country: U.S.A. **ID:** E-225

EFFECT OF THE TEMPERATURE OF THE STEAM GENERATOR TUBE ON THE STRESS CORROSION CRACKING OF ALLOY 600

Keywords: COMPONENT RELIABILITY; STEAM GENERATOR; STRESS CORROSION CRACKING; CORROSION; CRACKING; STEAM GENERATOR DESIGN; ALLOY 600; HEAT TREATMENT; OPERATING EXPERIENCE; RESIDUAL STRESS; WATER CHEMISTRY

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Objectives: To assess the relative importance of operating temperature compared to other factors such as the use of stress-relieved tubing, low design stresses, stringent water chemistry control, and non-stagnant working fluids in the stress corrosion cracking of alloy 600 tubes in steam generators.

Comments: This review study examines and compares a number of steam generator designs including Babcock & Wilcox Canada Recirculating steam generators, Babcock & Wilcox USA Once-through steam generators, and designs from other major manufacturers.

Potential for dose limitation: Analysis of laboratory data showed that the use of stress-relieved (621^{O}C(1,150^{O}F)/ 10 hrs) or thermally treated (704^{O}C(1,300^{O}F)/ 16 hrs) tubing stressed to no higher than 70% of the yield strength will survive 33 times longer at 343^{O}C(650^{O}F) than mill-annealed tubing stressed to 125% of yield strength. This improvement is at least an order of magnitude larger than the increase in mill-annealed tubing life obtainable by reducing operating temperature from 327^{O}C(620^{O}F) to 304^{O}C(580^{O}F).

References: Vaccaro, F.P., G.J. Theus and B.P. Miglin, "The Effect of Steam Generator Tube Temperature on the Stress Corrosion Cracking of Alloy 600", Nuclear Journal of Canada **1**, No.4, Dec. 1987, pp.302-312.

Duration: **from:** 1987 **to:** 1989 **Funding:** N/A

Status: In progress. **Last Update:** February 22, 1988

Country: FRANCE **ID:** E-226

STRATEGY FOR STEAM GENERATOR REPLACEMENT

Keywords: COMPONENT RELIABILITY; STEAM GENERATOR; REPLACEMENT; INCONEL-690; INCONEL-600

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Objectives: To develop a strategy for replacement of the steam generator. This strategy is to be based both on a statistical approach, in which experience from Westinghouse steam generators is projected on to EdF units, and on an individual analytical approach which is probabilistic and addresses all known failure mechanisms and their likely evolution.

Comments: At present, EdF has set a maximum on the proportion of plugged tubes in a given steam generator of about 15% (i.e. 500 tubes per SG). In 1987 the proportion was 8 % in the worst SG. Thus there is a margin available. Use of sleeving and other remedies could further slow the number of tubes requiring plugging.

EdF and Framatome have made contingency arrangements for SG replacement from 1988. Two spare sets of three SGs are available, the tubes of one set being of Inconel 690 and that of the other of Inconel 600.

On completion of this project it should be possible to determine when, and at which plant, SG replacement will be necessary. Also, it will be possible to perform SG replacement at a plant, given 6-months notice.

Potential for dose limitation: Due to the intensive preparations, it is estimated that the required outage period for a 3 loop SG replacement project will be about 5 months, with a cumulative personnel exposure of 700 rem.

References: de Surgy, J., J.P. Hutin, R. Serres, "Tube performance at EdF 900MWe units-an update", Nucr. Eng. Intl. 33, No.402,(1988),pp.21-23.

Duration: from: 1985 to: 1990 **Funding:** N/A

Status: In progress. **Last Update:** February 24, 1988

Country: U.S.A. **ID:** E-227

BORIC ACID RECLAMATION BY MEMBRANE TECHNOLOGY

Keywords: CONTAMINATION REMOVAL; BORIC ACID WASTE RECOVERY; MEMBRANE TECHNOLOGY; REVERSE OSMOSIS; ULTRA-FILTRATION; VOLUME REDUCTION

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Project Manager:

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Objectives: To develop and demonstrate a system based on advanced membrane technology to process borated process and waste water to allow recycling of otherwise waste boric acid, thus reducing waste generation and disposal from LWRs.

Comments: KLM's Boric Acid Reclamation System (BARS) has operated for nearly two years in a nuclear plant environment and processed various waste and process streams containing boric acid with excellent performance, ease of operation, minimal maintenance and a high degree of cost-effectiveness.

Potential for dose limitation: A waste volume technology which reduces borated waste and subsequent waste disposal. Remote operations exceeded expectations with minimal occupational exposure. Modular design allows ease of shielding and ease of access.

References: 1. "Boric Acid Reclamation System (BARS)", Waste Management Symposium, Tucson, Arizona, March 1986. 2. "KLM's Boric Acid Reclamation System (BARS)-An update", Waste Management Symposium, Tucson, Arizona, March 1987. 3. KLM's Optimized BARS for Silica and Waste Removal", 48th Annual Meeting International Water Conference, Pittsburgh, November 1987. 4. "Membrane Treatment Removes Silica from Borated Water, Reduces Radwaste", POWER, September 1987.

Duration: **from:** 1984 **to:** 1987

Funding: $ 700,000

Status: Commercialization efforts and long-term operational test program.

Last Update: June 13, 1988

BNL ALARA Center Data Base

Country: FRANCE **ID:** E-228

MEASURES TO IMPROVE STEAM GENERATOR TUBE PERFORMANCE AT EDF'S 900MWE PWRS

Keywords: COMPONENT RELIABILITY; STEAM GENERATOR; STEAM GENERATOR TUBE; CORROSION; INSPECTION; PEENING; PLUGGING

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Objectives: To monitor the performance of steam generator tubes in EdF 900 MWe units; to analyze trends and take appropriate measures to improve performance.

Comments: By far the most important cause of steam generator tube plugging in Edf 900MWe plants has been corrosion on the primary side. 2,551 tubes were plugged, due to defects in the roll transition zone. The defects usually appear as short longitudinal cracks. 70 per cent of steam generators with mill-annealed mechanically rolled tubes are afflicted.

The tubes are inspected by axial eddy current probe; the cracks are classified as follows:- A and B:short cracks; C and D:more significant cracks, subsequently inspected by rotating probe to determine crack length.

Plugging is done in the roll transition region when:- (a) A leak is detected by hydraulic test; (b) A significant leak is detected by helium test, or a >16-mm crack is detected by rotating probe.

Potential for dose limitation: To counter primary side stress corrosion, the following measures are being used:
* Heat treatment of all tubes from the 26th 900 MWe PWR onward.
* Shot peening or rotopeening of all mill-annealed tube bundles in the hot leg roll transition zone.
* In-situ heat treatment of small radius bends.

References: de Surgy, J., J.P. Hutin, R. Serres, "Tube performance at EdF 900MWe units-an update", Nucr. Eng. Intl. **33**, No.402,(1988),pp.21-23.

Duration: from: 1986 **to:** 1988 **Funding:** N/A

Status: In progress. **Last Update:** March 3, 1988

Country: NETHERLANDS **ID:** E-229

VERALIGHT - A NEW LIGHT MANIPULATOR FOR STEAM GENERATOR INSPECTION

Keywords: COMPONENT RELIABILITY; STEAM GENERATOR; STEAM GENERATOR; INSPECTION; STEAM GENERATOR TUBE PLUGGING; ULTRA-SONIC INSPECTION

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Project Manager:

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Objectives: To design and develop a purpose-built, light, inspection robot to inspect for the inspection of steam generators. To optimize it for positioning flexibility and low radiation exposures.

Comments: The designers of the FLEXIVERA device, a perfect tool for steam generator maintenance because of its accuracy and lifting capability, have successfully designed a light inspection robot which is well suited for carrying out steam generator inspections. This new robot, named VERALIGHT, is designed to perform light duties such as tube plugging, positioning of eddy current multi-probe holders, ultra-sonic inspection, and boroscope inspection.

Positioning flexibility and radiation exposure limitation are particular features. The weight reduction greatly improves the robot's ease of handling in front of the steam generator manway. Flexibility of positioning is optimized by the unique feature of the Veralight being able to clamp on to the tube sheet with its arm, so enabling the shoulder to swing out to a new base plate position.

Potential for dose limitation: One reason for low radiation exposures is the fact that VERALIGHT is installed from outside the steam generator channel head in less than two minutes.

References: "Veralight manipulator promises dose reduction", P.J. van Hoogstraten, Nuc. Eng. Int. **33**, Jan. 1988, 30-31.

Duration: from: 1986 **to:** 1988

Funding: N/A

Status: In progress.

Last Update: June 10, 1988

Country: U.S.A. **ID:** E-230

IN-PLANT APPLICATIONS OF NEW EDDY CURRENT INSPECTION TECHNIQUES

Keywords: COMPONENT RELIABILITY; STEAM GENERATOR; STEAM GENERATOR TUBE DEFECTS; INSPECTION; EDDY CURRENT INSPECTION

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Objectives: To develop and commercialize new inspection techniques for easier, quicker and more accurate in-plant inspections of steam generator tubes.

Comments: Three new techniques were developed for eddy current SG tube inspections:-

COMPUTER DATA SCREENING has been shown to be faster, more reliable, and more accurate than conventional analysis. It uses a computer program with special algorithms which perform the analysis with the same logic used by interpreters for the past 20 years.

The THREE FREQUENCY MIX technique, pioneered in F.R. Germany and developed in the United States, enables inspections to be performed much more rapidly, at normal production speeds.

The ROTATING PROBE technique for inspecting Row 1 and Row 2 U-bends is so accurate that during 100% inspections of three SGs which had been previously inspected by conventional techniques, twelve additional cracked tubes were located, and several tubes which previously were identified as defective were found to be free from defects.

Potential for dose limitation: The speed, ease of operation, and the improved accuracy of these techniques should result in considerable savings in radiation exposure. The rotating probe technique will also prove to be very important in assessing the degree of improvement obtained by in situ heat treatment techniques now being used for Row 1 bends.

References: Denton, C., "New inspection techniques yield good results in the field", Nucr. Eng. Intl. **33**, No.402,(1988),pp.31-32.

Duration: **from:** 1986 **to:** 1989 **Funding:** N/A

Status: In progress. **Last Update:** February 24, 1988

Country: U.S.A. **ID:** E-231

OPTIMIZATION PROGRAM FOR INDUCTION HEATING STRESS IMPROVEMENT (IHSI)

Keywords: COMPONENT RELIABILITY; IHSI; IGSCC; BWR; OPTIMIZATION

Principal Investigator:

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Project Manager:

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Objectives: To optimize the use of the induction heating stress improvement (IHSI) technique on new weldment configurations, new materials, and conditions of cooling water flow.

Comments: To date, IHSI appears to prevent intergranular stress corrosion cracking (IGSCC) completely in unflawed weldments with applied cyclic stresses of up to 1.5 times ASME Code allowable magnitudes, and to completely arrests pre-IHSI IGSCC crack depths of up to 50 per cent through-wall for cyclically applied stresses up to 1.0 times these Code allowables.

The expected completion results of the Degraded Pipe Test Programme will further boost confidence in the long-term effectiveness of IHSI as an IGSCC remedy. Also, a new EPRI-sponsored IHSI testing of Type 316NG weldments with simulated weld repairs is expected to confirm the ability of IHSI to mitigate local stress anomalies not completely addressed by previous qualification programmes.

The development of new temperature sensing systems, more efficient equipment mobilization/demobilization hardware configuration, and craft support management practices will increase productivity rates and cost-effectiveness of IHSI.

Potential for dose limitation: IHSI has been applied to approximately 4000 IGSCC susceptible weldments in over 45 plants. To date, IHSI has an excellent record of mitigating IGSCC in susceptible piping. Post-IHSI IGSCC non-destructive examination results at 11 units showed that 1067 welds were treated and only 11 "New" post-IHSI flaws were observed if changes to automated UT inspections are excluded.

References: Froehlich, C.H., N.G. Cofie, and J.R. Sheffield, "IHSI proves its worth", Nucr. Eng. Intl. **33**, No.402,(1988),pp.47-48.

Duration: *from:* 1986 *to:* 1988 **Funding:** N/A

Status: In progress. **Last Update:** June 10, 1988

Country: U.S.A. **ID:** E-232

FEASIBILITY STUDIES OF FULL SYSTEM DECONTAMINATION

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; FULL SYSTEM DECONTAMINATION; COST-BENEFIT STUDIES; PWR; BWR

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Objectives: To investigate the feasibility and the costs of carrying out full system decontamination in pressurized and boiling water reactors.

Comments: Full system decontamination is an important aspect of the research work being carried for light water reactors. Among its advantages is that where decontamination has been used regularly, it has facilitated maintenance, inspection, repair, and replacement of reactor components on CANDU reactors. Other potential advantages are increased reactor availability, plant life extension, and the reduced rate of recontamination when the fuel is decontaminated also.

Potential for dose limitation: During the study it was estimated that 5,000 cu.ft. of resin would be needed for full system decontamination of a typical BWR using the LOMI process and 1,700 cu.ft. using CAN-DECON with the fuel in place. For PWRs, 1,000 cu.ft. would be needed for both processes. The cost of waste processing and disposal were estimated to be 1/2 million dollars for each case.

The cost of each person-rem saved would range from $1,442 for BWR/LOMI, to $738 for BWR/CAN-DECON. For PWRs, the respective costs would be $1,500 for LOMI, and $1,800 for CAN-DECON. The cost for each person-rem saved with the fuel removed was higher by a factor of three.

References: LeSurf, J. E., P. Denault, and C. Herzog, "Full System Decontamination Feasibility Studies", Proceedings, EPRI Seminar on Primary Water Chemistry and Radiation Field Control, Berkeley, California, March 1988, ed. C.J. Wood, Electric Power Research Institute, Palo Alto, California.

Duration: from: 1987 to: 1989 **Funding:** N/A

Status: Completed. Final Report to be issued by EPRI. **Last Update:** June 28, 1988

Country: U.S.A. **ID:** E-233

CHEMICAL DECONTAMINATION OF BWR FUEL AND CORE MATERIALS

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; FULL-SYSTEM DECONTAMINATION; COST-BENEFIT STUDIES; BWR; NUCLEAR FUELS; REACTOR MATERIALS

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Objectives: To decontaminate two discharged BWR fuel bundles and highly irradiated stainless steel specimens using the CAN-DECON and LOMI solvents; to establish whether the solvents adversely affect the fuel bundles; and to dispose of the unique radioactive waste generated.

Comments: During tests, irradiated BWR fuel bundles and type 304 stainless steel specimens were decontaminated using AP/CAN-DECON and AP/LOMI solvents. The specimens were placed in a specially designed chamber for the tests and solvents were passed over them. After analysis the low level wastes were solidified and shipped for disposal.

Chemical analysis showed that AP/CAN-DECON removed 536 g of metals containing 31 Ci of activity and AP/LOMI removed 456 g of metals containing 106 Ci. Iron was the principal metal removed and 90% of the activity removed was from cobalt-60.

Measurements demonstrated that there was no adverse effect either on the fuel bundles or the stainless steel from the decontamination processes. There was no sign of either chemical or intergranular attack on the stainless steel. The absence of zirconium-95 in the waste resins suggested that the solvents did not attack the protective layer of zirconium oxide on fuel rods.

(continued)

Country: U.S.A. **ID:** E-233

(continued)

Potential for dose limitation: The tests established the feasibility of fuel system decontamination. Re-irradiation and examination of decontaminated fuel is the next logical step before proceeding to a full-system decontamination. Related work under this project is investigating the feasibility and cost-benefits of full-system decontamination. Full system decontamination of BWR plants is expected to save a large amount of radiation dose, particularly during major maintenance tasks.

References: 1. "Chemical Decontamination of BWR Fuel and Core Materials", EPRI Report NP-5722, April 1988, Research Reports Center, Box 50490, Palo Alto, CA 94303.

2. Ocken, H. and W. Walschot, "BWR Fuel Decontamination Qualification Project", Proceedings of EPRI Seminar on PWR Water Chemistry, Berkeley, March 1988, Ed. C.J. Wood, Electric Power Research Institute, bOX 10412, Palo Alto, CA 94304.

Duration: **from:** 1986 **to:** 1988 **Funding:** N/A

Status: Completed. **Last Update:** June 9, 1988

BNL ALARA Center Data Base

Country: U.S.A. **ID:** E-234

QUALIFICATION PROGRAM FOR WESTINGHOUSE FULL-SYSTEM DECONTAMINATION

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; FULL SYSTEM DECONTAMINATION; COST-BENEFIT STUDIES; PWR

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Objectives: To investigate the impact of full-system decontamination on PWR plants designed by Westinghouse.

Comments: A detailed program is underway to test the integrity of all primary system components to ensure that there is no adverse impact from full-system decontamination. The program is expected to be completed within three years.

Potential for dose limitation: Significant reductions in background levels are possible in plants performing full-system decontamination.

References: Miller, P., "Full System Decontamination Qualification for Westinghouse PWR Plants", Proceedings of EPRI Seminar on PWR Water Chemistry, Berkeley, 1988, ed. C.J. Wood, Electric Power Research Institute, Box 10412, Palo Alto, CA 94304.

Duration: **from:** 1988 **to:** 1992

Funding: N/A

Status: In progress.

Last Update: June 10, 1988

BNL ALARA Center Data Base

Country: U.S.A. **ID:** E-235

INVESTIGATION OF THE EFFECT OF LITHIUM HYDROXIDE CONCENTRATIONS ON PRIMARY WATER STRESS CORROSION CRACKING IN PWR PLANTS

Keywords: COMPONENT RELIABILITY; STEAM GENERATOR; STRESS CORROSION CRACKING; PWR; INCONEL-600

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Objectives: To determine the influence of PWR primary water pH on the susceptibility to cracking of Inconel-600 tube samples.

Comments: Concerns have been raised about the role of lithium hydroxide on primary water stress corrosion cracking (PWSCC) of PWR steam generator tubes. An investigation therefore was launched on the effect of lithium hydroxide concentrations on primary water stress corrosion cracking in PWRs. During the tests, conducted by Westinghouse under EPRI sponsorship, approximately 40 PWR steam generator tube samples, in a highly stressed state, were placed in an environment where PWR chemistry conditions were maintained. Data was developed to indicate the fraction of tubes which had cracked as a function of time of exposure to the chemical environment. Results indicated that:

* There were no discernable differences in the cracking rates as a function of pH for these highly susceptible materials.
* There was virtually no cracking when the materials were in either a low stressed or a low susceptible metallurgy condition; for the small number of cracks that were observed there was no indication of any influence of pH.

Potential for dose limitation: Preliminary results indicate that elevated pH chemistry should be benign to PWR steam generator materials. Since such chemistry appears to have a beneficial effect on reducing radiation fields, the results are very encouraging.

References: Shaw, R.A., "Update on Primary Side Cracking", Proceedings, EPRI Seminar on PWR Water Chemistry and Radiation Field Control, Berkeley, March 1988, ed. C.J. Wood, Electric Power Research Institute, Box 10412, Palo Alto, California.

Duration: from: 1987 **to:** 1989 **Funding:** N/A

Status: In progress. **Last Update:** June 17, 1988

Country: U.S.A. **ID:** E-236

UTILIZATION OF ENRICHED BORIC ACID IN PRESSURIZED WATER REACTOR PLANTS

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; COMPONENT RELIABILITY; ENRICHED BORON; REACTOR COOLANT CHEMISTRY; ALARA ISSUES; FUEL CYCLE EXTENSION; PWR

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Project Manager:

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Objectives: To investigate the costs and benefits associated with converting a PWR plant utilizing natural boric acid to operations using enriched boron.

Comments: A number of potential advantages are associated with the use of enriched boron in PWRs. For example, if an extension in fuel cycle was undertaken:

* An expected saving in fuel costs due to the reduction of burnable poisons used in extended cycle fuel rods.
* Elimination of boric acid as a factor in limiting fuel cycle length.
* Eliminating the need for plant modifications, which would be required if the use of natural boric acid was retained.

Potential for dose limitation: Additionally, there are ALARA related advantages. Thus, operating with elevated pH of 7.4 at 300^{o}C reduces dose rates and therefore radiation dose in PWR plants. However, this necessitates the use of additional quantities of lithium to reduce the acidity of the coolant. The use of excessive amounts of lithium is considered to be undesirable because of potential deleterious effects on fuel cladding and steam generator materials. The use of enriched boron will greatly reduce the quantity of lithium required to operate at the elevated pH values.

References: Battaglia, J.A., and J. Roesmer, "Westinghouse Studies on the use of Enriched Boron for Pressurized Water Reactors", Proceedings of EPRI Seminar on PWR Primary Water Chemistry and Radiation Field Control, Berkeley, California, 1988, ed. C.J. Wood, EPRI, Box 10412, Palo Alto, CA 94304.

Duration: from: 1987 **to:** 1989 **Funding:** N/A

Status: In progress. **Last Update:** May 27, 1988

Country: SWEDEN **ID:** E-237

THE OZONE DECONTAMINATION PROCESS (ODP)

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; PWR; OZONE; RADWASTE

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Project Manager:

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and
Swedish State Power Board
Phone: 021 107000

Objectives: To develop a one-step, room temperature, decontamination process for pressurized water reactors which does not use chelating agents and produces relatively little waste.

Comments: ODP development started with laboratory tests at Studsvik on material from Ringhals 2 and 3 reactors in 1982. Application technology for the process was developed by ASEA-ATOM in 1984-85. Field applications and recontamination measurements were carried out at Ringhals 2, under the joint sponsorship of Studsvik, ASEA-ATOM and the Swedish State Power Board in 1986-87. At present, the U.S. qualification program is in progress.

The process was been tested for general, crevice and stress corrosion with good results. In recent tests, the technique was used in realistic field conditions by decontaminating a letdown heat exchanger in the reactor coolant system. In addition, active specimens from other reactors were decontaminated in autoclaves.

Potential for dose limitation: During the tests with the heat exchanger an area of $45m^2$ was decontaminated. The decontamination time was 35 h and the rinsing time an additional 14 h. Of the original activity of 8.6 Ci(3.2 E11 Bq) almost all was removed. It was estimated that a dose of 29 milliperson-Sievert would be required. The actual dose required to complete the decontamination was 16 milliperson-Sievert.

References: Menon, S., "The ODP Decontamination Process", Proceedings of EPRI Seminar on PWR Primary Water Chemistry and Radiation Field Control, Berkeley, California, 1988, ed. C.J. Wood, EPRI, Box 10412, Palo Alto, CA 94304.

Duration: from: 1982 **to:** 1989 **Funding:** N/A

Status: In progress. **Last Update:** June 6, 1988

Country: U.S.A. **ID:** E-238

EVALUATION OF THE STUDSVIK/ASEA-ATOM OZONE DECONTAMINATION PROCESS (ODP)

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; PWR; OZONE; RADWASTE

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Project Manager:

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Objectives: The program has the following objectives:-

* To quantify the ODP process effectiveness for inconel steam generator tube material.
* To optimize the application parameters for effectiveness and corrosion.
* To determine the effects of boric acid and velocity.

Comments: The one-step ODP decontamination process, designed for PWR oxides, has a number of advantages. It can be applied at ambient temperature so that heatup/cooldown cycles are eliminated and equipment reduced. Moreover, recontamination is very low, and there are no chelates or hazardous wastes so waste disposal is simplified. Lastly, because it is a single step process, waste volume is reduced and minimum operations are required.

Potential for dose limitation: During the qualifying program inconel SG tubes from 3 PWRs and a stainless steel manway cover from one will be prepared by PNS. Corrosion specimens for a range of materials of interest will also be included.

For more information on the ODP process see also sheet E-88-237 of the data base.

References: Schneidmiller, D., "Evaluation of the ODP Decontamination Process", Proceedings of EPRI Seminar on PWR Primary Water Chemistry and Radiation Field Control, Berkeley, California, 1988, ed. C.J. Wood, EPRI, Box 10412, Palo Alto, CA 94304.

Duration: from: 1987 **to:** 1989 **Funding:** N/A

Status: In progress. **Last Update:** June 24, 1988

Country: U.K. **ID:** E-239

DECONTAMINATION WASTE MANAGEMENT DEVELOPMENTS

Keywords: CONTAMINATION REMOVAL; RADWASTE; CHELANT DESTRUCTION; DECONTAMINATION; WASTE VOLUME REDUCTION

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Project Manager:

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Objectives: To develop a low temperature ambient pressure process for the destruction of chelant bearing IX resins as a defence against future regulatory exclusion of chelants from burial sites.

Comments: The presence of chelants can enhance the mobility of radionuclides in groundwater within a wet waste site and so there is some risk that the burial of chelants may be restricted or curtailed. A number of processes are available to destroy the chelants removed from reactor systems during decontamination. The destruction of chelants brings other advantages, such as waste volume reduction, increased resistance to radiolytic damage and mitigation of the instability of organic resin. The aim of the program is to perform the 'underwater combustion' of the resin, using hydrogen peroxide as an intermediate reagent. At present, significant progress has been made with the oxidation of decon resins with hydrogen peroxide. The inherent mildness of the conditions and chemicals is the main advantage of the process, which has been designed for use inside the containments of nuclear power plants.

Potential for dose limitation: The project has demonstrated that oxidation of chelant bearing organic ion exchange resins is feasible. The process developed is mild and quite comparable with reactor decontamination. Development work, including demonstration at a U.S. plant, is continuing. Use of hydrogen peroxide renders the process uneconomic relative to direct burial. However, further improvements are in train which may lead to a process which is economic on volume reduction grounds alone.

References: Bradbury, D., "Decon Waste Management Developments", Proceedings of EPRI Seminar on PWR Primary Water Chemistry and Radiation Field Control, Berkeley, California, 1988, ed. C.J. Wood, EPRI, Box 10412, Palo Alto, CA 94304.

Duration: **from:** 1987 **to:** 1989 **Funding:** N/A

Status: In progress. **Last Update:** June 7, 1988

BNL ALARA Center Data Base

Country: U.S.A/F.R. GERMANY **ID:**E-240

ANTIMONY REMOVAL USING A HYDROGEN PEROXIDE PROCESS DEVELOPED BY KRAFTWERK UNION

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; ANTIMONY REMOVAL

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Project Manager:

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Objectives: To remove activated antimony from the RCS to enable refueling operation with minimal man-rem exposure.

Comments: Removal of antimony at all three units of Arizona Nuclear Power Project's Palo Verde nuclear power plant has been carried out. The first process application was at Unit 3 before the unit went critical to prevent the activation of antimony. Subsequent to this application Unit 1 was cleaned in October 1987 and Unit 2 in February 1988.

Potential for dose limitation: During the third application at Unit 2, a total of 2,000 Ci of Sb-122 and 1,700 Ci of Sb-124 were removed from the Reactor Cooling System within two days. The estimated dose rate at the refueling bridge decreased from 3,000 mR/h to less than 5 mR/h.

References: Mason, R.C. and K. Froehlich, "Recent Decontaminations at Palo Verde", Proceedings of EPRI Seminar on PWR Water Chemistry and Radiation Field Control, Berkeley, March 1988, ed. C.J. Wood, EPRI, Box 10412, Palo Alto, California.

Duration: **from**:1987 **to**: 1988 **Funding**: N/A

Status: Completed. **Last Update**: October 6, 1988

Country: NETHERLANDS **ID:** E-241

KWU FULL SYSTEM DECONTAMINATION QUALIFICATION-BÖRSSELE EXPERIENCE

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; FULL-SYSTEM DECONTAMINATION; PWR; CORD PROCESS

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Objectives: High radiation fields were present on the entire primary system surfaces of the 470 MWe PWR type Börssele plant. This seriously restricted the time that workers could stay in the steam generator channel heads to carry out maintenance tasks and lead authorities to require a full system decontamination of the entire primary system. The radiation fields had reached levels of up to 45 R/h in the steam generator channel heads.

Comments: The three available processes, LOMI, CANDECON, CORD were evaluated regarding decontamination, material compatibility, application technology, waste and application time. All three processes gave similar results regarding decontamination factors and materials capability. CORD was selected since the volume of waste was smaller, there were no hangover effects caused by residual chemicals and there was no need to cleanup the solvents between the various steps.

After an extensive testing and qualification program the CORD process has been qualified for full-system decontamination and the decontamination of the plant could be performed at any time. However, the decontamination was not performed because of the drastic reduction in radiation levels as a result of the successful program to eliminate Co-59 from the nickel plating on the fuel assembly grid spacers. It may be noted that the entire decontamination was to have been carried out with the aid of existing primary and auxiliary systems. No additional equipment would have been required.

Potential for dose limitation: Börssele nuclear power plant provided the following criteria and data: (a) the system volume was 210 m3 (b) activity inventory was 4000 Ci (c) time available for decontamination was 5 days (d) maximum allowed waste production was 6m3.

References: Olijve, J.G. and R. Riess, "KWU Full-System Decontamination Qualification at Börssele", Proceedings of EPRI Seminar on PWR Water Chemistry and Radiation Field Control, Berkeley, March 1988, ed. C.J. Wood, EPRI, Box 10412, Palo Alto, California.

Duration: **from**: 1983 **to**: 1987

Funding: by PZEM

Status: Completed.

Last Update: August 1, 1988

Country: F.R. GERMANY **ID:** E-242

KWU APPROACH TO REDUCING RADIATION FIELDS

Keywords: CONTAMINATION PREVENTION; CONTAMINATION REMOVAL; PWR; WATER CHEMISTRY

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Objectives: To develop a plant specific procedure for optimizing Lithium-Boron coordination in order to reduce radiation fields in PWR plants.

Comments: Previous work on the two three-loop nuclear power plants, Gosgen and Neckarwestheim demonstrated the importance of cobalt-free fuel assembly grid spacers in reducing radiation fields in PWR plants.

This work starts from the premise that since the chemical composition and morphology of the oxides that form crud is not well known and since operating at lower or higher pH values is likely to increase dose rate buildup, a plant specific, optimum value of pH should be determined by empirical methods.

The method seeks to use as input operational data on B, Li, H_2, and operating temperature, obtained on a daily basis, and attempts to generate the solubility as output. In doing so, the available data on boric acid, magnetite, nickel-cobalt ferrite, and lithium hydroxide solubility and disassociation are utilized.

The outcome is plant specific optimum values for Li/B coordination.

Potential for dose limitation: Optimization of plant water chemistry is probably the most cost effective approach to lowering radiation exposures. The present approach is likely to further improve plant chemistry and reduce radiation levels in KWU's PWR plants.

References: Riess, R., "KWU Approach to Reducing Fields", Proceedings of EPRI Seminar on PWR Primary Water Chemistry and Radiation Field Control, Berkeley, California, 1988, ed. C.J. Wood, EPRI, Box 10412, Palo Alto, CA 94304.

Duration: from: 1985 **to:** 1989

Funding: N/A

Status: In progress.

Last Update: July 7, 1988

Country: U.S.A. **ID**: E-243

SLUDGE REMOVAL IN PWR STEAM GENERATORS

Keywords: COMPONENT RELIABILITY; REMOTE SYSTEMS; PWR; STEAM GENERATOR; SECONDARY SIDE CORROSION; ROBOTICS; SLUDGE REMOVAL

Principal Investigator:

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Project Manager:

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Objectives: A project was initiated in 1985 to assess the available techniques for non-destructive examination of secondary side sludge and its removal with high pressure water jets (lancing). During the course of this project a prototype robot named CECIL was developed, which has the ability to accurately manipulate a high pressure lance and inspection arm within the full extent of steam generator tubes above the tubesheet. The robot was successfully piloted in an Indian Point 2 steam generator and provided excellent video pictures of sludge within the steam generator.

An ongoing extension to this work seeks to improve on this experience: A smaller robot, with easier access to all areas, will be developed and tested. A capability to lance harder sludge will be incorporated and higher water jet pressures and improved orifice configurations will be designed and tested. A movable arm compatible with the robot for foreign object search and retrieval will be designed and tested.

Comments: The three pressurized water reactors in New York State utilize recirculating steam generators. As a consequence of steam production even with high purity water, large quantities of corrosion products (sludge) concentrate on the secondary side of the tubing and lower support structure (tubesheet). This sludge, which has amounted to as much as a ton in a reactor fuel cycle in some reactors, collects aggressive impurities and alters local chemistry, such as in crevices, thereby causing tube and tubesheet corrosion.

Potential for dose limitation: The corrosion of steam generator tubes and tubesheets results in the need to plug the tubes and can necessitate replacement of steam generators with high cost both in terms of money and radiation dose. The effective removal of sludge could defer the need to replace steam generators and substantially reduce these costs. Moreover, the removal of sludge by robotic techniques would also result in greatly reduced dose to personnel.

References: None to date.

Duration: **from**: 1988 **to**: 1989 **Funding**: $ 1,7M (ESEERCO).

Status: In progress. (The project also has other co-sponsors). **Last Update**: July 6, 1988

Country: U.S.A. **ID**: E-244

TEST LOOPS FOR DOSE REDUCTION IN LWR

Keywords: CONTAMINATION PREVENTION; RADIATION BUILD-UP; WATER CHEMISTRY; HYDROGEN WATER CHEMISTRY; PWR; BWR

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Objectives: This project is intended to provide a versatile test facility to investigate LWR radiation build-up and materials issues and to conduct experiments as a basis for determining remedies.

Comments: The project will entail the construction of test loops to simulate PWR and BWR water chemistry. The PWR experimental program will examine the coordination of lithium hydroxide and boric acid at various concentrations to control acidity (pH). These concentrations are believed to affect the dissolution, transport and deposition of radioactive species (principally cobalt isotopes) in the reactor coolant system and consequently the build-up of radiation fields. In the BWR experimental program, the impact of hydrogen water chemistry (HWC) on electrochemical potential, concentration of oxidizing species, and nitrogen-16 will be investigated. HWC is being considered as a remedy to minimize the cracking of stainless steel BWR recirculation piping.

A large range of diagnostic tools will be utilized, including radioactivity transport models. Data analysis will seek to provide new interpretations, particularly in the case of BWR related measurements.

Potential for dose limitation: With present operating practices, radiation fields increase in LWRs as a function of time. The fields result in worker exposure and significantly complicate maintenance. Costly and time consuming measures which can impact outage length, such as decontamination, may be required. As an indicator of the costs to utilities, worker exposures in the 1984 and 1986 outages at Indian Point 2 was 577 and 180 man-rem. If through improved measures this exposure were halved, the resultant saving would approach $ 2 million. In addition, the lengthening of an outage costs about $ 0.5 million /day.

References: None to date.

Duration: **from**: 1986 **to**: 1990 **Funding**: $ 1,2M (ESEERCO).

Status: In progress.(The project has other co-sponsors) **Last Update**: July 6,1988

Country: U.S.A/F.R. GERMANY **ID**: E-245

DECONTAMINATION OF REACTOR COOLANT PUMP IMPELLERS AND SHAFTS

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; REACTOR COOLANT PUMP

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Objectives: To lower the radiation field to acceptable levels to enable inspection, maintenance, and/or repair of the RCP shafts.

Comments: The MOPAC process developed and patented by Kraftwerk Union, was applied at Palo Verde Unit 1 and Unit 2. This process is known for its high decontamination factors, short application time and excellent record for pump decontaminations: with more than 100 applications in Europe it was used for the first time in the U.S.

Potential for dose limitation: The dose rate in 3 ft. distance was reduced to less than 100 mR/hr from as high as 7,000 mR/hr during RCP decontamination at Unit 2 in March 1988. Process time was two days.

References: Mason, R.C. and K. Froehlich, "Recent Decontaminations at Palo Verde", Proceedings of EPRI Seminar on PWR Water Chemistry and Radiation Field Control, Berkeley, March 1988, ed. C.J. Wood, EPRI, Box 10412, Palo Alto, California.

Duration: **from**:1987 **to**: 1988 **Funding**: N/A

Status: Completed. **Last Update**:October 6, 1988

BNL ALARA Center Data Base

Country: U.S.A. **ID**: E-246

SURBOT-T: A MOBILE ROBOT FOR POWER PLANT SURVEILLANCE AND INSPECTION

Keywords: REMOTE SYSTEMS; ROBOTICS; SURBOT; SURBOT-T; REMOTE INSPECTION; NUCLEAR POWER PLANT INSPECTION

Principal Investigator:

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Objectives: To develop a mobile surveillance robot to perform remote visual, sound, radiation, and other environmental surveillance tasks within hazardous areas.

Comments: The all-terrain robot will perform remote visual and audio inspections while simultaneously monitoring the environment within contaminated areas of a nuclear power plant. The robot can be equipped with a number of sensors, instruments and even manipulator arms in order to perform a variety of inspection functions. All this is accomplished while the operator is positioned in a clean area with the portable control station and the vehicle performs its functions in a contaminated one.

High-quality data is obtained, digitized by an on-board computer, and transmitted to a portable control console where it is displayed in real-time and videotaped.

Potential for dose limitation: In-plant testing of the prototype SURBOT confirmed that it will reduce both operating costs and radiation exposure to workers at nuclear power plants (NUREG/CR-4815).

References: 1. White, J.R., "A Mobile Robot for Power Plant Surveillance and Inspection", Proc. ANS/ENS International Conference, Washington, D.C., October 1988, ed. E. Silver, ORNL, M/S 5, P.O. Box 2009, Oak Ridge, TN 37831.
2. NUREG/CR-3717
3. NUREG/CR-4815

Duration: **from**:1986 **to**:1989 **Funding**: N/A

Status:In progress. **Last Update**: November 1, 1988

Country: U.S.A. **ID**: E-247

RM-10A ROBOTIC-MANIPULATOR SYSTEM

Keywords: REMOTE SYSTEMS; ROBOTICS; REMOTE MAINTENANCE; INSPECTION; MAINTENANCE

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Objectives: To develop a manipulator unit which will replace people in hazardous environments, be easily maintainable with off-the-shelf parts, and will be able to carry out an assortment of tasks.

Comments: The portable unit can be teleoperated with man-in-the-loop controls for unstructured tasks or pre-programmed to perform routine tasks automatically. The slave assembly of the manipulator system weighs only 175 lbs. and can be transported within a hazardous area by a variety of means including a crane hook, telescoping tube bridge, forklift or an ANDROS (Advanced Remote Controlled Vehicle) base. Each arm has seven motions and can lift up to 35 lbs. at full 42 in. reach. The portable control console and master arms are located in a non-hazardous area and can operate the slave arms with man-in-the-loop or be programmed much like robots in the automated manufacturing industry.

Potential for dose limitation: Options available include radiation hardening up to 10exp8 Rad and booting of the slave assembly for wet spray decontamination. RM-10a units are being proven in nuclear power plant environments for maintenance and spent filter packing, to perform remote operations and maintenance in high-radiation cells, and by NASA to develop tooling/techniques for remote work in space.

References: 1. White, J.R., "The RM-10A Robotic-Manipulator System", Proc. ANS/ENS International Conference, Washington, D.C., October 1988, ed. E. Silver, Oak Ridge National Laboratory, Bldg. 9201-3, M/S 5, P.O. Box 2009, Oak Ridge TN 37831.
2. NUREG/CR-3717
3. NUREG/CR-4815

Duration: from:1986 to: 1989 **Funding**: N/A

Status:In progress. **Last Update**: November 1, 1988

BNL ALARA Center Data Base

Country: U.S.A. **ID**: E-248

MISR: A MINI-INSPECTION AND SURVEILLANCE ROBOT

Keywords: REMOTE SYSTEMS; ROBOTICS; REMOTE INSPECTION; NUCLEAR POWER PLANT INSPECTION; INSPECTION; SURVEILLANCE

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Project Manager:

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Objectives: To develop a small, mobile surveillance robot to perform remote visual, sound, radiation, and other environmental surveillance tasks within space restricted hazardous areas.

Comments: The robot is only 15 in. wide, 17 in. long, and 13 in. high and weighs 30 lbs. It is designed to perform remote visual and audio inspections. The tracked chasis can turn about its own axis, climb a 30 degree slope, over a 1.5 in. high obstacle, and pass through water up to 3 in. deep. It is equipped with a low-light-level camera, 6:1 zoom lens, floodlights, pan/tilt, and two-way audio. Other sensors can be installed and integrated with its on-board computer as desired.

Potential for dose limitation: In-plant testing of the protype MISR confirmed that it will reduce both operating costs and radiation exposure to workers at nuclear power plants.

References: 1. White, J.R., "A Mobile Robot for Power Plant Surveillance and Inspection", Proc. ANS/ENS International Conference, Wahington, D.C., October 1988, ed. E. Silver, ORNL, M/S 5, P.O. Box 2009, Oak Ridge, TN 37831.
2. NUREG/CR-3717
3. NUREG/CR-4815

Duration: **from**:1986 **to**: 1989

Funding: N/A

Status:In progress.

Last Update: November 1, 1988

BNL ALARA Center Data Base

Country: U.S.A. **ID:** E-249

ANDROS: AN ADVANCED REMOTE CONTROLLED TRACKED VEHICLE

Keywords: REMOTE SYSTEMS; ROBOTICS; MAINTENANCE; SURVEILLANCE

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Objectives: To develop a double-articulating, tracked based vehicle to traverse all types of terrain during remote inspections and maintenance.

Comments: The vehicle traverses nearly all types of terrain including 45 degree stairs or slopes, mud, gravel, snow, water. It can go over 16 in. high obstacles, 20 in. wide ditches and turn 180 degrees in a 44 in. wide aisle.
Standard ANDROS equipment includes the base with on-board electronics, a manipulator arm (35 lbs. lift capacity at 6 ft.), color television cameras, lights, and portable control console. Power signal transmission is via tether cable (portable on-board reel) or wireless controls.

Potential for dose limitation:

References: 1. White, J.R., "A Mobile Robot for Power Plant Surveillance and Inspection", Proc. ANS/ENS International Conference, Washington, D.C., October 1988, ed. E. Silver, ORNL, M/S 5, P.O. Box 2009, Oak Ridge, TN 37831.
2. NUREG/CR-3717
3. NUREG/CR-4815

Duration: **from**:1986 **to**:1988 **Funding**: N/A

Status:In progress. **Last Update**: November 1, 1988

11. HEALTH PHYSICS TECHNOLOGY PROJECTS

Country: U.S.A **ID: T-114**

CONTROL ROD DRIVE FLANGE RADIATION SHIELDS

Keywords: RADIATION SHIELDING; CONTROL ROD DRIVE FLANGE; RADIATION SHIELD, PORTABLE SHIELDS

Principal Investigator:

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Project Manager:

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Objectives: To minimize exposure during BWR under vessel maintenance and control rod drive placement.

Comments: The design of the radiation shields is such that they are light, quickly installed, and reusable for the life of the plant.

Potential for dose limitation: An actual reduction in under vessel radiation exposure of 72% is achievable.

References: Brown, F., Dedrich M., "CRD Flange Radiation Shields," Pennsylvania Electric Association, Fall Meeting, 1985.

Duration: from: 1982 to: 1986 **Funding**: 3 Man-years

Status: Results Published **Last Update**: August 26, 1988

Country: JAPAN **ID: T-115**

OPTIMIZATION OF RADIATION EXPOSURE IN BWRS

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; BWR; DOSE REDUCTION; ULTRASONIC INSPECTION; AUTOMATION; DECONTAMINATION

Principal Investigator: **Project Manager:**

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Objectives: A program is underway to reduce exposures in BWRs.

Comments: Measures which have played a significant role in reducing radiation dose in the new BWRs are: (1) larger containment, (2) automatic CRD handling, ultrasonic inspection, and overhauling of CRDs, (3) reactor well decontamination facility. Moreover, the number of welds has been reduced to decrease in service inspection.

Potential for dose limitation: Exposure has been reduced from the 900-1500 man-rem range/outage for the early units to 90 man-rem/outage for new BWR.

References: Nucl. Eng. Int., Vol. 31, June 1986, pp. 27-28.

Duration: from: 1978 to: 1988 **Funding**: N/A

Status: In progress. **Last Update**: August 26, 1988

BNL ALARA Center Data Base

Country: FINLAND **ID: T-116**

APPLICABILITY OF OPTIMIZATION TO PLANT SYSTEMS

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; RADIOLOGICAL PROTECTION; EQUIPMENT; PROCEDURES; HUMAN FACTOR

Principal Investigator:

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Project Manager:

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Objectives: To determine the common principle in the applied radiation protection philosophy in the Nordic nuclear power plants.

Comments: Information on 320 jobs was collected and 100 jobs were analyzed in detail to find the common principle. It was found that no formal optimization of dose reducing actions is performed and the actions are based on radiological limits and requirements set by the authorities.

Potential for dose limitation:

References: P. Hellstrom et al., "Applicability of Optimization Principle Related to Plant Systems and Constructions," RAS-410 (86) 2, August 1986.

Duration: from: 1985 to: 1986 **Funding**:N/A

Status: Completed. **Last Update**: August 26, 1988

Country: U.S.A **ID: T-117**

THE GRAND GULF NUCLEAR STATION VALVE IDENTIFICATION AND MAINTENANCE INFORMATION PROGRAM

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; VALVE IDENTIFICATION; MAINTENANCE

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Project Manager:

System Energy Resources, Inc.
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Phone:

Objectives: To reduce planning time and dose expended when working on valves in radiologically hazardous areas.

Comments: Engineering drawings do not lend themselves to quick and easy valve locating. By pre-identifying valves and posting an easy-to-read locator map at room entrances, less time and dose will be expended.

This information is also entered into the Maintenance Department data base. Valve location, required wrenches, packing material and other significant data to work on the valve can be easily and quickly found.

Potential for dose limitation: The program should result in dose savings of 4 Rem/year during plant operations, with potential for higher savings as the plant ages.

References: None.

Duration: from: 1987 to: 1988

Funding: None above normal operating cost.

Status: In progress. 85% of targeted rooms complete.

Last Update: April 21, 1987

Country: U.S.A **ID:** T-118

OPERATIONAL USES OF CLOSED CIRCUIT TELEVISION

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; CLOSED CIRCUIT TELEVISION; DOSE REDUCTION; RADWASTE REDUCTION

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Objectives: Dose reduction and reduced radwaste generation by allowing operators, health physics personnel and maintenance personnel to perform remote visual inspections of areas in which high levels of radiation and/or contamination are present.

Comments: The CCTV systems were used extensively during the refueling outage for coverage and taping of refueling activities, MSIV repairs, recirculation pump seal replacement and RHR pump removal/inspection. In addition to exposure savings, the tapes generated during these activities can be used for future pre-job planning and worker briefings.

Potential for dose limitation: Estimated dose savings for the first year of operation is 10.5 Man-Rem.

References: None.

Duration: **from:** 1986 **to:** 1988 **Funding:** Open

Status: Closed circuit television equipment is currently installed in 4 areas.

Last Update: April 21, 1987

Country: U.S.A **ID: T-119**

THE OPTIMIZATION OF THE CONTROL OF CONTAMINATION AT NUCLEAR POWER PLANTS

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; CONTAMINATION; OPTIMIZATION; MONETARY WORTH OF MAN-REM; SKIN DOSE; EXTREMITY DOSE; RADIATION-INDUCED INEFFICIENCIES; PERSONAL COMPUTER SPREADSHEET

Principal Investigator:

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Objectives: (1) To develop a conceptually simple procedure for the optimization of the control of contamination. To ensure that the procedure is able to handle a variety of problems associated with contamination control. (2) To attempt to quantify the inefficiencies and loss of productivity resulting from working in a radiation environment. (3) To develop a procedure for introducing skin and extremity doses as a component in the optimization process. (4) To propose a method for evaluating the monetary value of dose savings.

Comments: A method for selecting the optimum from a number of techniques available for the control of contamination has been developed. The method is adaptable for important engineering modifications to save radiation exposure as well as for small scale ALARA programs.

It can be used in an optimal manner in the form of spreadsheets for IBM compatible personal computers, although the manual approach is also possible. Spreadsheet software for use with the Lotus SYMPHONY program is available from the ALARA Center. The method proposed also introduces the quantification of the loss of productivity as a result of the radiation environment. It proposes a procedure for evaluating the monetary worth of radiation dose saved and also takes into account skin and extremity doses.

Potential for dose limitation: The method enables the quantification of dose and cost savings resulting from different approaches. It takes account of extremity and skin dose savings if so desired. This information should be extremely useful in making decisions on projects related to dose reduction.

References: Khan, T.A., Baum, J.W., "Optimization of the Control of Contamination at Nuclear Power Plants", NUREG/CR 5038, May 1988.

Duration: from: 1986 to: 1988

Funding:N/A

Status: Completed.

Last Update: April 24, 1987

Country: U.S.A **ID: T-120**

ASSESSMENT OF REACTOR HEAD TENSIONER SYSTEMS AT SURRY NUCLEAR POWER PLANT

Keywords: REMOTE SYSTEMS; PWR; REACTOR VESSEL HEAD TENSIONER; CLAM SHELL TENSIONER; PULLER BAR TENSIONER

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Objectives: (1) To incorporate a reactor vessel head tensioner system at Surry 1 for the 1988 outage. (2) To carry out an ALARA cost-benefit assessment of two alternate head tensioner systems.

Comments: The two types of head tensioners evaluated were: (a) the "clam shell" type (b) the power puller bar type. The ALARA evaluation showed that considerable time and man-rem can be saved by either tensioner over current operations. However the "clam shell" tensioner would cost approximately $ 1 million more to purchase than the power puller bar type. Only an additional 0.5 man-rem and 2.7 man-hours could be saved using the "clam shell" system.

Potential for dose limitation: Analysis showed that over the 17 remaining refuelings at Surry 1 Nuclear Power Plant, the Power Puller Bar tensioner could save 84 man-rem at the rate of $891/ man-rem. The Clam Shell type would save 93 man-rem at the rate of $6417/ man-rem over the same period.

References: None.

Duration: from: 1987 to: 1987 **Funding**:N/A

Status: Completed. **Last Update**: May 13, 1987

Country: U.S.A **ID: T-121**

ALARA ASSESSMENT OF USE OF EXTENDED LIFE VERSUS STANDARD LIGHT BULBS AT NUCLEAR POWER PLANTS

Keywords: COMPONENT RELIABILITY; EXTENDED LIFE LIGHT BULBS; MAINTENANCE; OPERATIONAL & MAINTENANCE TECHNIQUE

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Objectives: To evaluate the dose, cost and man-hours savings if extended life light bulbs are substituted for standard bulbs in Radiological Control Areas.

Comments: Certain light bulb suppliers offer industrial grade incandescent bulbs rated at a bulb life of 10,000 hours. North Anna and Surry Nuclear Power Plants currently use bulbs rated at 3,500 hours. The annual changeouts for the standard bulbs amount to 1,300 requiring 433 man-hours compared to 455 changeouts and 152 man-hours for the extended life bulbs. The comparative annual material costs for the standard and extended life bulbs are $1,768 and $1,219 respectively. Tests of the extended life bulbs at North Anna indicate that after 8700 hours 92% of the bulbs were still burning. The extended life bulbs have 15% less brightness than standard bulbs of equivalent wattage, so it may be necessary to use bulbs of slightly higher wattage in areas of minimal existing lighting.

Potential for dose limitation: Although exposure expenditures have so far not been tracked for this activity they are expected to be significant.

References: None.

Duration: from: 1987 to: 1987

Funding:N/A

Status: Completed.

Last Update: May 13, 1987

BNL ALARA Center Data Base

Country: U.S.A. **ID: T-122**

REACTOR VESSEL HEAD STAND SCISSOR JACK TYPE SHIELD AT PRAIRIE ISLAND

Keywords: RADIATION SHIELDING; RV HEAD STAND SHIELD; DOSE REDUCTION

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Objectives: To reduce the dose required to: (a) remove used reactor head "O" rings, (b) clean the head "O" ring surfaces, (c) install new head "O" rings, (d) replace broken "O" ring clip screws.

Comments: Cost for engineering and equipment was $45,000 per unit.

Potential for dose limitation: Dose for replacing "O" rings was reduced from 1 rem/year to less than 0.4 rem

References: None.

Duration: from: 1986 to: 1987

Funding: $ 90,000

Status: Completed on both units.

Last Update: May 11, 1987

BNL ALARA Center Data Base

Country: U.S.A. **ID: T-123**

MODIFICATION OF UPPER INTERNALS DUE TO SPLIT PIN AND CRD SHAFT PROBLEMS AT PRAIRIE ISLAND

Keywords: COMPONENT RELIABILITY; UPPER INTERNALS REPLACEMENT; SPLIT PIN REPLACEMENT

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Objectives: Remove split pins which have caused loose part problems at other plants and replace with new style and new material. Also replace flexureless inserts.

Comments: Instead of replacing just the split pins, the entire upper internals were replaced. The old internals are stored on site in a shielded cask.

During this work a critical path time saving of three weeks was achieved.

Potential for dose limitation: Dose received for unit 1 was 3.6 man-rem; for unit 2 the dose was 2.1 man-rem. Dose to replace just the split pins was estimated to be 16 to 50 man-rem. Therefore a savings of 11.5 to 45.5 man-rem was realized.

References: None.

Duration: from: 1986 to: 1986 **Funding**: $ 11,380,000

Status: Completed on both units. **Last Update**: May 11, 1987

ULTRA FILTRATION

Keywords: CONTAMINATION PREVENTION; FILTERS; CRUD; PWR; PRIMARY COOLANT SYSTEM

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Objectives: Reduce particulate activity in reactor coolant, reduce general area radiation levels, reduce hot spots, and reduce hot particle problems.

Comments: NSSS chemistry studies have shown typical crud particle sizes to be between 0.3 and 0.8 micron (99%). Three PWRs have tried ultra-small mesh size filters (0.45 micron). At first, the filters loaded up rapidly and had to be changed due to pressure drop. As the systems were cleaned up, the filter replacement rate dropped.

Previously, plants were reluctant to use ultra-small mesh size filters because it required changing the existing filter housing. Now cartridge filters are available in the ultra-small mesh size to fit the existing housings. As a result, other plants will be testing the new filters.

Potential for dose limitation: Reactor coolant activities and general area radiation levels have been reduced significantly. The radiation levels on the filters were low due to the preponderance of rust and gel-like hydrated ferric hydroxide. The radiation levels remained low due to the lack of activation of the particles which were intercepted before they could get into the core.

References: Roesmer, Josef, Ph.D., Westinghouse Electric Corp. "Coolant Composition of a Pressurized Water Reactor Near Operating Temperature," VGB Conference, Power Plant Chemistry, Essen, Germany, October 2-3, 1985.

Duration: from: 1987 to: 1989 **Funding**: N/A

Status: In progress. **Last Update**: March 1, 1989

Country: U.S.A. **ID: T-125**

SNUBBER REDUCTION AND MODIFICATION AT LA SALLE

Keywords: COMPONENT RELIABILITY; INSPECTION; TESTING; OPERATIONAL PRACTISES

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Objectives: To reduce snubber inspection exposure by an average of 50% at each refueling outage by reducing, modifying, and replacing snubbers.

Comments: The reliability of snubbers will reduce the associated inspection requirements, thus increasing plant availability.

Potential for dose limitation: It is estimated that average dose savings will be about 50 person-rem during an outage from reducing the number of snubbers, using replacement snubbers and from improvements in snubber reliability. The person-hours required for testing and inspection will also be reduced.

References: none

Duration: from: 1988 to: 1989 **Funding**: N/A

Status: Initiated. **Last Update**: March 30, 1988

Country: U.S.A. **ID: T-126**

CHEMICAL DECONTAMINATION OF PRIMARY COMPONENTS AT LA SALLE

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; BWR; EXPOSURE REDUCTION

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Objectives: To reduce primary system and RWCU piping dose rate, reduce the level of contamination and to increase the worker efficiency and performance.

Comments: Reducing the dose rate and contamination level will reduce the need for protective clothing and respirators, thereby increasing the workers efficiency, mobility, performance. In addition, less time will be spent in high radiation fields leading to diminished exposure.

Potential for dose limitation: An estimated 220 person-rem will be saved by chemical decontamination.

References: none

Duration: from: 1988 to: 1988 **Funding**: N/A

Status: In progress. **Last Update:** March 30, 1988

Country: U.S.A. **ID: T-127**

FEASIBILITY STUDY ON ENRICHED BORON FOR SURRY POWER STATION, UNITS 1 AND 2

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; COMPONENT RELIABILITY; ENRICHED BORON; REACTOR COOLANT CHEMISTRY; ALARA ISSUES; FUEL CYCLE EXTENSION; PWR

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Objectives: To evaluate all the ramifications of utilizing enriched boric acid and to determine whether conversion to the use of enriched boron was technically feasible and practical.

Comments: A number of potential advantages are associated with the use of enriched boron in PWRs. Among these are lower lithium and boron concentrations, and ALARA related benefits. A cost-benefit analysis was carried out taking into consideration the following factors:(a) boric acid costs (b) recycling of enriched boric acid (c) heat tracing and boron recovery system maintenance (d) higher RCS pH operation (e) safety analysis and reactor core design (f) fuel cycle length (g) generation of waste from the conversion effort and (h) licensibility. Both recycling and not recycling enriched boric acid were considered.

The study concluded: (1) The costs of utilizing enriched boric acid outweighed the benefits unless lower lithium concentrations can be shown to be of great benefit to the station. (2) Recycling enriched boric acid was the cheaper by a significant margin than not recycling enriched boric acid.

(Continued on next page)

Country: U.S.A. **ID: T-127**

(Continued from previous page)

Potential for dose limitation: ALARA savings of \$16,000,000 to \$46,000,000 were estimated due to the possibility of operating at elevated RCS pH(300^{o}C) levels by use of enriched boron. If these savings were included, there would be a net positive benefit from utilization of enriched boron with recycling. However, these savings were disregarded since it appears that the use of enriched boron may not be necessary for operation at elevated pH levels. At Surry it should be possible to elevate RCS pH by increasing LiOH concentrations similar to what was done at Ringhals Nuclear Power Station.

References: Rodill, W.B., "Virginia Power Studies on the use of Enriched Boron for Surry Pressurized Water Reactors", Proceedings of EPRI Seminar on PWR Primary Water Chemistry and Radiation Field Control, Berkeley, California, 1988, ed. C.J. Wood.

Duration: from: 1987 to: 1988 **Funding:** N/A

Status: Completed. **Last Update:** June 27, 1988

Country: JAPAN **ID: T-128**

ASPECTS OF OPERATING EXPERIENCE OF KANSAI ELECTRIC'S PWR PLANTS IN JAPAN

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; COMPONENT RELIABILITY; WATER CHEMISTRY; SECONDARY SIDE CHEMISTRY; IMPROVED MATERIALS ; PWR

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Objectives: To further improve the performance of Kansai Electric's PWRs in terms of capacity factor and radiation dose.

Comments: The performance of Kansai Electric's PWRs has lately been very good. Some of the reasons for this are: (a) timely analysis of abnormal occurrences, (b) high grade water chemistry control, and (c) maintenance for optimum plant operation. Around 1980 problems associated with support pins of control rod cluster guide tubes were experienced and there was significant degradation of steam generator tubes. Adoption of new heat-treated material with improved anti-stress corrosion cracking characteristics solved the former problem. The later problem was solved by changing the secondary water chemistry from phosphates treatment to All Volatile Treatment (AVT) and various countermeasures were implemented such as sludge lancing of steam generator secondary side, improvement of eddy current testing technology, and development of preventive maintenance technology.

Potential for dose limitation: The average dose rate in the plant has been gradually reduced through the careful control of pH. The pH (at 285^{O}C) is kept at within 0.2 units of 6.9. Typical dose rates in the steam generator channel head of OHI unit 2 are between 6 and 7 R/h. Annual inspection is mandatory in Japan and includes the following tasks: (a) maintenance and functional tests of components and systems, (b) refueling, (c) preventive maintenance, and (d) voluntary maintenance. With the measures described above and careful outage planning, plant dose during annual inspection (for example for Takahama units 3 and 4) has been kept between 40 and 70 person-rem.

References:

Duration: from: 1986 to: 1989

Funding: N/A

Status: In progress

Last Update: May 31, 1988

Country: U.S.A. **ID: T-129**

CONSOLIDATED EDISON'S DOSE RATE REDUCTION PROGRAM FOR INDIAN POINT UNIT 2

Keywords: CONTAMINATION PREVENTION; COMPONENT RELIABILITY; CONTAMINATION REMOVAL; PH CONTROL; ELECTROPOLISHING; COBALT REMOVAL; FILTRATION; PWR

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Objectives: To reduce the radiation fields and plant dose at Indian Point Unit # 2.

Comments: A multifaceted approach is being used to reduce radiation fields and occupational radiation exposure at Indian Point # 2:
* Approval is being sought from NSSS vendor to increase lithium levels from 2.2 to 2.7 ppm to maintain vessel pH between 7.0 and 7.4
* Downstream demineralizer filters are to be changed to progressively reduce particle size that they will pass to 40, 20, and 10 microns.
* Fuel design has been changed from inconel to zircalloy grid straps to minimize sources of Cobalt-59.
* Oxygen concentration in make up water will be reduced to less than 100 ppb.
* Although there are no short term plans to replace steam generators, new ones, with low cobalt content in tube materials have been ordered and received.
* Cobalt deposition rate on a steam generator diaphragm that has been electropolished, is being monitored to see if electropolishing of new steam generators will be effective.
* Inspection of spent fuel elements is being carried out to investigate the efficacy of scrapping.

(continued)

BNL ALARA Center Data Base

Country: U.S.A. **ID: T-129**

Potential for dose limitation: Since elevated pH chemistry was initiated at Indian Point # 2 a number of improvement have been observed. Among these are:
* Lower contamination levels in steam generator channel heads.
* Lower corrosion product levels in the RCS.
* Lower Co-58 concentrations.
* Radiation levels on depleted resin are higher than measured previously.
* Radiation levels are lower on the filters and less frequent replacement of filter cartridges is required.

References: Parry, J.O., and W.A. Homyk, "Effects of Changing Reactor Coolant pH at Indian Point #2-1988 Update", Proceedings of EPRI Seminar on PWR Primary Water Chemistry and Radiation Field Control, Berkeley, California, 1988, ed. C.J. Wood Box 10412, California.

Duration: from: 1987 to: 1989 **Funding:** N/A

Status: In progress. **Last Update:** June 20, 1988

Country: U.S.A. **ID: T-130**

CONSOLIDATED EDISON'S FULL-SYSTEM DECONTAMINATION STUDIES

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

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Objectives: Due to certain design features and some past operational practices, Indian Point #2 has developed high radiation levels in the reactor coolant system. The object of this project is to qualify full-system decontamination to reduce the radiation fields and plant dose at Indian Point Unit #2.

Comments: Over the past three years Consolidated Edison has funded projects to test different solvents. During the tests the Indian Point # 2 corrosion film has been characterized, laboratory tests have been performed on plant samples, and plant alloys and materials have been tested for corrosion and cracking. The results were sent to the NSSS vendor for review. Three main concerns were identified from the vendor:
* The tests were not performed with boron present.
* The test were carried out at too low a flow rate.
* There was concern that the reactor coolant pump seals and the control rod drive mechanism would be damaged by any decontamination solvent.

Since then a three phase program has been developed to qualify at least two solvents for full system decontamination. The phases are: (I) complete plant specific tests, (II) perform safety evaluation and obtain NRC approval, and (III) perform cost benefit analysis and implement program.

Potential for dose limitation: The program should be completed in approximately three years. With the continued pressure to reduce radiation exposure and improve productivity, the justification is present for program implementation as soon as all safety concerns are resolved.

References: Parry, J.O., "Consolidated Edison's Study on Performing a Total Reactor Coolant System Decontamination", Proceedings of EPRI Seminar on PWR Primary Water Chemistry and Radiation Field Control, Berkeley, California, 1988, ed. C.J. Wood, EPRI, Box 10412, Palo Alto, CA 94304.

Duration: from: 1987 to: 1989

Funding: N/A

Status: In progress.

Last Update: June 20, 1988

Country: U.S.A. **ID: T-131**

LIVE-LOADING OF VALVE PACKING

Keywords: COMPONENT RELIABILITY; VALVE; VALVE PACKING; LIVE-LOADING

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Objectives: To mitigate or prevent leakage from valve packing. The "live loading" will allow for constant packing gland pressure on the valve stem.

Comments: Majority of valves have been identified to have the modification performed during the first refueling, 1st quarter of 1989.

Potential for dose limitation: Valve packings will require less maintenance. Leakage will be reduced. This will lead to lower airborne contamination levels.

References: None to date.

Duration: from: 1987 to: 1989 **Funding**: N/A

Status: In progress. **Last Update**: August 11, 1988

Country: U.S.A. **ID: T-132**

SNUBBER OPTIMIZATION

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; SNUBBER; SNUBBER REDUCTION; ALARA OPTIMIZATION

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Objectives: The objective of snubber optimization is to reduce the number of snubbers throughout the facility, concentrating on higher areas first. Snubbers have to be inspected on a periodic basis; thus this program will decrease the number of inspections performed.

Comments: Progress is slow. Each analysis requires 3-4 months to perform.

Potential for dose limitation: Fewer entries into high radiation areas to perform snubber inspections.

References: NUREG-1061

Duration: from: 1987 to: 19 **Funding**: N/A

Status: Initiated. **Last Update**: August 11, 1988

Country: U.K. **ID: T-133**

THE SIMVIDOSE SYSTEM FOR MONITORING PERSONNEL DOSE

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; MONITORING; VIDEO SYSTEM; DOSE; DOSE RATE

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Objectives: To develop a system which superimposes dose rates and total dose on video pictures to allow radiological protection staff to link dose and dose rate with particular tasks being carried out. Such a system would allow precise targeting of radiation protection measures and efficient use of the workers time, and could be used for remote dose control and as a training aid.

Comments: With the SIMVIDOSE (SIMultaneous VIdeo DOSE) system the worker concerned is watched by TV cameras and information about his dose and dose rate is superimposed on the picture, identifying any procedures where the radiation hazard is high.

Current developments are concerned with improving reliability and ease of use. Future developments could include multiple monitors and further data analysis software.

Potential for dose limitation: By recording the combined display the exact details of the exposure can be seen and protection measures targeted to the maximum effect.

The equipment can be set up in a low dose area from where health physics staff can continuously monitor the dose rates to which workers are exposed without incurring dose themselves. Contractors about to take over could see recordings of earlier entries and locate high dose areas. Modifications could then be planned to take account of the dose rates. In addition the system can also be utilized to inform when operational dose control limits were being approached so that workers could be instructed to stop at a convenient point in the job.

References: 1. Meggitt, G.C. and A. Cook, "Monitoring of dangerous environments", UK Patent Application G B 2142 500A.

2. Jackson, R.G. and G.C. Meggitt, "Keeping dose rates in view", Nucl. Eng. Int. 3, August 1988, pp 48-49.

Duration: from: 1988 to: 1990 **Funding**: N/A

Status: In progress. **Last Update**: September 26, 1988

Country: U.S.A./FRANCE **ID: T-134**

THE SURROGATE TOUR INTERACTIVE VIDEODISC SYSTEM

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; MAPPING; MONITORING; RADIATION MAPPING; VIDEODISC; SURROGATE TRAVEL

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Project Manager:

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Objectives: To use the combined technology of the laser VIDEODISC and the personal computer to simulate motion and provide detailed visual information of contaminated and restricted areas.

Comments: The laser VIDEODISC contains tens of thousands of pictures of the contaminated or restricted areas of the plant and related components and equipment. The computer software accesses this database to provide a photographic display on a TV monitor and to give supporting information to the user on the computer monitor. The system may be utilized to:
- access any plant area.
- travel through any area at various speeds using the joystick.
- look in any direction, view components, access equipment.
- use overlays on pictures.
- zoom and pan the photographic data base.
- program recommended routes.
- enter and view results of radiation surveys.
- get a hard copy print of any picture.

Potential for dose limitation: The Surrogate Tour can contribute to improved preparation and training of personnel. It also interfaces with a database that contains radiation survey information. The system automatically generates graphic overlays that show the most recent radiation levels superimposed on floor plans and survey maps. All previous data is saved and hard copy outputs of survey results and trends are available.

References: None to date.

Duration: from: 1987 to: 1989

Funding: N/A

Status: In progress.

Last Update: October 24, 1988

BNL ALARA Center Data Base

Country: U.S.A. **ID: T-135**

EFFECTIVE SHIELDING TECHNIQUES AT FARLEY NUCLEAR PLANT

Keywords: RADIATION SHIELDING; DOSE REDUCTION; WATER SHIELDS; PORTABLE SHIELDS; SHIELDING INSTALLATION TOOLS

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Objectives: To develop and deploy effective shielding configuration for improved exposure reduction to workers.

Comments: Examples of types of shielding being deployed are:

* Temporary Reactor Head Shielding - General area dose rates were reduced from 125 mR/hr to 20 mR/hr on unit 1 and from 100 mR/hr to 15 mR/hr on unit 2.
* Temporary Shielding Frames - base plates and shielding frames were installed in both units to allow installation of temporary shielding rapidly around the intermediate loop RCS piping and the regenerative heat exchanger. Reduced general area dose rates by one third to one half.
* Shielding the Grating of RCP/SG Cavities - Two layers of lead blankets were installed over the grating in all three cavities of each unit. Dose rates were reduced from 40 mR/hr to 15 mR/hr and 60 mR/hr to 15 mR/hr respectively in the two units.
* Shadow Shielding for Piping - shadow shields were installed under horizontal piping runs without need for loading evaluation. Dose rates were reduced from 220 mR/hr to 25 mR/hr.
* Water Shields - these are used in numerous plant areas. They can be rapidly moved into position for use. Using quick disconnects they can be filled or emptied remotely.
* New Blanket Sleeves - blanket decon time and radwaste have been reduced by means of these blanket sleeves.

(Continued on next page)

Country: U.S.A. **ID: T-135**

Potential for dose limitation: The exposure savings that resulted from use of these techniques are:

Reactor head temporary shielding - 12 to 13 Rem
Shielding intermediate loops and regenerative heat exchanger - 40 to 100 Rem
Shielding RCP/SG cavities - 5 to 12 Rem
Shadow shielding of piping - 4 Rem
Water Shields and portable shield frames - 1 Rem
Shielding blanket sleeves and miscellaneous tools - 0.5 Rem

References: Patton, P., "Effective shielding techniques during refueling outages to reduce worker exposures at Farley Nuclear Plant", Proceedings 10th Westinghouse REM Seminar, Pittsburgh, 1988, ed. C. Bergman, Westinghouse Electric Corporation, P.O. BOx 355, Pittsburgh, PA 10532.

Duration: from: 1987 to: 1988 **Funding**: N/A

Status: Completed. **Last Update**: September 27, 1988

BNL ALARA Center Data Base

Country: U.S.A. **ID:** T-136

COMPARISON OF FOREIGN AND DOMESTIC REACTOR PROCESSES WHICH CONTRIBUTE TO OCCUPATIONAL DOSE

Keywords: COMPONENT RELIABILITY; CONTAMINATION REMOVAL; CONTAMINATION PREVENTION; OPERATIONAL AND MAINTENANCE TECHNIQUES; REMOTE SYSTEMS; COBALT; CHEMISTRY

Principal Investigator:

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Objectives: To examine in detail operational reactor processes which affect occupational exposure in foreign and domestic nuclear power plants.

Comments: Emphasis will be placed on collecting and analyzing information on the factors identified in NUREG/CR-4381, " Summary of Comparative Assessment of US and Foreign Nuclear Power Plant Dose Experience ". Particular attention will be given to chemistry and to water purification in this study.

Potential for dose limitation: As shown in NUREG/CR-4381, it is likely that plant chemistry and primary system purification have impacts on occupational exposure which may cause plant-to-plant differences as large as 70%. Other factors identified ranged from 20 to 60%. Documentation of successes in each area are important for continued improvements in US collective dose.

Data will also be collected on the modifications already made to nuclear plants and their associated degree of success in dose reduction. The information gathered during this study will be made available to the nuclear industry and should be helpful in their dose reduction efforts.

References: Baum, J.W. and J.R. Horan, "Summary of Comparative Assessment of US and Foreign Nuclear Power Plant Dose Experience", NUREG/CR-4381, October 1985.

Duration: **from:**1988 **to:**1989 **Funding:** $100,000

Status: In progress. **Last Update:** November 21, 1988

BNL ALARA Center Data Base

Country: U.S.A. **ID:** T-137

HOT PARTICLE PRODUCTION, MITIGATION AND DOSIMETRY

Keywords: CONTAMINATION REMOVAL; CONTAMINATION PREVENTION; HOT PARTICLE; DOSIMETRY; SOURCE TERM; MITIGATION

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Objectives: To provide data on the production, transport, mitigation, dosimetry, and biological effects of small radioactive particles such as are produced in the primary systems or other related systems of nuclear power plants.

Comments: Small radioactive particles, ranging in size from typically one micron to a few hundred microns in diameter, have become a major problem in nuclear power plants. They have high specific activity and when deposited on the skin or inhaled, cause intense irradiation of small regions of tissue.

This study will review control and mitigation techniques, study the potential for inhalation and ingestion, investigate related biological impacts and related dosimetric interpretations, and recommend needed animal experiments. It will also define methods for assessing dose from hot particles.

Potential for dose limitation: The health hazard from hot particles is known to be much different than for equal deposition of energy over a larger volume of tissue. However a number of uncertainties remain. Moreover, there is a growing concern not only with regard to potential health effects but also personnel relations and morale. This study is designed to better define biological effects and appropriate dosimetric methods related to hot particles and serve as a basis for rule making.

References: None to date.

Duration: **from**:1988 **to**:19

Funding: $143,000 to date.

Status: In progress.

Last Update: November 21, 1988

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12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

This is the third volume in a series of reports that provide information on dose-reduction research and health physics technology for nuclear power plants. The information is taken from data base maintained by Brookhaven National Laboratory's ALARA Center for the Nuclear Regulatory Commission

This report presents information on 80 new projects, covering a wide area of activities. Projects on steam generator degradation, decontamination, robotics, improvement in reactor materials, and inspection techniques, among others, are described in the research section. The section of health physics technology includes some simple and very cost-effective projects to reduce radiation exposures.

Collective dose data from the United States and other countries are also presented. In the conclusion, we suggest that although new advanced reactor design technology will eventually reduce radiation exposures at nuclear power plants to levels below serious concern, in the interim an aggressive approach to dose reduce remains necessary.

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