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Data Base on Dose Reduction Research Projects for Nuclear Power Plants

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ABSTRACT

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

This report presents information on 118 new or updated projects, covering a wide range of activities. Projects including steam generator degradation, decontamination, robotics, improvement in reactor materials, and inspection techniques, among others, are described in the research section of the report. The section on health physics technology includes some simple and very cost-effective projects to reduce radiation exposures.

Included in this volume is a detailed description of how to access the BNL data bases which store this information. All project abstracts from this report, as well as many other useful documents, can be accessed, with permission, through our on-line system, ACE. A computer equipped with a modem, or a fax machine is all that is required to connect to ACE. Many features of ACE, including software, hardware, and communications specifics, are explained in this report.

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PREFACE

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

The first three reports in this series described approximately 380 projects; this report presents 118 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 **R** projects are on research, the **H** projects describe efforts related to health physics technology at nuclear power plants. In addition, the projects in the two data bases have been divided into six categories to make it easier to access areas of interest.

Two sets of indices are included in this report. The first set lists all research projects from this volume only, while the second refers to the health physics reports from this volume. The indices start by listing all the projects sequentially by identification number, followed by indices for the six major categories, for the project manager and principal investigator, and for sponsoring and contracting organizations. The final index of each set 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 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 the first section of this report. 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 this report for sending information to the ALARA Center. Our telefax number is (516) 282-5810.

EXECUTIVE SUMMARY

As part of a project sponsored by the U. S. Nuclear Regulatory Commission, staff at the ALARA Center of the Brookhaven National Laboratory are carrying out a program to monitor worldwide dose reduction efforts. This program includes on the one hand research that is oriented in the direction of ALARA and on the other techniques that have been found to be particularly effective in reducing exposures at nuclear power plants. As a part of this effort staff at the ALARA Center maintain two computerized databases that contain this information. The information is periodically published as a series of NUREG reports. The present report is the fourth volume in this series. It is intended to summarize the information in our database at a particular point in time.

The report presents information on 118 new or updated projects, covering a wide range of activities. Projects including steam generator degradation, decontamination, robotics, improvement in reactor materials, and inspection techniques, among others, are described in the research section of the report. The section on health physics technology contains some simple and very cost-effective projects to reduce radiation exposures. For example, these projects include an evaluation of Hot Spots associated with the spent fuel system, a laser disc based plant tour system, elimination of cobalt from the recirculation pump, effect of respirators on doses, use of robotics to remove control rod drives and an innovative tool for evaluating dose rates.

The format of the present report is somewhat different from the previous reports in this series. We have split the report into two sections: the first section is on the research projects the second is on projects related to health physics technology. The research projects are identified by the letter R (for Research) and the Health Physics Technology projects by the letter H (for Health Physics).

Each of the two sections has its own set of indices. These indices include a list of all the projects in the section. The projects are in chronological order with each new project number incremented by one. The projects in each section are then split into six categories. Thus the next index is the category index. It is hoped that by looking for projects in a particular category the reader will also be made aware of other projects not directly related to his specific subject and yet not be encumbered by a host of projects on topics of little or no interest to him.

The category index is followed by indices for project managers and principal investigators. If, for example, the projects under a particular project manager are of interest, then one of these two indices should be examined. The next two indices are for the sponsoring organization and the contracting organization. If all the projects sponsored by a particular organization are required then the Sponsor Index should be scrutinized.

The last index is the Subject Index. Each project is identified by a number of subjects. These may be subjects that the project is directly related to or subjects on which the research has an indirect impact. The subject ind[illegible] been created with some care so that the rea[illegible]an rapidly find projects of particular interest to him.

A word about the format of the information sheets is also in order. It is not the intent of the present report to give all the details of a project. It is intended to give the main objectives, describe some of the most interesting aspects in the section on comments and then summarize the conclusions in the section entitled "Remarks/Potential for Dose Limitation". Of particular interest to us is any quantitative implications as far as exposures and exposure rates are concerned. However, this information is often very difficult to infer and it would be pointless if it was just a guess.

The Remarks are followed by a list of references that can provide more information on the work. The names, addresses and telephone numbers of the persons responsible for the project have also been included. These persons may be approached for more information. Additional information on a number of these projects is also available in our archives.

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 of the NRC for his 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, our secretary, for her help in producing this report. Finally, we thank the members of the ALARA Center's industry advisory committee for reviewing this document and providing their valuable comments. The members of the advisory committee are:

C. Bergmann	Westinghouse Electric Corporation
R. Crandall	Northeast Utilities
C. Hinson	U.S. Nuclear Regulatory Commission
G. Hudson	Tennessee Valley Authority
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G. Powers	U.S. Nuclear Regulatory Commission (formerly of GE Co.)
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J. Sears	Niagara Mohawk Power Corporation
L. Smith	Institute of Nuclear Power Operations
C. Wood	Electric Power Research Institute

ALARA Center's On-line Information System: Simplification and Fax Capability

1. Introduction

The ALARA Center maintains an on-line information system on aspects of ALARA. This information system is called ACE, an acronym for ALARA Center Exchange. ACE is constantly updated and may be accessed by a personal computer and a modem. It has been described in journal articles,[1] publications[2] and presentations.[3,4,5] Information may be obtained 24 hours a day, seven days a week from the on-line system. Very recently we have greatly simplified ACE and have also added a fax feature to the system. The new fax feature allows us to expand the capabilities of the system by making it possible to transmit charts, graphs, and photographs of items of interest. The fax system will also enable persons who do not have personal computers to access the system through fax machines in their offices.

Although we try to make all the information in our databases available to our users by publishing it, there is, of necessity, a delay in the publication. The on-line system was intended as a source to provide up-to-date information to our users as it becomes available. It also was designed to enable users to search through large data bases. Thus, the data bases on the existing on-line system contain all the information since the inception of each data base, including both published and unpublished material. The new modifications have made searching and retrieval of this information extremely simple. The intent behind the fax system is different. Instead of on-line search capabilities, the system is designed to provide rapid access to charts, graphs, photographs and the as-yet-unpublished material which may be particularly relevant or topical.

In this article, we describe how to seek information through the simplified ACE system and how the fax feature may be used to assist in day-to-day ALARA research. We have now operated ACE for a sufficiently long period to have a better understanding of the needs of our users and of some of the problems they encounter. To address these needs we have decided to radically simplify the system into a handful of commands that should address most user needs for information. The first part of this article will describe the new and simplified way to use ACE, the second part will describe our two fax systems, the third part will provide other information which should assist in better utilization of ACE. At the end of this article we try to succinctly summarize the most important steps required in extracting information from ACE on a single sheet of paper. We call this sheet "The Quick Reference Guide to ACE'. It is intended that ACE users detach this sheet and keep it by them for quick reference when they access ACE.

2. Using the Simplified ACE System

We have modified ACE so that when you connect to the system and give your password you will be shown a new 'Quick Menu'. Nearly all the commands that you will need for basic searches of data are shown on this menu. Although you could go back to the old interface if you are more comfortable with that, we recommend you utilize ACE exclusively through the new commands. We will describe how each command works in a step-by-step manner.

2.1. Connecting to ACE

(1) Switch on your computer and modem.

(2) Start the ATERM communication program.

(3) Type F2 to connect to ACE.

After a brief message, you will be asked to hit RETURN and then to type in your password. Once connected you will see some explanatory screens. These screens are intended to remind you about how to use ACE. If you are already familiar with the system you can rapidly bypass these introductory screens by pressing any key when prompted to do so at the end of the screen. After the introductory screens you will see our new 'Quick Menu', shown in Figure 1.

Fig.1: The Quick Menu Of The Ace System.

Alt-B	Browse through database
Alt-K	Kwick search database
Alt-E	English Query (EQ)
Alt-F	Fax
Alt-P	Print a form
Alt-S	Print a Stack of forms
Alt-T	Print Report (EQ)
Alt-Q	Show Quick menu
Alt-H	Hangup and Exit

2.2.Browsing Through A Database

For your first attempt at working with ACE we recommend that you browse through the various databases to become familiar with the content of each one. The browsing procedure is extremely simple. Just follow these steps:

(1) Hold down the ALT key and type **B**.

You will be shown the list of databases that comprise ACE. Select a suitable database by highlighting it with the UP/DOWN arrow keys. As you highlight each database an enhanced line of text at the bottom of the screen will describe what kind of information the database contains.

(2) Select a database and hit ENTER.

Every form in the database will be opened for you, starting from the latest to the earliest.

(3) To see other screens of a form use PAGE UP and PAGE DOWN keys (not UP ARROW/DOWN ARROW).

(4) To see the next form, type F10; to see the previous form again, type F9.

2.3.Printing a Form with Your Printer

While you are familiarizing yourself with the various databases it may be a good idea to print one form from each. This will not only indicate how to obtain a hard copy of information but also show you what kind of information each database contains. This is important because you will then know which database to search for a certain type of information.

You can print any form you are viewing in the following manner:

(1) Type ALT-P.

You do not need to do anything else. The printer is already set up for you.

2.4.Rapid Searches through Keywords

You will now be familiar with the contents of each database and the type of keywords in use. The next thing we suggest is to search a database through keywords. Once again the technique is very simple. Just follow these steps:

(1) Type ALT-K.

(2) Select a database in the same manner as for browsing and hit ENTER.

You will see a blank form of the database, called RETRIEVE SPEC. Your cursor will be in a field called KEYWORDS (or use the TAB key to get to this field if the cursor is not in it).

(3) Type in a keyword and hit ENTER.

We suggest you type a partial keyword rather than the full word. Type two dots after the partial word (but not after a full word). By using partial words you are more likely to get the forms you want. For example, type CORROS.. if you are interested in CORROSION. The whole stack of forms on the subject of corrosion will be selected for you, from the latest to the earliest.

(4) To see the other screens of a form, use PAGE UP and PAGE DOWN (not UP ARROW or DOWN ARROW). To see the next form, hit F10; to see the previous form hit F9.

You may like to print either the form you are viewing or the whole stack of the selected forms.

(1) Type ALT-P to print the form you are viewing;
type ALT-S to print the whole stack of forms on the subject of CORROSION.

Once you get used to keyword searches, you may wish to refine your search. You can do so by giving several keywords together, separated by a semicolon. Thus CORROSION; ZINC will select all forms on either the subject of ZINC or CORROSION or both. The keywords &CORROSION; ZINC will select only those forms that have both ZINC *and* CORRO-

SION as keywords. Do not forget the space after each semi-colon.

Note that instead of searching through keywords, you may search through any other field. For example, sometimes you may rather wish to search by plant name. If so, instead of the field called KEYWORD use the tab key to go to the field called PLANT and type the partial name of a plant, followed by the two dots and hit ENTER. Thus, it would be better to type BYR.. for Byron.

2.5. Exiting ACE

At this point let us explain to you the extremely simple procedure to exit ACE and hang-up your telephone line. Use the following command to hang-up from any screen:

(1) Type ALT-H.

This command will close all files, exit from ACE and hang-up your telephone line.

2.6. The 'Quick Menu' and 'Old Menus'

You do not need to give the new quick commands shown on the 'Quick Menu' while looking at this menu. You can give the new commands from any screen, except for the print commands. The print commands ALT-P and ALT-S must be given while you are looking at the form you wish to print. ALT-T must be given while looking at the report you have generated through English Query. If you wish to see the 'Quick Menu' to remind yourself of the main commands available. You can do so at any time by the following command:

(1) Type ALT-Q from any screen.

Although we do not recommend it, we have retained the provision to let you revert to the old system interface, if you are more comfortable with it. To go to the old interface type the following command:

(1) Type ALT-O from any screen.

Even after going to the old interface you may return to the new 'Quick Menu' by typing ALT-Q.

2.7.Using the English Query Feature

Each of the ACE databases may be queried in simple English language. At first sight this appears to be a really simple feature. However, it has to be used with care if you wish to extract useful information. We therefore suggest that you use the other features described above to become truly familiar with the contents of each database before utilizing the English Query feature. However, once you know how to utilize this feature it has some significant advantages. One principal advantage is that you can ask for multi-column reports. Another is that you can ask the computer to do some simple calculations, like getting the average dose rate from a set of readings. You use the English Query feature in the following manner:

(1) Type ALT-E from any screen.

(2) Use the UP/DOWN ARROW keys to select the database you wish to query in English and hit ENTER.

The enhanced text at the bottom of the screen describes what each database contains.

(3) Type your query in English and hit ENTER.

The computer will flag and let you correct spelling and other mistakes. It may tell you that a certain word has a number of connotations and ask you which one you have in mind. Try to answer these questions as best you can and ask the computer to go ahead. It will give you an answer.

The best reason for using English Query is to ask for a columnar report, perhaps with some calculations. For example, you may request the following information from the database called DOSES: "Show a list of all BWR plants, their annual collective dose, the contact person and his phone number, sorted by plant name and WNEC." You will be shown a columnar list of all BWR plants listed alphabetically by plant name, their collective doses, the contact person at each plant and his phone number. The 'WNEC' above means "With No Extra Columns." The computer understands this abbreviation. Use the RIGHT / LEFT ARROW keys to see columns cut off by the screen. Use the PAGE UP / PAGE DOWN keys to see other screens for the complete report.

Another kind of question you may ask the same DOSES database may involve computation. The question could be as follows: "List all PWR plants, their dose rates and the average dose rate for all plants." You will get a report listing all the PWR plants, the steam generator channel head dose rates for each plant and at the end the average dose rate for all plants. This time the data will not be sorted alphabetically by plant name because you did not make that request.

Although it is much easier to obtain the forms of a database by the methods described previously, you can do so in English Query mode also. This may be done, for example, by asking to see specifically *the forms* of all plants whose dose rates are above a certain value.

2.8. Printing a Table Obtained Through English Query

You can print the table that you obtained through English Query through the following command:

(1) Type ALT-T while viewing the table.

This will give you a print out of the report that you have generated through English Query. Note, however, that if you asked specifically for a form or stack of forms through English Query, and wish to print the form or the complete stack of forms, then you must use the old commands ALT-P or ALT-S.

2.9. Using the Fax Feature

Powerful fax features are now available to ACE users. The use of these features requires only a handful of commands. These commands are described below in section 4. In order to start the Fax program from ACE you use the following command:

(1) Type ALT-F from any screen.

You will be taken to two new explanatory screens, which will describe how to view and fax charts, graphs and other documents available through fax. You will be asked to press any key to leave the screens and start the Fax program.

Note that: (a) you must leave the fax welcome screens before you can use the commands; (b) almost all the commands of the 'Quick Menu', except the three print commands, will work from any screen of the fax program also. This includes ALT-H, the hang-up command.

3. The Information Available through the Fax System

The ALARA Center has developed a fax service which is particularly useful for transmitting charts, graphs, photographs and finished documents. The list of the documents available through this service is also obtainable by fax. The documents are in three groups, with each document having an identification number in the group. The documents identified by the letter 'G' (for General documents) form the first group. This group is intended for our newsletter, for sections from interesting publications, for charts, graphs, drawings and photographs.

Perhaps the most interesting document on the fax system is the first document on the list with identification number G1. The ALARA Center collects information of interest in the area of ALARA research from various sources. After several items have been gathered, the Center publishes them in a newsletter, called "ALARA Notes." The items in the newsletter are generally topical. There is a time lag in publishing the newsletter because first, we have to wait for the items to accumulate, and second, because of the delays inherent in publication, which involves editing, proofreading, reviewing, and printing. Document G1, the unedited draft of the newsletter, is intended to bypass this delay. As new information becomes available, we insert it at the beginning of the draft newsletter. Eventually the draft is prepared for printing, but meanwhile, users will have access to the unedited newsletter through the fax system. The same procedure can be followed for the table of contents of the newsletter. Thus, if users want to be kept apprised of the latest ALARA developments, periodically they can have the document faxed to them. They can look at the contents on the first page to see if there are any new items of interest. The new items are located after the table of contents. When these items are received through the fax system, users can stop their fax machines so that they will not be burdened with a long facsimile of information they obtained through a previous fax.

The second document, G2, is the up-to-date list of all the documents available on the fax system. Document G3 is the present article which describes how to use ACE and its fax system. G4, G5, G6 and so on are reserved for topical charts, graphs, or photographs that we find particularly interesting and would like to bring to your attention. At present these include a chart that lists the dollar value assigned for a person-rem by all U.S. nuclear power plants.

Presently, the rest of the material available through the fax system is from our Health Physics Technology and Research data bases. The documents identified by the letter H (for Health Physics Technology) describe innovative ALARA-related techniques being used at nuclear power plants. The documents identified by the letter R describe research projects that have an impact on ALARA. This information is eventually published in the NUREG/CR-4409 series of reports.

We hope that the users will obtain the list of documents and peruse the titles for items that they can use in their work. Once the users select documents, they may fax the documents to themselves.

4. Description of the FAX System

To satisfy the users' needs, staff at the ALARA Center created two fax systems for two different types of users. The first system is called ACEFAX-1 and is intended for users who do not have access to personal computers or who do not wish to use personal computers. These users may obtain the information through their fax machines. The second system, called ACEFAX-2, is a part of our modified and enhanced on-line system. Users can view the documents available through the fax system, including charts, graphs and photographs, on their computer screen, and then, if they wish, fax documents to themselves.

The ALARA Center provides its services at no charge to the users. However, because our budget is limited, it is not possible to fax a dozen or more documents to our users every day. We therefore worked out a system by which the callers fax the documents to themselves at their own expense. One question that the user commonly asks is "How do I fax a document to myself?" In nearly all fax machines there is a mode called "polling to receive." By using this mode, you can fax to yourself any document that has been set up for polling at another fax machine or computer. In some of the newer fax machines, polling simply implies going to the fax machine, calling the number where a document has been set up for polling, and pressing a button. In the older fax machines, you need to press two or three extra buttons, but the procedure is very simple (follow the instructions in your fax machine manual).

4.1.The ACEFAX-1 System

The ACEFAX-1 system is designed for simplicity; however, it provides the ALARA-related unpublished information that is in the files of the ALARA Center. It works as follows:

- If you wish to see the latest information we have gathered for our as yet unfinished newsletter "ALARA Notes," poll the number (516) 282-7361 (FTS 666-7361 for the U.S. Federal Government), using the polling feature of your fax machine. The unfinished draft will automatically be transferred to your fax machine.
- If you wish to obtain some other document from our list of documents (also available by fax) telephone either (516) 282-4012 (FTS 666-4012) or (516) 282-3228 (FTS 666-3228) and give the identification number of the document. Then fax the document to yourself, using the fax number and polling procedure as described in the previous paragraph. Someone will normally be available at one of the two numbers from 9 a.m. to 5 p.m. U.S. Eastern Standard Time. If an answering machine picks up your call, give your name, your organization, your phone number, fax number, and the identification number of the document you need. We will fax the document to you as soon as possible. At your request, we will also fax those documents to you that are in our data bases but not available for polling so long as your request is for a reasonable number of documents.

4.2. The ACEFAX-2 System

This fax system was designed for users of our existing computer accessible on-line system, ACE. The user can see on the screen the documents that are available through a fax machine, including graphs and charts. You can select the appropriate document before faxing it to yourself through polling (how to poll is described in the previous section). Once you have selected a document for faxing, the system has been designed to give you sufficient time to go to your fax machine and fax the document to yourself. After the document has been faxed, the computer will automatically log you off the system and hang up. Your computer must have a graphics monitor to utilize ACEFAX-2. Nearly all modern personal computers are equipped with graphics cards and monitors.

4.3. Starting the ACEFAX-2 Program

The process to start the program is as follows:

(1) Use the software and the password provided to log on to ACE as described in section 2.1. After a few welcome screens, you will get to the 'Quick Menu'.

(2) Type ALT-F to go to the Fax program.

You will see two successive welcome screens describing how to use the fax system. These screens will describe *All* the commands you require to fax documents. Press any key at each screen to get to the fax program.

4.4. Using ACEFAX-2 to Fax Documents

We have crystallized the use of ACEFAX-2 into eight macro commands. **We strongly recommend that you only use these eight commands rather than the commands of the program itself. The Fax program commands are rather complicated and some of them do not work from a remote computer.** Note that the eight macro commands will work only after you leave the welcome screens. Remember also that at that point most of the commands of the 'Quick Menu' may also be used, as described in section 2.9.

4.4.1. Viewing the List of Documents on ACEFAX

To see the list of documents on the Fax system:

(1) Type ALT-L from any screen of the Fax program (not the welcome screens).

You will have a slight wait since the list will be shown in full graphics form. We have chosen the graphics form because other documents that you may wish to see may contain charts and pictures.

You will see most of the first page, including all the documents in the important 'G' group. To see the rest of the page hit the PAGE DOWN key.

You may only see the first page of a document. This is because you are at a remote computer and your capability is limited. However, the number of pages the document contains will be shown on the top right corner of the document. The single page view is not a major handicap, since most of the documents and charts are on a single page. Moreover the document view is intended to give you a preview so you can decide whether that is the document you need. This information applies not only to the list, but to all the documents available for faxing.

4.4.2. Viewing Other Documents on ACEFAX

To see other documents available for faxing:

(1) Type ALT-V from any screen of the Fax program (except the welcome screens).

You will see the list of document files, organized according to document groups and document numbers.

(2) Highlight the document file you wish to view by the UP ARROW/DOWN ARROW keys.

(3) Type ALT-R.

After a slight wait, the document you have selected will be shown to you on the white graphics screen.

4.4.3. Setting up Documents for Faxing

You should be on the graphics screen, looking either at the list of documents or any other document you have selected for viewing. You have several options:

(1) Type ALT-H to exit and hang-up.

or

(2) Type ALT-Q to exit Fax and return to 'Quick Menu'

The above commands imply that you have decided against setting up documents for faxing. These commands may be executed from any fax program screen.

or

(3) Type ALT-Y *from the graphics screen* to set up a short document (up to four pages) that you are viewing for faxing through the polling process.

You will have 10 minutes to go to your fax machine and fax the document to yourself. Ace will automatically hang-up after 10 minutes. Please note that the number you must call from your fax machine to poll the documents on the ACEFAX-2 system is different from the number you dialed to connect. The fax number for ACEFAX-2 is (516) 282-7891 (FTS 666 7891).

or

(4) Type ALT-Z *from the graphics screen* to set up the long document you are viewing for faxing through the polling process.

You will have 20 minutes to go to your fax machine and fax the document to yourself. ACE will automatically hang-up after 20 minutes.

The automatic hang-up procedure is intended to let you forget about logging off ACE while you are busy faxing documents and also to save long distance charges in case you forget to hang-up. It also ensures that ACE will be available for other users in such a situation. However, if you wish to hang-up sooner than the 10 or 20 minutes selected, then use the following procedure at your computer:

(1) Type CONTROL-ALT-DELETE

This will re-boot your computer and also hang-up the phone.

Note that it will also terminate the faxing process. The command should only be initiated after the fax has been completed or you have stopped the fax at the fax machine.

4.4.4. Clearing Errors and Pending Events on the Fax Program

When someone sets up a document for polling, it will be listed as a 'pending event' on the first screen of the fax program. It is possible that a previous user set up a document for polling but then for some reason did not fax it to himself. If such is the case, you will see one pending event listed on the screen when you enter the fax program. It is necessary that you wipe out this event before you set up your own document for faxing, else you will get his document instead.

To wipe out a pending event use the following command:

(1) Type ALT-W.

A number of persons obtain only partial documents from ACEFAX-2. This is often the case when someone wants a part of our newsletter 'ALARA Notes'. One can stop a fax machine at any time while it is transmitting a document without any harm to the system. This process, however, leads to a red flag for the next user, informing that an error has occurred. You must clear this error before you proceed for everything to function properly.

If you see a red error flag upon entering the fax program do the following to clear it:

(1) Type ALT-A.

4.4.5. Additional Advice on Using ACEFAX-2

There are a few additional points we should make you aware of:

- If you want more than one document through the same fax and are sure of the documents you need (you do not need to view them) then use the macros ALT-V to get to the list of document files; highlight each of the documents you want (except the last one) and press <ENTER>. After the last document has been highlighted, type ALT-R to view the last one on the graphics screen. Now type ALT-Y or ALT-Z to set up all the selected documents for polling through a single polling event.

  We suggest you try the multiple-document procedure only after some experience with the single-document approach.

- From the fax program you can return to the 'Quick Menu' of the main data management program or Hang-up the line by using the macros ALT-Q or ALT-H. However, you may do so only if you have not set up a document for faxing through the macros ALT-Y or ALT-Z. Once you have selected a document for faxing through these macros, the system will no longer respond to commands, but will log you off after the specified period. The only way to log off sooner is to re-boot your computer, as described in section 4.4.3.

  We therefore advise you to complete all searches of the regular on-line databases before you enter the fax program and be sure that you are prepared to fax before you use the two macros ALT-Y or ALT-Z.

- It is important to wait patiently while the screen refresh is completed, especially if you are using a slow modem and are in the graphics screen. When in the graphics screen, wait for an arrow head to appear at the top left corner of your display and the left margin bar refresh to be completed. The efficacy of a fast modern modem is especially apparent when using the fax program.

5. Advice on Using ACE

5.1. Loss of Connection

Sometimes a telephone connection may break during normal communication. If this happens, our system is designed to reset itself. We built some powerful features into the system that maintain data integrity and security during normal use. Because of this, it takes about three minutes for the system to reset in case of a break in the connection, so please wait for a few minutes before trying to log on again. Note that this only applies to a break in the connection while you were already on the system in mid-session, after your password was accepted. If, on the other hand, your log-on was rejected, for example, because you entered the wrong password, then you need wait only for one minute while our modem initializes. Generally you will connect sooner if you are patient and give the system sufficient time to reset.

5.2. Recording an ACE Session or Capturing Screens

One of the best ways to obtain and retain useful information from ACE is to electronically record sessions on your fixed disk and play them back off-line. Another important option is to capture screens of interest on your disk. You can select either approach or both together. The simple way to record sessions or capture screens is described in section 8 of the PC Anywhere III manual.

5.3. Querying the Databases in English

Querying the databases is a very powerful approach, because you can use everyday language. However, you must use this feature with care. Sometimes the information you request may not be in that database but in another one. It is important to familiarize yourself with the contents of a database by browsing through it before you try to extract information from it through English Query. For example, if you are asking the RESEARCH database to provide information about plant doses and dose rates, you will not get the answer. You must query the DOSES database.

Sometimes the information you request is of the correct type, but it has not been provided to us, and, therefore, is not in the database. In such a case, a supplementary question may clarify the problem. For example, someone queried the DOSES database about the average dose per worker in France. The query was correct, but France was not in the database, and so the word "France" was not recognized as a country. A preliminary command, such as "list all countries in this database" could have clarified this to the user.

Frequently, when the English Query Assistant asks a question that you cannot readily answer, tell the assistant to go ahead and try to answer the question. Often the answer is still the correct one. It also pays to ask the English Query Assistant to show all the forms, or show forms from a plant or country or about a subject. The information will be shown; you will not have to retrieve the information through multiple queries.

5.4. Modems

When we first designed the on-line system, we wanted to make it so simple that we would never have to write this section. However, modem technology has changed in the last year; modem communication is no longer as simple as obtaining a Hayes-compatible modem and calling a number. Advanced high-speed modems are now very economically priced. They correct data transmission errors and also compress data so that effective data transfer rates may be two to four times faster than their nominal baud rates. However, some advice is necessary in their use to get the best performance.

We use an Intel 14,400E , modem. The modem cost about $500 and has long since paid for itself by savings on long-distance charges. It is set it to communicate at all the standard values between 300 and 14,400 baud. We have conservatively set the data transmission rate at 19,200 baud (9,600 x 2). We have set the hardware flow control (CTS/RTS) lines to ON. The modem not only obeys the latest international CCITT protocols, but also the U.S. MNP 4 and 5 protocols. It also is fully compatible with the older Hayes-compatible modems.

To ensure that ACE is fully compatible with most available modems, we have linked this modem with modems from eight different manufacturers without any problems. Among these were Hayes Smartmodem 1200 and Ultramodem 9600, three Intel modems including a 9600EX modem and the SatisFaxtion board, a Vantel, a Zeos, two Practical Peripherals, a Toshiba, and a CTS Datacomm modem.

Before issuing a password, we always check it out by using one of these modems. Whenever someone who was having problems has called us, we have always been able to get him on the system. However, troubleshooting can be a problem when we do not have the user's modem manual and other pertinent information. Therefore some suggestions are provided below which may avoid problems. If you still have difficulties, please feel free to call us, either at (516) 282-4012 or (516) 282-3228.

5.4.1 Some Basic Suggestions

Any Hayes-compatible modem should be able to connect to the system. However, for modems which are not shown in the list of modems in the 'Connect Type' field of the ATERM software, you may need to do some work. You should also be made aware of the best way to get the optimum performance from the new modems which compress the data during transmission. This will not only save money but also make interaction with ACE much more pleasant. The best way to avoid any problems is to duplicate our setup at this end. Here are some useful items to check:

- Use only the remote part of the pcANYWHERE III software called ATERM to link with our system. Any other communications package will not work properly. Make sure that your modem is on, the telephone line is properly attached, and the cable connecting the modem to the serial port is in sound condition.
- For absolute compatibility with ACE, use an Intel 14.4EX or an Intel 9600EX modem. For high performance, you may also use other modems on the market which check transmission errors and compress data. These obey either the U.S. MNP4, MNP5 protocol or the CCITT V.32, V.42 bis protocol. If you use one of these modems, set the baud rate on the ATERM software to twice the modem baud rate for MNP5 and four times the baud rate for V.42 bis modems, up to a maximum of 19,200 baud. Do not set the baud rate to 'Auto'. This way you will make use of the data compression capabilities of these modems and vastly increase the data transmission speed.
- If you are using an external modem, you will normally be using the Comm Port COM1. However, you could be connected to COM2, so make sure you know which Comm port you are using.
- If you have an older copy of pcANYWHERE III your modem may not be in the list when you select 'Connect Type'. You will not find many of the new data compressing modems, such as Intel 9600EX in the modem list offered. For these modems, for optimum performance, you will need a new file, either from us or from the bulletin board of Symantec, who market the pcANYWHERE III software. This file is called AWMODEM.CNF. You will have the older version of this file which you will need to update. Proceed as follows:

  Using DOS, rename the existing AWMODEM.CNF file in your ATERM directory AWMODEM.OLD. Copy the new AWMODEM.CNF file to the ATERM directory. Then, when you look at the modem list, you will see the Intel 9600EX modem and a host of other modern modems listed. If your modem is still not in the list then you may select 'Hayes Compatible' for modems with no data compression. For modems with data compression capability (MNP5 or V.42bis) it is better to try selecting another data compressing modem similar to your own modem.
- You should set the hardware flow control (RTS/CTS) to YES. You can do so by hitting F7 for *Install* on the ATERM main screen, and selecting *Installation Options*. After this, you should be able to connect at up to 19,200 baud without any problems.
- It is better to use the highest baud rate the modem is capable of rather than the *Auto* setting. However, you may need to experiment here and see which setting gives a good connection and the highest baud rate.
- When using the Fax program and older modems that do not have error checking built-in, you may sometimes lose your screen if you are on a noisly telephone line. This is because the fax program transmits a lot of information, which really taxes a modems quality. If this happens, type ALT-H to hang up the line. You will be logged off and your screen will come back. You may like to try again on another less noisy line or at another time when there is less traffic on the telephone lines. The best long term approach is to get an error checking modem obeying one of the two protocols described in item (2). These modems are now very reasonably priced and will pay for themselves very rapidly. The screen blackout should only occur in the fax program, when using a non-error checking modem and a noisy telephone line.

5.5.Providing Information to ACE

A large part of the information on ACE has been obtained from the nuclear power industry and we provide it free to our users. However, one of the main intentions behind ACE is for users to exchange information among themselves, with minimum interaction from us. Thus, we have made provision for users to provide information to the system on-line. We also designed the system so that users may obtain a printout of the blank form of each database directly from the on-line system; users can then fill out the form at their leisure.

To provide new information to the ACE databases on-line, the user needs a special password, namely UPDATE. This password takes you to a special section of the system, which contains the previous forms provided by the utilities. To avoid filling out information which your plant has already provided, such as plant name, rated power output, and in-service date, you need to only update the information and change the date of entry. To update your form select FILE from the main menu and then

SEARCH/UPDATE from the file menu. Give the first few letters of your plant name, followed by two dots, in the field called PLANT (e.g. BYR.. for Byron) and hit F10. If you are providing information for the first time, select FILE from the main menu and then ADD from the next menu. This will create a new blank form for you to fill out.

The first field in each database asks whether this is a new form. Please type a 'Y' for Yes in this field, so that the person who examines your entry and transfers it to the main databases can tell whether or not a new entry has been made. Call us at (516) 282-3228 and apprise us of your entry. We will examine your form and, after minor editing, transfer it to the main database.

If you do not wish to update the information on-line, but would rather do it in the seclusion of your office, we suggest that you connect to our system and obtain a blank copy of the form. You may do so by using the on-line updating password, UPDATE. Select FILE and then SEARCH at the menus. Tab to the field entitled 'No.', type 'MAX' in this field and then hit F10. You should see a blank form on your screen. To print this blank form to your local printer, hit F2 and then F10. Once you have filled out the blank form you may mail or fax it to us.

The final procedure for providing information for our databases is the conventional way. You may call us by phone at (516)282-3228 and request blank forms for any database through the mail or by fax machine. We will send them to you. You can complete them and send them to us at your convenience.

6. Conclusion

The ALARA Center's information system was started with modest objectives. In the course of time a varied and many faceted system has evolved which provides the Nuclear Regulatory Commission and the ALARA community with a great deal of information in areas of ALARA research. A large amount of credit for the success of this endeavor goes to the sources that provide us information or allow us to freely use their information. For this enterprise to remain successful, it is important that our users not only continue to use the system, but also to feed information into it, so that it may be used by others for the benefit of all radiation workers. To ease this task, we have made it possible to update information on-line. We also are ready to receive any interesting information on paper and input it into the system. We hope that the new capabilities of the ACE system will prove to be very useful because up-to-date information will be available to users both through searches of the databases and through the fax system. If you have any suggestions on how to improve the system, we will be very pleased to consider them.

BNL ALARA CENTER TELEPHONE NUMBER REFERENCE GUIDE	
For information and assistance	(516) 282-4012 or (516) 282-3228
To access ACE and ACEFAX-2 (by computer and modem only)[1]	(516) 282-3481
To fax a selected document to yourself using ACEFAX-2	(516) 282-7891
To use ACEFAX-1 (no computer needed):	
• To fax the most recent ALARA Notes to yourself	(516) 282-7361
To receive a document other than ALARA Notes from the fax document list (including that list) supplied by the ALARA Center:[2]	
• Choose document(s) from list; call the ALARA Center to make your request.	(516) 282-4012 or (516) 282-3228
• Go to your fax machine and use the following number to poll to receive.	(516) 282-7361

[1]For information on how to apply for access to ACE, please call the ALARA Center information number.

[2]For current fax document list, use ACEFAX-1, ACEFAX-2, or call the ALARA Center information number.

References

1. Khan, T. A., J. W. Baum and B.J. Dionne, "ACE-ALARA Center's Information Service," *Radiation Protection Management*, Volume 8, No. 4, (July/August 1991), pp. 24-34.

2. Zheng, H., and T.A. Khan, "Make a Connection to the ALARA Center - An Introductory User Manual," BNL Technical Report, November 1989.

3. Khan, T. A., "ACE-ALARA Center's Dose-Reduction Information System," Invited presentation to the 12th Annual Westinghouse Radiation Exposure Management Seminar, October 28-31, 1990, Pittsburgh, PA, BNL-NUREG-54265.

4. Khan, T. A., "The ALARA Center and Its Information Service - ACE"' Presented at the Annual Meeting of the BWR Owner's Group, Seattle, Washington, August 1991, BNL-NUREG-46542.

5. Khan, T.A., B.J. Dionne, H. Zheng, and J.W. Baum, "Interim Results of the NEA (ISOE) Pilot Project Obtained at BNL," *Proceedings of the International Workshop on New Developments in Occupational Dose Control and ALARA Implementation at Nuclear Power Plants and Similar Facilities*, Upton, NY, September 18-21, 1989, NUREG/CP-0110, BNL-NUREG-52226, February 1990.

ACE: Quick Reference Guide

Name:____________________________ Password:__________________

Organization:____________________________ User ID:______________

1. **Connecting to ACE**
   (1) Switch on your computer and modem.
   (2) Start the ATERM communication program.
   (3) Type F2.
   (4) Hit ENTER when prompted to do so.
   (5) Type password and hit ENTER.
   (6) A 'Quick Menu' will appear, showing the 9 commands.

2. **Exiting ACE**
   (1) Type ALT-H from any screen.

3. **Browsing through a Database**
   (1) Type ALT-B from any screen.
   (2) Highlight a database with UP/DOWN arrow keys and hit ENTER.
   (As each database is highlighted its contents are described by an enhanced line at the bottom of the screen).
   (3) To see other screens of a form, use PAGEUP/PAGEDOWN keys (not UP/DOWN arrow keys).
   (4) To see the next form, type F10, to see previous form, type F9.

4. **Printing a Form**
   (1) Make sure your printer is ON.
   (2) Type ALT-P while looking at the form you want to print.

5. **Searching through Keywords**
   (1) Type ALT-K from any screen.
   (2) Highlight a database with UP/DOWN arrow keys and hit ENTER.
   (A blank form of the selected database will be opened, with the cursor in or near a field called KEYWORDS).
   (3) If necessary, use TAB key to move cursor to the KEYWORDS field.
   (4) Type a partial keyword, followed by two dots, and hit ENTER.
   (e.g. for CORROSION type CORRO..).
   (5) Use PAGEUP/PAGEDOWN to see other screens of a form.
   (6) Use F10 to see the next form on the subject; use F9 to see the previous form.

6. **Printing a Stack of Selected Forms**
   (1) Type ALT-S, while looking at one of the forms in the stack.

7. **To re-view the 'Quick Menu'**
   (1) Type ALT-Q from any screen.

8. **To use 'English Query'**
   (1) Type ALT-E from any screen.
   (2) Use UP/DOWN arrow keys to highlight the database you wish to query and hit ENTER.
   (3) Type your query in English and hit ENTER.
   (Try to read section 2.7 of the manual before using this feature).

9. **Printing a Table Report obtained through 'English Query' (blue screen).**
   (1) Type ALT-T while looking at the table.

10. **Printing Forms obtained through 'English Query' (white screen).**
    (1) Type ALT-P to print the form you are viewing.
    or
    (2) Type ALT-S to print the stack of forms selected through 'English Query'.

11. **Starting the Fax Program**
    (1) Type ALT-F from any screen.
    (You will be shown two explanatory welcome screens, describing *All* the commands required to fax documents).
    (2) Press any key to leave each welcome screen.
    (3) Give an appropriate Fax program macro command (described below).

12. **If You get disconnected during an ACE Session**
    (1) Wait 5 minutes before re-calling ACE.

13. **If Your Log on is rejected when trying to connect to ACE**
    (1) Wait 1 minute before re-calling ACE.

How to Use ACEFAX-2 to Fax Documents

14. Viewing the List of Documents on ACEFAX

(1) Type ALT-L from any screen of the Fax program (except the welcome screens)
(2) Wait for the graphics screen to be fully painted before giving the next command.
(3) The number of pages in the document will be displayed in the top right corner of the document (because you are at a remote computer, you will only be able to see the first page of a document).
(4) You will see 2/3 of the first page. To see the last 1/3 of the page use PAGEDOWN key. To go back, use PAGEUP.

15. Viewing Other Documents on ACEFAX

(1) Type ALT-V from any screen of the Fax program (except the welcome screens).
(2) You will see the list of document files, organized according to document groups.
(3) Highlight the file you wish to view using the UP/DOWN arrow keys.
(4) Type ALT-R.

16. To Exit ACE without Faxing Documents

(1) Type ALT-H from any screen (except the welcome screens).

17. To Return to the 'Quick Menu' without Faxing Documents

(1) Type ALT-Q from any screen (except the welcome screens).

18. To Set up a Short Document for Faxing

(1) Type ALT-Y from the *graphics screen* to set up a short document for faxing (up to three pages).
(2) Go to your fax machine and *poll* the document by calling (516)282-7891 (the fax machine manual will show how to poll a document set up for polling).
(3) Your computer will automatically log off and hang up after 10 minutes.

19. To Set up a Long Document for Faxing

(1) Type ALT-Z from the *graphics screen*.
(2) See item 18. above.
(3) Your computer will automatically log off and hang up after 20 minutes.

20. To Hang up before the 10 or 20 minutes set for auto hang up

(1) Stop your fax machine, if faxing is in progress.
(2) Type CONTROL-ALT-DELETE to hang up and reboot your computer (and ACE).

21. Wiping out a 'Pending Event' from a previous User

(1) Type ALT-W if you see 'Pending Events' are not set to zero on the screen when you *first* enter the Fax program.

22. Clearing All 'Errors' on the Fax Program

(1) Type Alt-A if you see a Red Error Flag upon entering the Fax program.

BNL ALARA CENTER

TELEPHONE NUMBER REFERENCE GUIDE

• To access ACE by computer and modem (dial this number from your computer).	(516)282-3481 or FTS 666-3481
• To fax a document, selected through ACE, to yourself by polling it through a fax machine (dial this number from your fax machine only).	(516)282-7891 or FTS 666-7891
• To apply for access to ACE	(516)282-3228 or FTS 666-3228

Information Request Form

BNL ALARA Center Data Base

1. Title:
2. Investigator(s): 3. Project Manager:
4. Objectives:
5. Comments:
6. Remarks/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

Information Request Form

BNL ALARA Center Data Base

1. Title:	
2. Investigator(s): (Name) (Organization) *Contracting* (Mailing Address) *Organization* (Phone)	**3. Project Manager:** (Name) (Organization) *Sponsoring* (Mailing Address) *Organization* (Phone)
4. Objectives: (Outline main objectives of the project.)	
5. Comments: (Give any comments which would shed light on the project, e.g., regarding background, perspective, progress, or significant findings.)	
6. Remarks/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, completed)	
10. References: (Recent reports, articles, or publications related to the project.)	

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

LIST OF RESEARCH PROJECTS

ID	TITLE
R-229	VERALIGHT - A NEW LIGHT MANIPULATOR FOR STEAM GENERATOR INSPECTION
R-250	DEVELOPMENT, FABRICATION, AND TEST OF THE ODEX-3 MAINTENANCE VEHICLE
R-251	SOURCEBOOK FOR CHEMICAL DECONTAMINATION OF NUCLEAR POWER PLANTS
R-252	THE NATURE AND BEHAVIOR OF PARTICULATES IN PWR PRIMARY COOLANT
R-253	PWR RADIATION CONTROL DEMONSTRATION
R-254	FIELD TESTS OF RADIATION CONTROL TECHNIQUES - 1
R-255	EFFECT OF SURFACE TREATMENTS ON RADIATION BUILDUP IN STEAM GENERATORS
R-256	MILLSTONE 1 ZINC INJECTION EVALUATION
R-257	PWR STEAM GENERATOR PRECONDITIONING STUDIES
R-258	THE TREATMENT OF RADIOACTIVE ION-EXCHANGE RESINS
R-259	PWR CORROSION TESTS USING LOMI
R-260	CRUD TRANSPORT CHEMISTRY
R-261	QUALIFICATION OF COBALT-FREE HARDFACING ALLOYS FOR LWR
R-262	PRODUCTION OF NOREM HARDFACING ALLOYS
R-263	COBALT REPLACEMENT GUIDELINES
R-264	BWR COBALT DEPOSITION STUDIES
R-265	RESEARCH REACTOR LOOP WATER CHEMISTRY STUDY
R-266	RADIATION FIELD AND DOSE DATA ASSESSMENT
R-267	PASSIVATION AND SURFACE CONDITIONING
R-268	FEEDWATER FLOW ELEMENT IMPROVEMENT

ID	TITLE
R-269	COOLANT CHEMISTRY AND RADIOLOGIS IN BOILING REACTOR COOLANT
R-270	ON-LINE MONITORING TECHNIQUES FOR REDOX POTENTIAL, HYDROGEN CONCENTRATION, AND PH IN NUCLEAR REACTOR COOLANT CIRCUITS
R-271	IN-PLANT SYSTEM FOR CONTINUOUS LOW-LEVEL ION MEASUREMENT IN STEAM-PRODUCING WATER
R-272	RESIN SEPARABILITY TO IMPROVE POLISHING UNDER MORPHOLINE AVT
R-273	RADIATION FIELD TRENDS IN WESTINGHOUSE-DESIGNED PLANTS
R-274	OXYGEN TRANSPORT IN BWR C CLE
R-275	REMOTE REPAIR TECHNIQUE FOR MSIVS
R-276	INTELLITORQUE : A SYSTEM FOR MONITORING ROOT CAUSE OF MOV MALFUNCTIONS
R-277	USING ULTRASONICS TO AVOID CHECK VALVE DISASSEMBLY
R-278	A "WET MOTOR" SEALLESS PUMP FOR REACTOR WATER CLEAN UP SYSTEM IN BWRS
R-279	A ROTATING UT SYSTEM FOR INSPECTION OF STEAM GENERATOR TUBES
R-280	THE ALOK 3 ULTRASONIC INSPECTION SYSTEM
R-281	ACOUSTIC LEAK MONITORING IN JAPAN
R-282	USE OF VIBRATION MONITORING TO ASSESS REACTOR COOLANT PUMP INTEGRITY
R-283	IMPROVED TEST METHODS FOR PLANT PROTECTIVE COATING
R-284	AUTOMATED CONTROL ROD DRIVE BOLTING WRENCH SYSTEM TO SUPPORT BOILING WATER REACTOR MAINTENANCE

ID	TITLE
R-285	MEASUREMENT OF OXIDE FILM RELEASED AS PARTICLES DURING THE CAN-DEREM DECONTAMINATION PROCESS
R-286	"WET MOTOR" SEALLESS PUMP FOR REACTOR WATER CLEAN UP SYSTEM IN BWRS.
R-287	REACTOR WATER CLEAN-UP(RWCU) SEALLESS PUMP
R-288	EXPOSURE REDUCTION MEASURES IN THE DESIGN OF SIEMENS/KWU PWR PLANTS
R-289	FULL SYSTEM DECONTAMINATION OF THE BR-3 PWR PLANT
R-289	FULL SYSTEM DECONTAMINATION OF THE BR-3 PWR PLANT
R-290	MITIGATION OF THE IMPACT OF REDUCED RADIATION EXPOSURE LIMITS ON NUCLEAR POWER PLANT OPERATIONS
R-291	SOURCES OF COBALT-60 IN THE PRIMARY SYSTEMS OF PRESSURIZED WATER REACTORS
R-292	PERFORMANCE OF IRON BASE HARDFACING ALLOYS UNDER PRESSURIZED WATER REACTOR CONDITIONS
R-293	U.K. PROGRAMME TO QUALIFY COBALT-FREE HARDFACING ALLOYS
R-294	SUPPLYING COBALT-FREE NUCLEAR VALVES
R-295	AN EXAMINATION OF FOREIGN APPROACHES TO CONTROLLING RADIATION-FIELD BUILDUP IN BOILING WATER REACTORS
R-296	GUIDELINES FOR THE REDUCTION OF COBALT FROM REACTOR SYSTEMS
R-297	BWR RADIATION FIELD TRENDS
R-298	STATUS OF ZINC INJECTION IN BOILING WATER REACTORS
R-299	EXPERIENCE WITH ZINC INJECTION AT MILLSTONE 1

ID	TITLE
R-300	CONTRO_ OF RADIATION FIELDS AT BOILING WATER REACTORS BY REDUCING IRON INPUT
R-301	EFFECT OF PRECONDITIONING ON COBALT CORROSION RELEASE RATES
R-302	RADIATION FIELD ISSUES IN SWITCHING TO HYDROGEN WATER CHEMISTRY
R-303	QUALIFICATION OF ELECTROPOLISHING FOR REPLACEMENT STEAM GENERATORS
R-304	FRENCH EXPERIENCE WITH ELECTROPOLISHING STEAM GENERATOR CHANNEL HEADS
R-305	SURFACE PRETREATMENT OF PRIMARY SYSTEM COMPONENTS TO REDUCE RADIATION BUILDUP
R-306	REDUCING RADIATION BUILDUP BY SURFACE COATING OF PRIMARY SYSTEM COMPONENTS
R-307	PWR PRIMARY WATER CHEMISTRY GUIDELINES - REVISION 2
R-308	REDUCTION OF RADIATION FIELDS BY ELEVATED PH CONTROL AT MILLSTONE-3
R-309	LOOP EXPERIMENTS ON ZINC INJECTION UNDER PWR CONDITIONS
R-310	CORROSION CONTROL AND DOSE RATE REDUCTION
R-311	EFFECTS OF PH AND LI ON PWSCC INITIATION AND GROWTH
R-312	RADIOACTIVITY PICK-UP BY CARBON STEEL AND STAINLESS STEEL IN SLIGHTLY OXIDIZING LITHIATED COOLANT
R-313	LESSONS LEARNED FROM RECENT BWR CHEMICAL DECONTAMINATION APPLICATIONS
R-314	DECONTAMINATION OF BEAVER VALLEY SGS USING THE CAN-DEREM PROCESS
R-315	FULL RCS CHEMICAL DECONTAMINATION

ID	TITLE
R-316	BWR FULL SYSTEM DECONTAMINATION
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VERALIGHT - A NEW LIGHT MANIPULATOR FOR STEAM GENERATOR INSPECTION

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

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Objectives: To design, develop and build a light robot for the specific purpose of inspecting nuclear power plant steam generators. To optimize this robot for positioning flexibility and for 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.

Remarks/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, Nuclear Engineering International, January 1988, Vol. 33, No. 402, pp. 30-31.

Duration: from: 1986 to: 1988

Funding: N/A

Status: Completed

Last Update: November 21, 1991

DEVELOPMENT, FABRICATION, AND TEST OF THE ODEX-3 MAINTENANCE VEHICLE

Keywords: REMOTE SYSTEM; ROBOTICS; MAINTENANCE

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Objectives: To develop, fabricate, and test a maintenance vehicle (robot) according to the detailed functional specifications in EPRI-NP-3941.

Comments: This project uses the previous research and demonstration walking robots (Odex I, II, and the SRL system with turret and arm) as the basis for an improved, commercial system for nuclear power plant maintenance functions. A set of end effector tools will be developed for maintenance tasks. Odex-3 will be thoroughly tested and then comprehensive in-plant testing will be conducted by the New York Power Authority personnel at Indian Point 3 and Fitzpatrick.

Remarks/Potential for dose limitation: The robot will be operated through a remote console. It should have an appreciable impact on dose reduction. This will be evaluated more fully in plant testing at Indian Point 3 and Fitzpatrick.

References: "Robot Applications for Nuclear Power Plant Maintenance," EPRI NP-3941, MARCH 1985.

Duration: from: 1985 to: 1989

Funding: N/A

Status: Completed

Last Update: April 1, 1992

SOURCEBOOK FOR CHEMICAL DECONTAMINATION OF NUCLEAR POWER PLANTS

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; CONTAMINATION REMOVAL; DECONTAMINATION; RADIATION BUILDUP

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Objectives: To compile a sourcebook of technical information on chemical decontamination technology.

Comments: The sourcebook provides an overview of chemical decontamination technology, with detailed discussions of associated technical issues, cost-benefit analysis, planning, and future developments. Highlights are as follows:

- Chemical decontamination is highly cost-effective for BWR recirculation systems; each rem saved costs less than $1000. PWR channel head decontaminations are more expensive, at about $5000 per rem saved.
- The low-oxidation-state metal ion (LOMI) process appears innocuous and is generally acceptable for full-system BWR decontamination.
- Recontamination has not proved to be a major problem with dilute chemical decontamination systems.
- New waste management techniques should substantially reduce waste volumes. Using lower resin loadings with LOMI will help avoid solidification problems.
- Improved decontamination planning has substantially reduced critical path delays and increases effectiveness.
- An EPRI methodology for cost-benefit analysis can help utilities determine when to perform decontamination.

Remarks/Potential for dose limitation: This so book includes summaries of the highlights from EPRI studies of decontamination technology, and is an important guide for the EPRI Occupational Radiation Control Program. Technologies described in the sourcebook indicate lessons learned by utilities and ways of reducing radiation exposure for future activities.

References: "Sourcebook for Chemical Decontamination of Nuclear Power Plants", EPRI NP-6433, Special Report, August 1989.

Duration: from: 1989 to: 1991

Funding: N/A

Status: Completed

Last Update: April 16, 1992

THE NATURE AND BEHAVIOR OF PARTICULATES IN PWR PRIMARY COOLANT

Keywords: COMPONENT RELIABILITY; PARTICULATES; FILTRATION; PWR; WATER CHEMISTRY; PWR COOLANT; RADIATION BUILDUP

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Objectives: To study the nature and behavior of particulates in the primary systems of operating PWR plants.

Comments: Studies at operating PWRs show that coolant-borne particulates are the source of a significant proportion of the radioactive corrosion products deposited on out-of-core surfaces. Maximum releases of particulates occur at startups and shutdowns. Lowest concentrations of particulates occurred during operation with elevated pH. Researchers installed particulate sampling systems on four PWR plants at Doel in Belgium. Isokinetic sampling provided samples whose size, radioactivity, and chemical composition could be determined. Measurements were made in operating reactors and at the initial startup of the Doel-4 plant. The results were correlated with the operating mode of the plant at the time the samples were collected.

During reactor commissioning, high concentrations of fine nickel-rich particles were present in the PWR coolant, indicating release from the Inconel steam generator tubing. The nickel content decreased from 80 to 20% as full power approached. Peak releases of particulates occurred at reactor scrams during shutdown and at startup after refueling. In the latter case, particulates contribute between 50 and 90% of activity transport.

Remarks/Potential for dose limitation: Due to the particle decay half-lives of 40 to 200 minutes, the normal chemical volume control system (CVCS) purification rate with a half-life of 8 hours cannot compete with out-of-core surfaces for particle removal. However, maximum filtration by the CVCS is important, particularly in commissioning, startup, and shutdown. Minimum particulate levels were observed in the Doel-4 reactor when operating with a pH of 7.4. The results of this project suggest that, as with soluble species, particulate activity transport can be minimized by the use of elevated pH.

References: "The Nature and Behavior of Particulates in PWR Primary Coolant", EPRI NP-6640, Final Report, December 1989.

Duration: from: 1980 to: 1989

Funding: N/A

Status: Completed

Last Update: April 1, 1992

PWR RAD!ATION CONTROL DEMONSTRATION

Keywords: CONTAMINATION PREVENTION; OPERATIONAL AND MAINTENANCE TECHNIQUES; PH; PH CONTROL; WATER CHEMISTRY; PLANT DEMONSTRATION; RADIATION BUILDUP; PWR

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Objectives: To evaluate the effects of elevated pH and lithium hydroxide on radiation field buildup and on fuel-cladding corrosion during 18-month fuel cycles at Millstone-3.

Comments: The main project task was fuel surface examination which included TV visual examination, determination of the thickness of zirconium dioxide by eddy technique, and analysis of deposited corrosion products on selected fuel elements. Subtasks involved monitoring coolant chemistry and measuring radiation exposure rates and radionuclide concentrations. After the start of the plant's second cycle, the elevated lithium demonstration was ied out with a maximum pH of 7.4 and 3.35 ±0.15 ppm lithium. Based on data obtained after one cycle of Millstone-3 elevated coolant pH operation, preliminary conclusions are as follows:

- The results do not indicate a significant lithium effect on the corr sion rate of fuel cladding because of the relatively high variability of thickness measurements on similar rods.
- There was no observable effect of the higher pH operation on any of the other fuel assembly components.
- The overall radiation exposure rate trends from cycle 1 to cycle 2 indicated a favorable effect of the higher pH operation. Steam generator channel head fields decreased by 10% and piping fields increased by 30%, compared with predicted increases of 33 and 100%, respectively.

Remarks/Potential for dose limitation: The beneficial effects of elevated lithium/pH on radiation fields during the plants' second cycle exceeded expectations, but fuel oxide thicknesses were at the high end of the Westinghouse database. A mid-1991 progress report describes the results of the third cycle of operation.

References: "PWR Primary System Chemistry: Experience with Elevated pH at Millstone Point Unit 3", EPRI NP-6938; EPRI NP-6950.

Duration: from: 1984 to: 1992

Funding: N/A

Status: In progress

Last Update: April 1, 1992

FIELD TESTS OF RADIATION CONTROL TECHNIQUES - 1

Keywords: CONTAMINATION PREVENTION; ZINC INJECTION; BWR; RADIATION FIELD CONTROL; HYDROGEN WATER CHEMISTRY; COBALT-60; ZINC-65; PLANT DEMONSTRATION

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Objectives: This is one of a series of projects whose aim is to demonstrate, on operating plants, techniques for minimizing the rate of radiation buildup on BWR recirculation piping and PWR steam generators. The object of this project is to demonstrate the technique of zinc injection passivation for operating nuclear power plants.

Comments: Selected techniques from previous EPRI projects will be field tested during this series of projects. The first EPRI project (RP2758-1), "BWR Zinc Injection Demonstration", will study the effectiveness of the zinc injection passivation process which resulted from earlier work performed in EPRI project (RP819-2).

The first EPRI project (RP2758-1) started in mid-1986. The present project involves measurement of radiation field buildup in BWR plants with injection of 5-15 ppb soluble zinc. Approximately four plants will be included, covering new plants from startup and old plants operating in both normal and hydrogen chemistry. The first plant studied is Hope Creek, which has already begun operation with zinc injection. After one year, cobalt-60 fields are amongst the lowest ever measured, while zinc-65 fields are higher than expected.

Remarks/Potential for dose limitation: The zinc injection passivation technique has been very successful in reducing radiation fields due to cobalt-60 build up. This demonstration should fully qualify the technique and is expected to play an important role in reducing radiation exposures at BWRs.

References: "Zinc Injection to Control Radiation Buildup at BWRs: Plant Demonstrations", EPRI Report NP-6168.

Duration: from: 1986 to: 1991

Funding: N/A

Status: Completed

Last Update: April 1, 1992

EFFECT OF SURFACE TREATMENTS ON RADIATION BUILDUP IN STEAM GENERATORS

Keywords: CONTAMINATION PREVENTION; ELECTROPOLISHING; PREFILMING; COBALT-60; SURFACE TREATMENT

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Objectives: The aim of this project is to determine the effects of electropolishing and prefilming on accumulation of radioactivity in PWR components; to characterize corrosion films that form during in-reactor exposure of treated surfaces.

Comments: This project was designed to establish the effectiveness of different smoothing and prefilming techniques in reducing radioactivity pickup of Type 304 stainless steel specimens exposed under PWR operating conditions. Steam generator manway cover inserts were treated and installed in the Chinon B1 PWR in France, the Doel PWRs in Belgium and Indian Point 2. Surface treatments included mechanical or electropolishing; prefilming treatments included preoxidizing in hot moist air or during hot-functional testing of the reactor. The project team evaluated the specimens by performing dose-rate and radioactivity-pickup measurements at the end of each fuel cycle. In addition, two seal plates were removed at the end of each fuel cycle, cut into small sections and subjected to metallurgical examinations.

Remarks/Potential for dose limitation: Electropolished surfaces reduced the radioactivity pickup of the three principal isotopes (cobalt-58, cobalt-60, and manganese-54) by a factor of five over as-received surfaces exposed for similar times at Chinon B1. Preoxidizing in moist air reduced radioactivity pickup by another factor of two over the electropolished-only surface. Preoxidizing during hot functional testing resulted in only a modest improvement over electropolishing. Surface examinations showed minimal nickel ferrite deposits formed on all surfaces, regardless of the surface treatment. Improvements observed in the Doel-2 reactor are much smaller than those reported for larger manway seal plates exposed in the Chinon B1 PWR. The data show that significant reductions in radioactivity pickup can be expected if key components of PWR steam generator channel heads are smoothed by electropolishing and then preoxidized under appropriate conditions before they are placed in service.

References: "Effect of Surface Treatments on Radiation Buildup in Steam Generators", EPRI TR-100059, Vol. 1 & 2, Final Report, November 1991.

Duration: from: 1987 to: 1989

Funding: N/A

Status: Completed

Last Update: April 1, 1992

MILLSTONE 1 ZINC INJECTION EVALUATION

Keywords: CONTAMINATION PREVENTION; OPERATIONAL AND MAINTENANCE TECHNIQUES; ZINC INJECTION; BWR; COBALT-60; ZINC-65

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Objectives: The aim of this project is to evaluate the effects of zinc injection at an older BWR plant.

Comments: Following a successful minitest during April and May 1987, continuous zinc injection began after the decontamination outage of August 1987. Radiation field build-up measured at the outage was approximately 50% of that measured at the end of the first cycle after an earlier decontamination. Moreover, contamination problems with Zinc-65 were minimal.

Remarks/Potential for dose limitation: This project is expected to provide additional new insight on the efficiency of the zinc injection process.

References: EPRI NP-6168, report.

Duration: from: 1987 to: 1989

Funding: N/A

Status: Completed

Last Update: April 1, 1992

PWR STEAM GENERATOR PRECONDITIONING STUDIES

Keywords: CONTAMINATION PREVENTION; PREFILMING; INCONNEL;

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Objectives: The aim of this project is to determine the effects of prefilming using hot moist air on the release of corrosion products from Inconnel steam generator tubing.

Comments: Loop tests will be undertaken to measure release from Inconel 600 and Inconnel 690. This work is linked to other EPRI studies and is directed at evaluating preconditioning methods that can be applied to replacement steam generators before installation.

Remarks/Potential for dose limitation:

References: None

Duration: from: 1987 to: 1989

Funding: N/A

Status: Completed

Last Update: April 1, 1992

THE TREATMENT OF RADIOACTIVE ION-EXCHANGE RESINS

Keywords: CONTAMINATION REMOVAL; ION EXCHANGE; RESIN; DECONTAMINATION; ORGANICS REMOVAL

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Objectives: To design and operate a pilot-scale system for demonstrating the resin wet oxidation process using hydrogen peroxide and to verify resin volume reduction and effective destruction of resin organic components, including chelates.

Comments: Using the process parameters and experience from earlier laboratory tests, the project team designed a pilot-scale system to perform wet (aqueous) resin oxidation for resin batch quantities of up to 5 cubic feet. Hydrogen peroxide was the oxidizing reagent. The resins were loaded with chemical species found in dilute chemical decontamination solutions to establish rates of reactions, chemical product, and volume reduction of the original resin used. The products of resin wet oxidation are primarily carbon dioxide and water vapor, free of radioactive species, which can be allowed to evaporate or be collected and recycled in the process. Instrumentation was designed to operate the pilot rig, to regulate the process, and to gain experience that will be useful in subsequent scale-up of the wet-oxidation process.

Remarks/Potential for dose limitation: Better than 99% destruction of organic material and minimum carry-over (decontamination ratios of 5000 and 10,000) were achieved. Difficulties encountered in operating the equipment were related to generation and carry-over of foam, which can be overcome in future undertakings by redesigning the equipment and choosing proper chemical foam suppressors. In operating the equipment, the project team made significant advances toward the development of a commercially viable process for oxidizing ion-exchange resins. These tests defined the key parameters necessary for larger-scale equipment design and operation. The simultaneous volume reduction and destruction of ion-exchange resins offer an attractive, cost-effective option for the disposal of radioactive wastes generated during reactor-system decontaminations.

References: "The Treatment of Radioactive Ion-Exchange Resins: Low-Temperature Resin Oxidation Process", EPRI NP-7368, Final Report, September 1991.

Duration: from: 1987 to: 1989

Funding: N/A

Status: Completed

Last Update: April 12, 1992

PWR CORROSION TESTS USING LOMI

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

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Objectives: The aim of this project is to qualify the LOMI decontamination process for full system decontamination of Westinghouse-designed PWRs with fuel removed.

Comments: A comprehensive corrosion test program with LOMI has been carried out in flow loops at the Westinghouse R&D Center using Westinghouse PWR materials and geometries. A similar program using the CAN-DEREM process is being funded by Con Edison. The results are being used in an engineering evaluation to be funded by other utilities. The project team conducted an extensive materials-testing program for subsystem decontamination. They addressed topics including applications parameters, materials performance, engineering evaluations of reactor coolant system (RCS) components, fluid system interaction, additional equipment requirements, radwaste issues, recontamination, radiologic issues, economics, equipment design, and licensing considerations. In the three-cycle exposure tests, more than 200 specimens of 39 materials in the primary coolant system were exposed in a loop at various velocities, simulating flow conditions that would be experienced in a plant decontamination using reactor coolant pumps.

Remarks/Potential for dose limitation: The main observations from the three-cycle AP/LOMI exposures were corrosion rates of metals, alloys, and ceramics that are typicaly less than would be observed after a few years of normal service.

As a result of the engineering evaluations, full RCS chemical decontaminations using either the AP/LOMI or AP/CAN-DEREM processes can be performed with a high degree of confidence and assurance of no significant impact on plant equipment. The safety evaluation concluded that either the AP/LOMI or AP/CAN-DEREM chemical decontamination processes can be applied as many as three times without adversely affecting the functional integrity of existing systems and mechanical equipment.

References: 1. "PWR Full-Reactor Coolant System Decontamination", EPRI NP-7512, Final Report, September 1991. 2. "The Qualification of Dilute Chemical Processes for PWR Full-Reactor Coolant System Decontamination", EPRI NP-7514, Final Report, November 1991.

Duration: from: 1988 to: 1990

Funding: N/A

Status: Completed

Last Update: April 12, 1992

CRUD TRANSPORT CHEMISTRY

Keywords: CONTAMINATION PREVENTION; CRUD; CRUD TRANSPORT; WATER CHEMISTRY

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Objectives: The aim of this project is to improve understanding of factors influencing crud formation, transport and deposition, including coolant and feedwater chemistry, and to prepare PWR primary water chemistry guidelines.

Comments: This is a research project involving both experimental studies and in-depth analysis of PWR and BWR data produced in other EPRI projects in the radiation control area. Earlier contracts in the CEGB project were directed at determining factors influencing cobalt deposition in PWR primary circuits (NWT project) and a review of work on BWR effects (another EPRI project).

Ongoing projects include an update of the Lindsay slurry-tank model of PWR crud transport, including a review of recent nickel ferrite solubility data. The model will be tested using the PACTOLE code. A review of the European work on PWR radiation control is being carried out. The first revision of the PWR primary water chemistry guidelines was issued in August 1988, with a partial update planned for the second half of 1988.

Remarks/Potential for dose limitation:

References: "PWR Primary Water Chemistry Guidelines: Revision 1", EPRI NP-5960 SR, August 1988.

Duration: from: 1984 to: 1989

Funding: N/A

Status: Completed

Last Update: April 1, 1992

QUALIFICATION OF COBALT-FREE HARDFACING ALLOYS FOR LWR

Keywords: CONTAMINATION PREVENTION; NOREM; VALVE; HARDFACING ALLOY

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Objectives: The aim of this project is to qualify promising cobalt-free hardfacing alloys, identified in laboratory tests for use in nuclear valves, by evaluating their performance in loop and field tests.

Comments: This project will evaluate the performance of three iron-based, galling wear resistant hardfacing alloys under prototypical conditions. The alloys to be evaluated are NOREM, the Stoody alloy EB5183, and the Thyssen alloy Everit 50. The alloys will be deposited on three inch gate valves and exposed in a loop facility designed to simulate LWR operating conditions. Valve performance and wear will be monitored and compared with that of a valve hardfaced with a standard cobalt-base alloy. The primary objective is to qualify new alloys for use in calves subjected to galling wear, because other work has identified a number of cobalt-free alloys for use in flow control valves.

Remarks/Potential for dose limitation: This project should play an important role in reducing radiation exposures by qualifying a number of cobalt-free substitutes for hardfacing alloys. The test valves were successfully hardfaced with the iron-base alloys using plasma transferred arc (PTA) technology. Initial valve testing, under PWR temperature, pressure, and chemistry, began this year.

References: None

Duration: from: 1987 to: 1990

Funding: N/A

Status: Completed

Last Update: April 1, 1992

PRODUCTION OF NOREM HARDFAC!NG ALLOYS

Keywords: CONTAMINATION PREVENTION; COMPONENT RELIABILITY; HARDFACING ALLOY; VALVE

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Objectives: The aim of this project is to obtain weld consumables (powder and solid weld wire) and cast products of the NOREM iron-base hardfacing alloys (developed under another EPRI project) using standard commercial practice. Minimize production costs of solid wire. License NOREM to hardfacing vendor(s).

Comments: This project will provide gas atomized powder for PTA applications to EPRI contractors and valve vendors. Develop procedures for producing solid weld wire that can be used by nuclear utilities and service groups for valve refurbishing operations. Evaluate attributes of parts consolidated by hot isostatic pressing. Make NOREM products available for evaluation by interested parties.

Anval Nyby and Stoody Deloro Stellite have produced gas atomized powder that is being used by AECL and U.S. valve vendors. Atek (as subcontractor to the EPRI NDE Center) has produced pilot quantities of solid weld wire. Latrobe Steel (as subcontractor to Anval Nyby Powder) has identified a regime where billets consolidated by hot isocratic pressing can likely be rolled into bar stock. If successful, this could substantially reduce costs of producing solid weld wire.

Remarks/Potential for dose limitation: Production of this hard facing alloy for industrial use will play an important role in reducing radiation exposure by making a cobalt-free substitute available for use in nuclear power plant components.

References: None

Duration: from: 1988 to: 1989

Funding: N/A

Status: Completed

Last Update: April 1, 1992

COBALT REPLACEMENT GUIDELINES

Keywords: CONTAMINATION PREVENTION; COBALT-FREE ALLOY; GUIDELINES; VALVE

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Objectives: The aim of this project is to develop guidelines that will assist nuclear utilities in introducing new cobalt-free hardfacing alloys into nuclear plant components, especially nuclear valves.

Comments: This project established a working group of utility representatives, nuclear steam supply system vendors, valve manufacturers, consultants, and EPRI staff to provide input and review of contractor's proposed guidelines. Different valve functions and operating conditions found in operating reactors were reviewed to identify those applications in which hardfacing alloys could be avoided or cobalt-free alloys could be used. The conditions under which cobalt-base hardfacing alloys or their equivalent should continue to be used were identified.

Remarks/Potential for dose limitation: These guidelines should simplify utility efforts to implement an aggressive cobalt reduction program to reduce radiation exposure. The results show that cobalt-base hardfacing alloys are not needed for a large number of plant applications. Specifically, hardfacing alloys are not needed in flow control valves or in the seats of check valves. Field data show that a number of stainless steels perform satisfactorily in these applications. A variety of nickel-base alloys are adequate for use in the pivot bushings of check valves and in globe or gate valves where contact stresses are less than 15 ksi. New iron-base hardfacing alloys exhibit wear resistance matching those of the cobalt-base alloys in laboratory tests. Utilities planning to refurbish valves during an outage or planning to order replacement valves should heed the recommendations of these guidelines. Cost savings will be realized from both reduced exposures and lower valve replacement costs.

References: "Cobalt Reduction Guidelines", EPRI NP-6737, Special Report, March 1990.

Duration: from: 1988 to: 1989

Funding: N/A

Status: Completed

Last Update: April 12, 1992

BWR COBALT DEPOSITION STUDIES

Keywords: CONTAMINATION PREVENTION; HYDROGEN WATER CHEMISTRY; PRECONDITIONING; ZINC INJECTION; IMPURITY REDUCTION; PALLADIUM COATING

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Objectives: The aim of this project is to identify the key factors that control cobalt deposition on stainless steel in BWR piping systems. The variables studied include water chemistry, water conductivity, pH, metallic and organic additives, and surface treatments.

Comments: Carefully controlled loop studies were performed in which cobalt-60 pickup measurements were made periodically on stainless steel specimens.

Experiments that address the effects of additives and water chemistry on cobalt-60 pickup have been completed. The results show that to minimize cobalt-60 deposition one should:

1. Reduce cobalt-60 concentrations in the water by replacing cobalt sources.
2. Reduce impurities in the reactor water and feedwater.
3. Maintain at least 5 ppb zinc in the reactor water.
4. Precondition new or replacement piping.

The remaining experiments are evaluating the effect of different methods of producing palladium coatings on the cobait-60 pickup.

Remarks/Potential for dose limitation: Earlier studies showed that an electrolysis layer of palladium was the most effective surface treatment to reduce cobalt-60 pickup.

References: C. C. Lin and F. R. Smith, "BWR Cobalt Deposition Studies: Final Report", EPRI NP-5808, May 1988.

Duration: from: 1983 to: 1990

Funding: N/A

Status: Completed

Last Update: April 1, 1992

RESEARCH REACTOR LOOP WATER CHEMISTRY STUDY

Keywords: CONTAMINATION PREVENTION; CORROSION PRODUCT TRANSPORT; CORROSION PRODUCT DEPOSITION; SURFACE CONDITIONING; WATER CHEMISTRY; PH; BORON

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Objectives: The aim of this project is to study the effects of water chemistry and surface condition on corrosion product transport and deposition in a reactor circuit. The initial series of PWR experiments will address the effects of pH and boron concentration on corrosion product deposition.

Comments: This is a joint project with ESEERCO, who is funding the loop construction which began in 1986. Loop operation, funded under this project began in late 1988. The initial commissioning run of the loop will simulate low pH PWR conditions.

Two loops, one simulating PWR conditions and the other simulating BWR conditions, will be constructed in the MIT research reactor.

Remarks/Potential for dose limitation:

References: EPRI TR-100156, December 1991.

Duration: from: 1986 to: 1990

Funding: N/A

Status: Completed

Last Update: April 1, 1992

RADIATION FIELD AND DOSE DATA ASSESSMENT

Keywords: CONTAMINATION PREVENT ION; OPERATIONAL AND MAINTENANCE TECHNIQUES; DOSE CORRELATION; RADIATION FIELD; SPECIAL MAINTENANCE; OUTAGE TIME; ALARA

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Objectives: The aim of this project is to correlate [illegible] dose data with plant specific parameters including outage time, special maintenance, and radiation fields, in order to establish the relative importance of general radiation fields and hotspots in determining worker doses. To establish trends in plant doses with the technologies identified below.

Comments: This project monitored BWR radiation fields, emphasizing plants that have improved water chemistry control, have replaced recirculation piping, decontaminated recirculation piping, and/or used preconditioning/passivation processes. Dose rate measurements were taken at 17 plants. The project started in April 1986. The lowest radiation fields have been associated with those new plants which operate with zinc in the feedwater. Two of these, Limerick and WNP-2, have naturally occurring zinc, and the third, Hope Creek, relies on zinc injections by the GEZIP process. Measurements were performed in 1989 at Limerick-1, Vermont Yankee, Clinton, River Bend, and Millstone-1.

During initial plant visits, cross calibration of plant instrumentation to program instruments was established using the plant calibration facility. Plant measurements were obtained and then incorporated into the database. In addition, chemical and radiochemical data have been collected at the sites and from the General Electric Fuel Warranty Operating Limits database.

Remarks/Potential for dose limitation: The main results of this project include the following:

- Replacement piping that has been electropolished and prefilmed shows lower dose rates than piping that was only electropolished.
- Zinc injection lowers dose rate buildup by about a factor of two.
- Nonzinc plants with forward-pumped heater drains have higher dose rates than plants with cascade drains.
- Recontamination trends at nonzinc and prefilmed plants are similar.
- Data from Dresden-2 indicates that hydrogen water chemistry has no effect on dose rate buildup, decontamination, or recontamination.
- Decontamination is an effective technique for periodically reducing dose rates.

This report is part of a broad program to identify and implement diverse radiation control measures developed by EPRI to reduce occupational radiation exposure of maintenance personnel at operating reactors.

References: "BWR Radiation-Field Assessment: 1986-1988", EPRI NP-6787, Interim Report, March 1990.

Duration: from: 1986 to: 1989

Funding: N/A

Status: Completed

Last Update: April 12, 1992

PASSIVATION AND SURFACE CONDITIONING

Keywords: CONTAMINATION PREVENTION; PASSIVATION; SURFACE CONDITIONING; PALLADIUM COATING; CORROSION; COBALT-60; ZINC INJECTION; PWR; BWR, ZINC ADDITION

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Objectives: The aim of this project is to qualify advanced passivation/preconditioning methods that reduce cobalt-60 pickup by using these methods on plant components whose performance will be monitored over a number of cycles; to run appropriate lab, loop and plant tests to insure that these methods will not adversely affect other plant components.

Comments: Metallic palladium and nickel films have been identified in loop experiments, under both simulated BWR and PWR coolant chemistries, as showing the greatest ability to resist cobalt-60 pickup. Small palladium-coated components have been inserted into operating reactors to confirm the loop tests. Loop tests will be rerun using a number of different palladium coating techniques to identify those best suited for reactor applications. Experiments at AECL are investigating whether zinc additions will reduce cobalt-60 pickup under simulated PWR conditions. Loop tests suggest the Zn injection under PWR conditions has no effect on corrosion product release from Inconel 600 but reduces cobalt-60 pickup on stainless steel.

Palladium coated sheet specimens have been exposed on the steam generator inserts of the Belgian Doel PWR, and a coated reactor water sample cooler has been inserted in the Quad Cities-2 BWR. The initial measurements at both sites are not encouraging, and studies will be initiated to elucidate the differences between measured valves in loop studies and in the field.

Remarks/Potential for dose limitation: Zinc injection passivation has been demonstrated to significantly reduce radiation fields at BWRs. One interesting outcome of this project will be to demonstrate the efficacy of the technique for PWRs.

References: H. Ocken, "Surface Treatments to Reduce Radiation Fields: Test Loop Studies and Plant Demonstrations", EPRI NP-5209-SR, April 1988.

Duration: from: 1984 to: 1990

Funding: N/A

Status: Completed

Last Update: April 1, 1992

FEEDWATER FLOW ELEMENT IMPROVEMENT

Keywords: CONTAMINATION REMOVAL; OPERATIONAL AND MAINTENANCE TECHNIQUES; CORROSION PRODUCT; SURFACE TREATMENT

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Objectives: The aim of this project is to minimize losses of power output that result from the deposition of corrosion products on the venturis that are used to measure feedwater flow.

Comments: This project will help to:

1. Identify coatings or other surface treatments that will minimize deposition of corrosion products on venturis.
2. Develop on-line techniques for removing fouling.
3. Qualify technology for accurately measuring flow rate in the presence of fouling.
4. Develop methods for measuring feedwater flow that do not rely on delta-P measurements.

Remarks/Potential for dose limitation: Reduction of manual intervention by the use of on-line techniques for removal of corrosion products should reduce occupational exposure.

References: None

Duration: from: to:

Funding: N/A

Status: Proposed

Last Update: April 1, 1992

COOLANT CHEMISTRY AND RADIOLYSIS IN BOILING REACTOR COOLANT

Keywords: COMPONENT RELIABILITY; WATER CHEMISTRY; RADIOLYSIS; BWR

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Objectives: The aim of this project is to determine just how much hydrogen is needed to suppress radiolytic oxygen and hydrogen peroxide in a boiling LWR core.

Comments: This project will use an in-core loop with fast-flow sampling lines designed to minimize sample line alterations of the hydrogen peroxide, oxygen and hydrogen contents of the water. The core inlet concentration of hydrogen will be varied from zero to a value at which all oxygen and peroxide disappear. Both PWR and BWR conditions would be established in the loop water in separate experiments.

The costs for this project are shared by AECL. Tests were completed in October 1988 and results were published in 1989.

Remarks/Potential for dose limitation:

References: None

Duration: from: 1986 to: 1989

Funding: N/A

Status: Completed

Last Update: April 1, 1992

ON-LINE MONITORING TECHNIQUES FOR REDOX POTENTIAL, HYDROGEN CONCENTRATION, AND PH IN NUCLEAR REACTOR COOLANT CIRCUITS

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; CONTAMINATION PREVENTION; LWR; ON-LINE MONITORING; HYDROGEN CONCENTRATION; PH; REDOX POTENTIAL

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Objectives: The aim of this project is to develop continuous, on-line, long-lived, reliable monitoring instruments to detect corrosive conditions in LWR water circuits at full operating temperatures.

Comments: A prototype probe for disso' ved hydrogen has been prepared for installation in the primary circuit at one of the Turkey Point PWRs. Two candidate probes for measuring pH at the same installation are undergoing laboratory tests. The weak link in reliability of the pH probes is the reference electrode, (current models tend to lose their internal electrolyte during month-long periods of use). The reference electrode is also the key to a reliable measure of the electromechanical potential (ECP) of a corroding material, a measurement of great importance to monitoring BWR circuits to avoid intergranular stress corrosion cracking (IGSCC) in the recirculation piping. The current approach toward improving the reference electrod [illegible] volv. the KCl internal electrolyte by the use of a polymer. Long-term tests of the polymer were completed in 1988. Delays at Turkey Point have caused the hydrogen meter to be operated in a loop at AECL where it was found to be reliably operative with a time constant of 5 minutes. It was also found that the alumina spindle on which the palladium wire sensor is wound dissolves slowly and will be replaced with one of zirconia.

Remarks/Potential for dose limitation: The development of on-line monitoring instruments to detect corrosive conditions in LWR water circuits will help to improve the reliability of reactor components greatly. This project should therefore play an important role in reducing occupational exposure resulting from maintenance and replacement work.

References: D. D. MacDonald, A. C. Scott, P. Wentrcek and M. C. H. McKubre, "Monitoring Techniques for pH, Hydrogen, and Redox Potential for Nuclear Reactor Circuits", EPRI Interim Report NP-2806, January 1983.

Duration: from: 1978 to: 1989

Funding: N/A

Status: Completed

Last Update: April 1, 1992

IN-PLANT SYSTEM FOR CONTINUOUS LOW-LEVEL ION MEASUREMENT IN STEAM-PRODUCING WATER

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; ON-LINE MONITORING; CORROSION; BWR

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Objectives: The aim of this project is to adapt a commercially available laboratory ion-chromatograph for semicontinuous measurements of corrosive species in liquid water in steam electric power plants. Of primary interest are the ions: chloride, sulfate, nitrate, phosphate, borate, chromate, oxalate acetate, formate, sodium, fluoride and ammonium- at the < 1ppb to several ppm concentration levels. Other ions can be measured also, but the study will be limited to those of concern in reducing corrosion in steam power fluid piping, heat exchangers, and turbines. On-line measuring instruments for hydrogen peroxide and the sulfate ion have been added tasks in 1987-1988.

Comments: The following results have been shown: 1) sulfate ions are a prime impurity in plants with polishers, 2) sulfate, once injected, is removed with difficulty because of slow kinetics that get even slower as anion resins age, 3) sulfate moves about the PWR secondary system and the BWR as a species far more volatile than sodium sulfate - probably as sulfuric acid, 4) early warning on very small condensor leaks can be followed well below the limit seen by cation conductivity meters, 5) leakage of sulfonated cation resin is a prime source of sulfate in one powdered resin demineralizer system, occurring most dramatically as a new precoat is put on-line, 6) corrosion product copper, nickel, and chromate ions can be followed in reactor water (LaSalle) nitrate from dissolved nitrogen gas entering with air at the condenser. Also, nitrate from anion resin leakage has been observed at two BWRs. The nitrate disappears when hydrogen is injected in hydrogen feedwater chemistry mini-tests.

Remarks/Potential for dose limitation: The semi-continuous and on-line monitoring of corrosive species in steam power fluid piping should improve piping reliability. This project should help reduce radiation exposures at BWR plants.

References: J. L. Simpson, M N. Robles, and T. O. Passell, "In-Plant System for Continuous Low-Level Ion Measurement in Steam-Producing Water", Proceedings of ASTM Symposium on Power Plant Instrumentation for Measurement of High Purity Water Quality, Milwaukee, WI, June 9 and 10, 1980, ASTM Publication STP-742, 1981.

Duration: from: 1979 to: 1989

Funding: N/A

Status: Completed

Last Update: April 12, 1992

RESIN SEPARABILITY TO IMPROVE POLISHING UNDER MORPHOLINE AVT

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; RESIN

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Objectives: The aim of this project is to measure the degree of separation of cation from anion resin required to give acceptably low sodium leakage in the mixed bed made up from regenerated resins in the laboratory.

Comments: A resin separation process based upon a vibrating filter is to be applied, followed by a carefully controlled hydraulic separation column to obtain several degrees of separation of the mixed resin, from the levels usually achieved in practice to the levels achievable by the advanced separation method. Subsequently, mixed beds will be made up with these resins after regeneration and the sodium leakage will be measured as the bed passes through morpholine (and other amines) breakthrough.

Remarks/Potential for dose limitation:

References: None

Duration: from: 1988 to: 1989

Funding: N/A

Status: Completed

Last Update: April 12, 1992

RADIATION FIELD TRENDS IN WESTINGHOUSE-DESIGNED PLANTS

Keywords: CONTAMINATION PREVENTION; RADIATION FIELD; DOSE RATE; RADIATION DOSE; RADIATION FIELD TREND; PRIMARY COOLANT CHEMISTRY; CHEMISTRY; COBALT REDUCTION; COBALT; STEAM GENERATOR

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Objectives: To observe collective dose and dose rate trends at Westinghouse PWR plants and, through these observations, evaluate the relative significance of the factors that affect plant dose rates and plant collective doses.

The principal factors considered were:

1. Primary Coolant Chemistry, including effect of non-coordinated, coordinated, modified, and elevated pH.
2. Variations in Cobalt Input, including inconel versus zircalloy fuel grids and use of low cobalt steam generator tubing.
3. Physical Factors, such as steam generator replacement, channel head decontamination and channel head surface treatment.

Comments: Sixty-six Westinghouse plants were a part of this study, including forty-seven in the U.S. 1. The data showed that the dose rates peaked after 6 EFPY of operation and the average channel head dose rate at the peak was 13 R/hr. 2. The effect of zircaloy (compared to inconel) grids was quite significant; utilization of low cobalt steam generator tubing and decontamination of the steam generator channel heads also had a marked effect on dose rates. 3. Although dose rates at plants using elevated pH were lower than the dose rates at other comparable Westinghouse plants, the results were less significant; in particular, it was difficult to see any effect on dose rates in moving from a pH of 7.2 to 7.4.

Remarks/Potential for dose limitation: Main conclusions regarding factors affecting dose rates were:

1. Use of low cobalt tubing and zircaloy grids result in lower dose rates compared to normal cobalt tubing and inconel grids.
2. Coordinated coolant chemistry results in lower dose rates than non-coordinated coolant chemistry.
3. Modified or elevated pH generally results in a reduction in the rate of radiation level buildup and/or lower dose rates than coordinated coolant chemistry.

Conclusions regarding the trends of plant doses were:

1. There was no apparent trend by refuelling cycle when plants were divided into four categories according to dose rates and amount of maintenance required.
2. Assuming a full system decontamination every ten years, 2800 to 6800 man-rem can be saved depending upon the maintenance-dose rate category of the plant.

References: Bergmann, C.A., T.M. Lloyd and J.D. Peroch, "Radiation Field Trends in Westinghouse-Designed Plants", Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: N/A

Status: Completed

Last Update: June 27, 1991

OXYGEN TRANSPORT IN BWR CYCLES

Keywords: COMPONENT RELIABILITY; OPERATIONAL AND MAINTENANCE TECHNIQUES; OXYGEN CONTENT; OXYGEN TRANSPORT; BWR

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Objectives: The aim of this project is to design software capable of estimating the dissolved oxygen content of water at key points in a BWR steam/coolant cycle for use in predicting erosion/corrosion of the piping.

Comments: In this project, software models previously used to calculate amine distribution in PWR secondary cycles will be used to compare computed oxygen levels with measured oxygen levels in BWRs. This software is to be incorporated in the CHECMATE code modified for BWR two phase flow regions (the wet steam piping).

A working draft of the software has been demonstrated at a recent CHEC users meeting. However, some uncertainty exists regarding the effect from oxygen of certain cycle components. This will need to be resolved before satisfactory agreement with available plant measurements is achieved.

Remarks/Potential for dose limitation: Improved techniques for predicting erosion/corrosion will increase piping reliability and should result in less exposure from maintenance and replacement.

References: None

Duration: from: 1988 to: 1989

Funding: N/A

Status: Completed

Last Update: April 12, 1992

REMOTE REPAIR TECHNIQUE FOR MSIVS

Keywords: REMOTE SYSTEM; MSIV; STEAM ISOLATION VALVE; REMOTE REPAIR; REMOTE WELDING

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Objectives: The aim of this project is to develop a process to refurbish the seats and guides of the main steam isolation valves used in nuclear power plants, using a remote controlled welding technique.

Comments: Orbital pipe welding machines with remote operation capabilities were evaluated for use to deposit hardfacing on the seat and guide ribs. The intent was to use these machines while employing the gas tungsten arc welding (GTAW) process to qualify welding procedure specifications in conformance with the American Society of Engineers Boiler and Pressure Vessel Code, Section IX requirements. After preliminary trials with standard boring and facing portable equipment, it became obvious that very specialized equipment would be required to produce the quality of machining anticipated. The machining requirements of MSIV refurbishing were met by: 1) an in-depth check of all parameters of the valve body affected by the machine which will perform the cutting operation, 2) use of a machine which is easily set up and easily aligned with maximum rigidity, and 3) use of a machine cutting operation which includes single point tooling for line boring and rough weld deposit.

Remarks/Potential for dose limitation: Main steam isolation valves (MSIVs) in BWRs are subjected to seat leakage tests to satisfy stringent technical specification requirements. During testing, a lapping operation performed on the seating surface is sometimes required to achieve a satisfactory leak rate. However, each lapping operation removes some of the hardfacing material which has been applied to the valve seat by welding. Whenever the operation exposes base material, it becomes necessary to restore the seat to its original condition.

This technique minimizes the exposure of welding personnel and related maintenance workers in refurbishing main steam isolation valves. It also limits the number of craftsmen required to do the job.

References: J. F. Plantz, "GE Perfects Remote Repair Technique for MSIVs", Nuclear Engineering International, September 1988, Vol. 33, No. 410, p. 46.

Duration: from: 1988 to: 1989

Funding: N/A

Status: Completed

Last Update: April 12, 1992

INTELLITORQUE : A SYSTEM FOR MONITORING ROOT CAUSE OF MOV MALFUNCTIONS

Keywords: REMOTE SYSTEM; MOV; MOTOR OPERATED VALVE; MONITORING; VALVE

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Objectives: The aim of this project is to identify the major causes of valve malfunctions and provide a means of effective corrective action.

Comments: The Intellitorque control unit can be located at the valve, at the motor control center of the valve operator, or in the main control room. The system uses a continuous input of information concerning valve stem position, valve stem thrust, and motor load (which is a function of motor voltage, motor current, and phase angle). The system electronically compares these input data with pre-adjusted set points to control the actuation of the valve. Adjustment of these set points is relatively simple and is made at the control unit. The control unit provides a visual display of valve stem thrust (tension or compression), valve stem position, valve stroking time, and the open and closed trip functions of position, thrust, and motor load. The system has successfully completed a comprehensive laboratory test program. Long term field tests have begun at two nuclear power stations: two units have been installed at Point Beach station (Wisconsin Electric Power Co), and two units will be installed in Comanche Peak station (Texas Utilities Electric).

Remarks/Potential for dose limitation: The failure to operate or the improper operation of motor operated valves has resulted in: 1) loss of plant availability 2) large expenditure of maintenance resources, 3) personnel radiation exposure, and 4) adverse impact on plant safety. The Intellitorque system is expected to reduce personnel exposures, increase plant availability and safety, and lessen the requirements for maintenance.

References: "Improvement in Motor Operated Valves", EPRI-NP-4254, November 1985.

Duration: from: to:

Funding: N/A

Status: Completed

Last Update: April 12, 1992

USING ULTRASONICS TO AVOID CHECK VALVE DISASSEMBLY

Keywords: COMPONENT RELIABILITY; ULTRASONIC; VALVE; MONITORING; REMOTE MONITORING

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Objectives: The aim of this project is to provide a diagnostic system, based on ultrasonics, to test valves for instability.

Comments: The CHECKMATE system monitors check valve disc position with ultrasonics. The ultrasonic beam travels from a sending transducer through the valve body and is reflected back to the receiving transducer. The valve disc is located by adjusting the transducer angles to focus the beam. Once the disc is located, the sound path travel distance is determined by measuring the time between transmitting a pulse and receiving it. The sound path distance is then combined with the transducer angles and the distance to the valve hinge pin to compute the disc position.

In previous testing, only a small percentage of values, which would have been anticipated to be unstable using design guidelines, have shown any signs of actual instability. If most of the check valves identified as potentially unstable using the design review approach are, in fact, stable, it is reasonable to assume that some check valves which are thought to be stable are not stable. The checkmate system can provide a reliable answer to the stability question.

Remarks/Potential for dose limitation: While check valve failures have become a significant concern in the nuclear industry, the actual percentage of valves that have failed (other than seal leakage) has been small. Using a diagnostic system based on ultrasonics, it is possible to test valves for instability, thereby avoiding extensive design review efforts, unnecessary dismantling, and occupational exposure.

References: 1) D. M. Ciesielski, "Using Ultrasonics to Avoid Check Valve Disassembly", Nuclear Engineering International, September 1988, Vol. 33, No. 410, pp. 42-44. 2) Electric Power Research Institute, "Application Guidelines for Check Valves in Nuclear Power Plants", EPRI NP-5479. 3) W. J. Raymeyer, "Application of Check Valves with Unsteady and Non-uniform Flow Conditions", Utah State University.

Duration: from: 1988 to: 1990

Funding: N/A

Status: Completed

Last Update: April 12, 1992

A "WET MOTOR" SEALLESS PUMP FOR REACTOR WATER CLEAN UP SYSTEM IN BWRS

Keywords: COMPONENT RELIABILITY; SEALLESS PUMP; BWR; RWCU; WET MOTOR PUMP

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Objectives: The aim of this project is to investigate the use of "wet motor" sealless pumps installed in reactor water clean up (RWCU) pumps in BWRs.

Comments: The first wet motor sealless pump in BWR RWCU service was installed at Hatch 2 in March 1987 and has been operating satisfactorily. It was guaranteed for five years of operation without maintenance. Another similar pump was installed at Hatch 1 and two more were fitted in Hanford 2 in 1988.

The sealless pump has many advantages: 1) it eliminates the need for shaft seals and has the capability to operate without any maintenance for five years or more; 2) it has the capability of more than 100 per cent of typical clean-up system flow - 480 gal/min at 525 ft differential head (which makes parallel pump operation unnecessary); and 3) it provides low occupational exposures.

Remarks/Potential for dose limitation: A number of problems have been experienced with pump seals in the reactor water clean up systems of BWRs with seal lives, in some cases, of as little as two weeks, or even less. Seal replacement can entail high costs and incur significant occupational exposures.

However, a particularly promising solution which has the advantage of being backfittable to existing plants is replacement of the original equipment with pumps of the "wet motor" sealless variety. The wet motor sealless pump has an excellent operating history and is well suited to the arduous duty experienced in RWCU applications. It reduces occupational exposures since it is more reliable and is replaced less frequently.

References: P. Kachel, "How the 'Wet Motor' Can Help Eliminate RWCU Leaks", Nuclear Engineering International, September 1988, Vol. 33, No. 410, pp. 51-52.

Duration: from: 1988 to: 1990

Funding: N/A

Status: Completed

Last Update: April 12, 1992

A ROTATING UT SYSTEM FOR INSPECTION OF STEAM GENERATOR TUBES

Keywords: REMOTE SYSTEM; ULTRASONIC TESTING; UT; STEAM GENERATOR; STEAM GENERATOR TUBE; REMOTE INSPECTION; NON-DESTRUCTIVE EVALUATION

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Objectives: The aim of this project is to develop a field deployable rotating ultrasonic system capable of inspecting steam generator tubes.

Comments: The key features of the UT-360 system include:

1. the option to mount up to three transducers in the head: one to measure wall thickness, one to check for axial cracks, and one to check for circumferential cracks.
2. a detachable head which permits specialized transducer configurations to be fabricated and tested without completely rebuilding the probe.
3. an interchangeable body.
4. axial and circumferential position encoding which allows for the UT response to be mapped accurately against the tube dimensions.
5. advanced display capabilities which includes A-scan, A-scan water falls, B- and C-scans, and terrain maps of time or peak amplitude responses.

In its first field application, the UT-360 system was used to evaluate eddy current indications at a nuclear site. The B- and C-color scans confirmed the eddy current indications to be wall thinning.

Remarks/Potential for dose limitation: The UT system is designed to provide additional non-destructive evaluation data to evaluate tube health. Possible dose savings may include: eliminating the need for a tube pull, detecting and characterizing ECT flaw signals to justify plugging tubes, and quality control for sleeving activity. In general, the system has worked well during field deployment and several lessons were learned to improve the performance for the next inspection. Since the first inspection, advancements have been made in signal quality using integral electronics, integration of eddy current coils, improved water delivery system and adaptation of single element high resolution transducers. The system has been used at two additional sites to confirm permeability variations, wall thinning and intergranular cracks. It has also been adapted for sleeve bond inspections and for OTSG inspections from the upper head. Continued development will include improving data interpretation capability, and enhancing speed of inspections.

References: 1) S. W. Glass, S. W. Shackelford, "Inspecting Steam Generator Tubes with a Rotating UT System", Nuclear Engineering International, April 1989, Vol. 34, No. 417, pp 41-42. 2) S.W. Glass and S.W. Shackelford, "Development of a Rotating UT Inspection System", 7th Annual EPRI Steam Generator Workshop, June 1988. 3) Rotating Ultrasonic Inspection System Development, Babcock & Wilcox, Empire State Electric Energy Research Corporation (ESEERCO), Report EP86-4, March 1988.

Duration: from: 1986 to: 1989

Funding: $500,000

Status: Completed

Last Update: April 12, 1992

THE ALOK 3 ULTRASONIC INSPECTION SYSTEM

Keywords: REMOTE SYSTEM; COMPONENT RELIABILITY; ULTRASONIC INSPECTION; ALOK 3; INSPECTION

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Objectives: The aim of this project is to speed up inspection time, obtain more reliable results, and document the results in a more efficient manner.

Comments: The ALOK 3 system can be used for in-service inspection of the lower sections of reactor pressure vessels and, with different manipulators, for primary system components. Testing is speeded up by the use of the new ZMM3 and ZMM4 central mast manipulators. They reduce the time required to change a shell examination system to about half an hour. Now, setting up a nozzle system takes no more than three hours. In May 1988, the ultrasonic examination of a reactor pressure vessel lower section in the Grohnde nuclear power plant was completed in less than seven days.

The unit simplifies the examination of areas in which geometrical or cladding echoes have had a distorting effect in the past. Such tasks cannot be performed by the previously used time-gate-guided systems, at least not as routine operations.

Remarks/Potential for dose limitation: Siemens/KWU has recently been working on a Phased Array Ultrasonic System using tomographic processing. This system has just completed its first use in the field. It is designed to improve defect detection and discrimination as well as sizing, and will be used in conjunction with the ALOK system to inspect components with complex geometries.

The increased reliability of the results will result in reduced occupational exposure from maintenance.

References: Humeuller, R., and Rathgeb, W., "Alok 3 Offers Faster Ultrasonic Inspection", Nuclear Engineering International, April 1989, Vol. 34, No. 417, pp. 46-47.

Duration: from: 1989 to: 1991

Funding: N/A

Status: Completed

Last Update: April 12, 1992

ACOUSTIC LEAK MONITORING IN JAPAN

Keywords: REMOTE SYSTEM; COMPONENT RELIABILITY; LEAK MONITORING; BWR; SAFETY RELIEF VALVE

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Objectives: The aim of this project is to examine the use of acoustic leak monitoring as a means of estimating leak rate and detecting BWR safety relief valve opening, without affecting operating conditions.

Comments: Acoustic emission (AE) sensors and accelerometers have been adopted to improve the accuracy and speed of response in leak detectors and to estimate leak rate and detect valve openings in boiling water reactors. An experiment simulating actual conditions had shown this detection method to be effective.

Following the experimental tests, acoustic monitoring was tested on-site at Hamaoka 3 (1100MWe). Two AE sensors and six accelerometers were attached to the safety relief valves (SRVS) and connected to the microcomputer monitoring system installed in the control room when the plant was under construction. Acoustic signals were measured during plant start-up and during the first operating cycle. Under valve opening conditions, both acoustic and thermocouple signals were high. Upon closing, the acoustic signals returned to normal levels immediately, allowing the operator to determine whether a leak had occurred. However, the thermocouple signals only decreased very gradually due to the large thermal capacity of the insulated steampipes. This meant that it took a few hours to confirm a leak.

Remarks/Potential for dose limitation: Such acoustic leak monitoring techniques should increase the reliability of relief valves and result in reduced occupational exposure from maintenance.

References: Sano, K., K. Koyama, Y. Matsunaga, A. Kojuma and K. Sakai, "Acoustic Monitoring of BWR Main Steam Safety Relief Valves", ANS Meeting, Washington, DC, October 31 - November 4, 1988.

Duration: from: 1989 to: 1990

Funding: N/A

Status: Completed

Last Update: April 12, 1992

USE OF VIBRATION MONITORING TO ASSESS REACTOR COOLANT PUMP INTEGRITY

Keywords: REMOTE SYSTEM; RCP; VIBRATION MONITORING; REACTOR COOLANT PUMP; PUMP

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Objectives: The aim of this project is to confirm the integrity of reactor coolant pump shafts at St. Lucie through vibration monitoring.

Comments: The vibration instrumentation comprises: x and y proximity probes mounted in the motor upper bearing oil reservoir to observe the motor shaft; a proximity probe mounted on the motor tophat to observe the end of the motor shaft and to display the axial movement of the motor shaft; two more probes (x and y) mounted just outboard of the lower motor shaft; and an additional proximity probe mounted just outboard of the lower motor bearing which provides a phase reference for all the vibration signals by observing a machined hole in the motor shaft. Two additional temporary probes were mounted just above the pump mechanical seal to monitor the pump shaft.

The signals from these probes are transmitted to a monitor in the control room, where vibration data are recorded on an FM tape recorder and processed by a digital vector filter, spectrum analyzer and computer. The monitors provide a visual indication of the amplitude of vibration and are also connected to alert and alarm circuits. The signals can be trended manually or by computer to forewarn of impending problems.

Remarks/Potential for dose limitation: Additional vibration monitoring hardware and software that automatically sample, document, plot trends, and alert operators to potential problems are being developed. The resulting increase in reactor coolant pump system reliability should have a beneficial effect on radiation exposure.

References: Sculthorpe, B., and Johnson, K., "Using Vibration to Assess Reactor Coolant Pump Integrity", Nuclear Engineering International, April 1989, Vol. 34, No. 417, pp. 59-61.

Duration: from: 1989 to: 1991

Funding: N/A

Status: Completed

Last Update: April 12, 1992

IMPROVED TEST METHODS FOR PLANT PROTECTIVE COATING

Keywords: CONTAMINATION PREVENTION; CONTAMINATION REMOVAL; DECONTAMINABLE COATINGS; DECOMMISSIONING; LIFE-EXTENSION; MAINTENANCE COST; ALARA; DECONTAMINATION

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Objectives: This project will provide the technical basis for utility guidance in testing, selecting, and maintaining, improved decontaminable coatings.

Comments: Experience at TMI-2 and other nuclear facilities indicate that design basis accident (DBA) qualified coatings do not provide non-destructive decontaminablilty. Currently, these coatings are safety-class components and are subjected to costly requirements. The primary reasons for poor coating performance are that the radionuclide absorption mechanisms are virtually unknown, aging and wear effects are not quantified, and the current DBA and decontaminability tests do not adequately simulate actual plant environments.

This project, with guidance from ASTM D-33 Committee on Coatings for the Power Generation Industry, will provide improved methods to evaluate/rate coating systems for containment and provide utility guidance on coating selection/procurement/maintenance.

Remarks/Potential for dose limitation: The benefits of improving the decontaminability of these coating systems are likely to be lower plant life-extension or decommissioning costs, reduced plant ALARA activities, and reduced maintenance costs.

References: Flaherty, J.J., and N.S. Hirota, "Proceedings of Coatings Research Workshop", March 25-26,1986, EPRI MEAC, Charlotte, N.C. (Available from EPRI project manager).

Duration: from: 1987 to: 1989

Funding: N/A

Status: Completed

Last Update: April 12, 1992

AUTOMATED CONTROL ROD DRIVE BOLTING WRENCH SYSTEM TO SUPPORT BOILING WATER REACTOR MAINTENANCE

Keywords: REMOTE SYSTEM; CONTROL ROD DRIVE; BWR; MAINTENANCE; AUTOMATION; REMOTE MAINTENANCE; BOLT

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Objectives: To develop a system which allows a single mechanic to completely remove or replace control rod drive bolts from beneath the reactor vessel without disturbing the control rod drive radiation shielding.

Comments: The system is designed to support boiling water reactor maintenance. Each bolt can be removed or replaced within 30 seconds on a continuous basis; total tool weight is only 20 pounds. A remote torque display allows direct monitoring of peak torque from a location outside the reactor containment, thereby reducing radiation exposure to quality control personnel. The system consists of two complete wrenches, power supplies, and remote digital displays of peak torque. One wrench is dedicated to breaking-out bolts, and one is dedicated to making-up bolts. Peak torque is delivered to the bolt by an internal hydraulic system capable of repeatable performance within 2% of prescribed torque limits. The bolting technology employed in this design can be adapted to other difficult bolting jobs, such as reactor coolant pump motors and heat exchanges.

Remarks/Potential for dose limitation: The system will reduce both radiation exposures and direct labor requirements. In addition, it offers the potential for increased quality control, while reducing critical path time and operator fatigue.

References: None

Duration: from: 1986 to: 1989

Funding: N/A

Status: Completed

Last Update: April 12, 1992

CANADA R-285

MEASUREMENT OF OXIDE FILM RELEASED AS PARTICLES DURING THE CAN-DEREM DECONTAMINATION PROCESS

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; CAN-DECON; CAN-DEREM; CHELATES

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Objectives: Measure the amount of oxide film released as particles during a CAN-DECON/CAN-DEREM process. Determine mass and concentration of particles and their effects on critical mechanical components.

Comments: Process removes contamination from reactor systems and equipment, reducing dose levels to workers doing maintenance and/or modifications.

Remarks/Potential for dose limitation: High potendal for dose limitation.

References: None

Duration: from: 1989 to: 1989

Funding: $ 80,000

Status: Completed

Last Update: April 16, 1992

U.S.A. R-286

"WET MOTOR" SEALLESS PUMP FOR REACTOR WATER CLEAN UP SYSTEM IN BWRS

Keywords: COMPONENT RELIABILITY; REACTOR WATER CLEAN UP; PUMP; SEALLESS WET MOTOR; RWCU; WET MOTOR; BWR

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Objectives: Reduce frequent seal replacement maintenance and improve system reliability by eliminating the two 50% capacity mechanical seal pumps initially installed in the RWCU System and replacing them with 100% capacity sealless pumps capable of withstanding system thermal transients.

Comments: After correction of initial design weakness in shaft axial bearing design, the pumps installed appear to be meeting the project objectives.

Remarks/Potential for dose limitation: On initial pumps, 1.5 to 3.5 person-rem were expended every six months for seal replacement maintenance, with curtailed reactor operations while system was not fully operational. New pumps require only annual inspection and minimal maintenance. ISI cycle inspection resulted in 0.24 person-rem, but was more intensive than anticipated in the future.

References: None

Duration: from: 1986 to: 1989

Funding: $1,600,000

Status: Completed

Last Update: February 21, 1992

REACTOR WATER CLEAN-UP SEALLESS PUMP

Keywords: COMPONENT RELIABILITY; PUMP; RWCU; SEALLESS PUMP; MAINTENANCE; RELIABILITY; EXPOSURE TIME REDUCTION

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Objectives: Provide RWCU pump not prone to mechanical seal failures; reduction in maintenance costs/dose received.

Comments: Original RWCU pumps achieved about an average 3-6 month service-life before rebuilding or replacement. New sealless pumps have been in service since the spring of 1987 (unit 2) and the winter of 1989 (unit 1) with no failures.

Remarks/Potential for dose limitation: The Ingersoll rand Horizontal (old RWCU pump) required unscheduled maintenance during the years 1985 and 1986 that accumulated approximately 20 man-rem. The Hayward Tyler Sealless (new RWCU) pump has required no unscheduled maintenance since installation in 1987. This has been a savings of $100,000 over a two year period with ALARA cost benefit analysis of $5000/man-rem.

References: None

Duration: from: 1986 to: 1989

Funding: Capital Improvement

Status: Complete

Last Update: November 13, 1991

EXPOSURE REDUCTION MEASURES IN THE DESIGN OF SIEMENS/KWU PWR PLANTS

Keywords: CONTAMINATION PREVENTION; COBALT; CHEMISTRY; ROBOTICS; HARDFACING ALLOY; COBALT REMOVAL; PWR; MAIN COOLANT PUMP; REACTOR PRESSURE VESSEL; VALVE

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Objectives: To monitor the radiation dose rates and occupational exposures at Siemens/KWU plants, correlate them with plant features, and apply this knowledge to plant operations and subsequent plant designs.

Comments: A methodical approach was adopted to reducing occupational exposures at KWU PWRs. The success of this approach is seen from the fact that annual collective doses per reactor have come down in successive generations of plants from about 500 person-rem to less than 50 person-rem.

The most successful dose reduction measures, in order of importance, were found to be: 1) Introduction of cobalt replacement materials, 2) Application of the "modified" B/Li chemistry, 3) Decontamination, 4) Job Management, 5) Use of robotics, and 6) Shielding.

The program for replacing cobalt was started with materials development for RPV internals in 1981. By 1988, all components in the primary circuit were replaced. In parallel with cobalt replacement, fuel assemblies with zircalloy guide tubes and spacer were developed.

Remarks/Potential for dose limitation: Introduction of these measures has reduced dose rates and exposures in the newer plants by one order of magnitude. The newer plants easily meet the ICRP recommendations of average annual dose to individuals of less than 2 rem per year.

Backfitting activities are expected to reduce exposures in older plants by 50%. Full system decontamination is a further tool which will aid in achieving this objective.

References: Riess, R., and T. Marchl, "Exposure Trends at Siemens/KWU Plants," Proceedings, EPRI Seminar on Radiation Field Control, Palo Alto, California, April 9-11, 1991, Editors C.J. Wood and H. Ocken, P.O. Box 10412, Palo Alto, CA 94303.

Duration: from: 1980 to: 1991

Funding: N/A

Status: Completed

Last Update: April 23, 1991

FULL SYSTEM DECONTAMINATION OF THE BR-3 PWR PLANT

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

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Objectives: To carry out the full-system decontamination of the BR-3 PWR plant, with the fuel removed, by means of the Chemical Oxidizing Reducing Decontamination (CORD) process.

Comments: Siemens/KWU has an ongoing program to carry out full system decontamination on a number of reactors using the CORD process. BR-3 is a prototype Westinghouse PWR which is the first European reactor to be decommissioned. Although the efficacy of the process has been proven in a number of partial decontaminations, including 200 main coolant pumps, 4 regenerative heat exchangers and 9 subsystem decontaminations, BR-3 is the first reactor on which a full system decontamination will be carried out with the process.

Potential for dose limitation: The reasons why the CORD process was selected were:

- It has a high oxidation potential.
- A lower chemical concentration is required.
- The treatment time is shorter.
- It provides higher decontamination factors.
- It produces lower waste volumes.
- No intermediate cleanup is required.
- No flushing is required.
- No cooling or heating up of the solutions is required during one cycle.
- Continuous operation is possible.

References: Wille, H., R. Riess, and F. Motte, "Full-system Decontamination of the BR-3 Plant," Proceedings, EPRI Seminar on Radiation Field Control, Palo Alto, California, April 9-11, 1991, Editors C.J. Wood and H. Ocken, P.O. Box 10412, Palo Alto, CA 94303.

Duration: from: 1980 to: 1991

Funding: N/A

Status: Completed

Last Update: April 1, 1992

U.S.A. R-290

MITIGATION OF THE IMPACT OF REDUCED RADIATION EXPOSURE LIMITS ON NUCLEAR POWER PLANT OPERATIONS

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUE; CONTAMINATION PREVENTION; COMPONENT RELIABILITY; CONTAMINATION REMOVAL; REMOTE SYSTEM; RADIATION EXPOSURE; RADIATION EXPOSURE LIMIT

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Objectives: The objective is to assess the impact of lower exposure limits on operations and maintenance activities at U.S. nuclear power plants. At present, the potential consequences are increased outage times, increased O&M costs, and increased training requirements to expand the labor pool. A second objective is to propose approaches which would help to mitigate the impact of the new dose limits.

Comments: Factors which are likely to be affected by reduced exposure limits are:

- Restriction in the use of experienced workers on high dose jobs.
- Restriction on utilization of key workers by the contractors.
- High dose work may require use of more crews to limit individual exposures.
- Reduced flexibility for job planning.
- Increase in the number of workers and in health physics coverage.

In response to the need to reduce these impacts, the industry is accelerating the implementation of advanced radiation control technology. Sources of radiation are being reduced by replacing cobalt containing materials, wherever it is reasonably feasible. Advances are being made in water chemistry for both PWR and BWR type plants. Control of out of core deposits is being achieved by preconditioning techniques. Robotics and full-system decontamination technologies are under development.

Remarks/Potential for dose limitation: The primary disadvantage with U.S. reactors is that they are of older design when not much attention was paid to avoiding in-core cobalt sources and tight water chemistry control, resulting in high cobalt-60 inventories, which can only be reduced slowly by subsequent actions. Despite this disadvantage, exposures have been steadily declining at U.S. plants.

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Some actions are showing good success. For example, zinc injection in BWRs has reduced radiation field buildup by 50%. Doses from steam generator replacement in PWRs, and pipe replacement in BWRs, have been appreciably reduced over the years. Steam generator fields at PWRS, using elevated chemistry, are 1.5-3 times lower than at plants using standard chemistry. However, since there is a large number of older plants in the U.S., average plant exposures tend to be distorted by this age factor.

References: Wood, C.J., "Mitigation of Reduced Exposure Limits on Nuclear Plant Operations," Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: **from:** 1990 **to:** 1992

Funding: N/A

Status: In progress

Last Update: April 12, 1992

SOURCES OF COBALT-60 IN THE PRIMARY SYSTEMS OF PRESSURIZED WATER REACTORS

Keywords: CONTAMINATION PREVENTION; COBALT-60; COBALT SOURCE; PRIMARY CIRCUIT; PWR; RADIATION EXPOSURE; RADIATION FIELD; STELLITE

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Objectives: Identification of the main sources of cobalt-60 in PWR primary circuits is necessary, if these are to be removed from new plants and slowly reduced in existing plants as component replacement occurs. The objective of this work was to assess cobalt-60 sources, using three independent approaches.

Comments: The basis of the three methods used was as follows:

1. Oxides on components removed from the primary circuit of several Westinghouse PWRs were greatly enriched in cobalt relative to the base metal (factors 3 to 36). After analysis it was concluded the excess cobalt did not arise from the principal materials, feedwater, or chemical addition. It appears that the cobalt has come from Stellite. This is supported by the absence of cobalt enrichment in reactors which are free of Stellite.
2. A synthetic assessment of cobalt-60 sources was carried out using recent corrosion release data, and gave similar results.
3. An analysis of isotope specific radiation field and coolant radionuclide data was carried out from a series of KWU stations where Stellite inventory has been successively reduced. Stations having the lowest Stellite inventories exhibit radiation fields about one order lower than earlier, fully Stellited plants. Reduction in fields correlated with coolant cobalt-60 content, which correlated with Stellite inventory. There was no correlation with inventories of Co-58 and Stellite.

Potential for dose limitation: Although each of the methods contains uncertainties, they all lead to the conclusion that Stellite, when present, is the main source of cobalt-60 in PWR primary circuits, and leads to elevated plant radiation fields. The only exception is Stellite contained in control rod drive mechanisms which does not appear to contribute significantly to cobalt-60 inventories. In plants where Stellite has been eliminated, radiation fields are low and are dominated by cobalt-58.

In the Sizewell 'B' PWR, under construction in the United Kingdom, Inconel 690 steam generator tubing (<0.015% Co) is estimated to contribute only 2% to the cobalt-60 inventories. Fuel grids with 0.03% cobalt are estimated to contribute about 20%.

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U.K. R-291

References: Swan, T., "Sources of Cobalt-60 in PWR Primary Systems," Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

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

Funding: N/A

Status: Completed

Last Update: May 2, 1991

PERFORMANCE OF IRON BASE HARDFACING ALLOYS UNDER PRESSURIZED WATER REACTOR CONDITIONS

Keywords: CONTAMINATION PREVENTION; COBALT-60; COBALT SOURCE; IRON BASE HARDFACING ALLOY; PWR; STELLITE; NOREM; RADIATION FIELD

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Objectives: To evaluate the performance of a number of iron base hardfacing alloys in comparison with Stellite-6 in simulated LWR and PWR conditions. To find suitable alternatives to cobalt based Stellites.

Comments: Researchers characterized weld deposits by using metallography, chemical analysis, hardness, and dye penetrant tests. They operated the valves in a test loop under simulated primary PWR coolant chemistry conditions. Materials such as EB 5183, Everit 50, NOREM B1, NOREM B4, as well as Stellite 6, were tested by stroke cycling. The valves were opened and closed 2,000 times over a period of one year in PWR conditions. The valve with Stellite 6 trim served as a standard.

Prior to testing a baseline inaugural inspection was performed. At approximately 3 month intervals, after 500 stroke cycles the performance of the alloys was monitored by nondestructive examination, profilometry, and leak testing. Also corrosion coupons were used to monitor loss from corrosion alone. After all 2000 cycles, the valve seats and disks were subjected to comprehensive destructive examinations.

Remarks/Potential for dose limitation: Corrosion testing indicated that metal loss was minimal. Based on profilometry, the candidate alloys were found to have excellent wear properties. Cold leakage through the valves was very low. Hot leakage on one valve with EB5183 and another with NOREM was found to be zero, indicating these are the best replacement choices. All the candidate iron base alloys have performed better than the Stellite 6 standard based on assessments of sliding wear damage from visual examinations, profilometry data, and seat leakage test results. The valves in this program have performed satisfactorily for more cycles than the vast majority of plant valves will experience over a plant lifetime. These data provide the nuclear utilities pursuing aggressive dose-reduction programs with evidence that iron base hardfacing alloys have the attributes needed to ensure adequate valve performance under PWR operating conditions.

References: 1. "Qualification Loop Tests of Cobalt-Free Hardfacing Alloys - PWR Phase, 1989-1990 Progress", EPRI NP-7030-D, Interim Report, November 1990. 2. Inglis, I., and V. Murphy, "Performance of Iron Base Hardfacing Alloys in a One Year Test Program Under

PWR Conditions," Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1992

Funding. EPRI

Status: Completed

Last Update: April 16, 1992

U.K. PROGRAMME TO QUALIFY COBALT-FREE HARDFACING ALLOYS

Keywords: CONTAMINATION PREVENTION; COBALT-60; COBALT SOURCE; PWR; RADIATION FIELD; HARDFACING ALLOY; IRON BASE ALLOY; NICKEL BASE ALLOY

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Objectives: The objective of this program was to qualify cobalt-free hardfacing alloys, with the focus on valve applications. A number of iron base alloys such as Nitronic 60, Delcrome 90 and 910 (equivalent to Everit 50), nickel base alloys such as Colmonoy 5 and 84, together with two wrought stainless steels (17-4 PH, and 440C), were tested. The reference cobalt base alloy used was Stellite 6.

Comments: The program was carried out in two phases:

In the first phase the hardfacing alloys were deposited using the PTA and mechanized TIG processes and evaluated for wear, corrosion, erosion and cavitation performance, in relevant environments. The wear and corrosion tests were conducted in primary coolant conditions, representative of both normal operations and plant shutdown. Based on the test results, Delcrome 910 and Colmonoy 5 were selected for further evaluation in valve operability tests.

In the second phase of the program, valves hardfaced with Stellite, which had undergone valve operability tests, were refurbished with the alternate alloys and the operability tests repeated. Three "gate valves" and one "globe valve" were selected for evaluation.

Remarks/Potential for dose limitation: In general, valve performance was comparable when hardfaced with Stellite 6 or the alternate alloys. However, the valve post-test examination revealed evidence of pitting in the Colmonoy hardfacings. Delcrome 910 (Everit 50) was considered the preferred alternate alloy.

References: Airey, G.P., "Qualification of Cobalt-free Hardfacing Alloys," Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: N/A

Status: Completed

Last Update: June 17, 1991

SUPPLYING COBALT-FREE NUCLEAR VALVES

Keywords: CONTAMINATION PREVENTION; COBALT-60; COBALT SOURCE; PWR; RADIATION FIELD; HARDFACING ALLOY; COBALT-FREE ALLOY

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Objectives: The Anchor/Darling Company has been a major supplier of ASME Section III "N-stamped" valves to the Power Industry. It has been involved in the development of non-Cobalt based hardfacing materials for the U.S. Navy and the commercial program for the last 25 years. Originally, the motivating force was the potential shortage of cobalt. Lately, it has been due to concerns about the radiation exposure of nuclear power plant maintenance personnel.

Comments: The materials evaluations include hardfacing deposition, brazing, material strength and wear characteristic evaluations.

Initially, deposition of the "new" hardfacing materials is used to evaluate the feasibility of the alloy deposition within the valve product line. Brazing evaluations are used to determine if the material can successfully withstand brazing at 2,000°F, rapid heating and cooling rates, post-braze heat treatment of the valve body or bonnet material, and whether it can maintain acceptable final form and physical properties.

To evaluate new materials for seat rings and discs, the new materials are prepared in their final shape and subjected to mechanical tests that simulate in-valve use of the parts. The test results are then compared to acceptance criteria. If acceptable, the material is then subjected to wear testing.

Wear resistance is used to evaluate the candidate non-cobalt based material's wear resistance as used in valve service.

Remarks/Potential for dose limitation: It should be noted that gate valves require seat ring and disc seating material that is highly wear resistant. Check valves and globe valves do not require high wear resistant materials. Control valve components require good erosion and wear resistance properties.

At present, we can provide 2" and smaller globe and check valves and 2 1/2" and larger gate, globe and check valves with non-Cobalt based seat and disc materials. The non-Cobalt based materials include 316 stainless steel, 416 stainless steel, 440C stainless steel, SA564-630, Tungsten Carbide, Nitronic 60, Colomonoy 5, Deloro 50 and Norem B4.

Although several acceptable non-Cobalt based materials have been found, this remains a continuous program to evaluate new candidate materials for these applications.

References: Bensinger, F. A., "Supplying Cobalt-free Nuclear Valves: An OEM Perspective," Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1992

Funding: N/A

Status: In progress

Last Update: April 17, 1992

AN EXAMINATION OF FOREIGN APPROACHES TO CONTROLLING RADIATION-FIELD BUILDUP IN BOILING WATER REACTORS

Keywords: CONTAMINATION PREVENTION; COBALT-60; COBALT SOURCE; BWR; RADIATION FIELD; IRON INPUT; WATER QUALITY

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Objectives: To identify approaches used at foreign BWRs that lower radiation fields and decrease worker exposures. To identify those measures that can readily be implemented at mature U.S. BWRs.

Comments: Several fundamental approaches have been identified:

- Cobalt input was reduced to as low as possible by replacement of cobalt alloys, use of low cobalt stainless steel and improvement of the condensate polishing system.
- Feedwater iron input was reduced <1 ppb by additions in the condensate filtration equipment, improvement in crud removal efficiency in the condensate polishing system, and control of dissolved oxygen in the feedwater.
- Cobalt-60 production and release from in-core surfaces was minimized by replacement of cobalt alloys, use of non-cobalt or low cobalt alloys and prefilming of the surfaces.
- Cobalt-60 release from fuel surfaces was reduced by minimizing Co-60 release during operations by reducing Fe crud input, control of Fe/Ni ratio in the feedwater in low crud plants and increase of the water pH to about 8. Modified shutdown procedures minimize release during shutdowns.
- Co-60 deposition on out-of-core surfaces was minimized by reducing deposition rate by use of prefilming and improving water quality to reduce corrosion rate.

Remarks/Potential for dose limitation: It was concluded that if the three fundamental approaches, namely removal of cobalt sources, reduction of corrosion product input, and improvements in water quality are properly implemented then the following should result:

- In "old" plants, radiation fields should decrease gradually to no higher than ~100 mR/h.
- In "new" plants the radiation fields should increase slowly and level of at no higher than ~50 mR/h.

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The three main criteria for water chemistry for reduced radiation field buildup are:

- Co-60 concentration in reactor water of <0.05 µCi/Kg
- Reactor water conductivity of < 0.2 µS/cm (25°C).
- Fe concentration in feedwater between 0.5 - 1 ppb.

References: 1. "Foreign Approaches to Controlling Radiation-Field Buildup in BWRs", EPRI NP-6942-D, Interim Report, August 1990. 2. Lin, C. C., "Foreign Approaches to Controlling Radiation-Field Buildup in BWRs," Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991 **Funding:** N/A

Status: Completed **Last Update:** April 16, 1992

GUIDELINES FOR THE REDUCTION OF COBALT FROM REACTOR SYSTEMS

Keywords: CONTAMINATION PREVENTION; COBALT-60; COBALT SOURCES; BWR; PWR; RADIATION FIELD; COBALT REDUCTION; GUIDELINES

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Objectives: The objectives of this program were:

- To identify valve/hardfacing design requirements.
- To review/summarize available data on cobalt-free hardfacing alloys.
- To identify appropriate valve and trim alloys in light of duty cycle.
- To summarize util y experience with cobalt-free alloys.
- To address prospective safety issues.
- To define tests to establish "equivalency" between Stellite and cobalt-free alloys.
- To identify cobalt impurity limits for structural alloys.

Comments: The main hardfacing requirements are wear resistance, corrosion resistance, and weldability.

The hardfacing equivalency issue includes wear tests in the laboratory for galling/adhesive wear, corrosion resistance, and weldability evaluations, followed by component tests under PWR and BWR chemistry development of in-situ repair procedures. All three iron-base alloys under test at Atomic Energy of Canada performed better than a cobalt-base standard under PWR chemistry conditions.

The following guidelines are proposed for cobalt impurity levels:

Inconels for steam generator tube replacement (<0.015% cobalt).

Stainless steels for BWR control blades and recirculation piping (<0.02% cobalt).

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

- Hardfacing is not needed in flow control valves.
- Cobalt-based valve trim or their equivalent should be limited to highly loaded gate and globe valves (>15 ksi).
- Loop tests are needed to establish equivalency between iron-base alloys and Stellite.
- Low cobalt impurity levels should be specified for structural alloys used in replacement components.

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References: Ocken, H., "Cobalt Reduction Guidelines," Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: N/A

Status: In progress

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U.S.A. R-297

BWR RADIATION FIELD TRENDS

Keywords: CONTAMINATION PREVENTION; COBALT SOURCE; BWR; RADIATION FIELD; RADIATION FIELD TREND; RECIRCULATION SYSTEM; DRY WELL DOSE RATE

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Objectives: The objectives of this program were:

1. To compile measurements of dose rates on BWR recirculation systems at standard point.
2. To make inter-plant comparisons from the data collected.
3. To analyze any trends.

Comments: The measurements were carried out at selected standard points in the recirculation systems, using Eberline ESP-1 with an HP220A probe and with Victoreen 450B survey meters.

The following plant characteristics were included in the analysis:

- Extent of exposure.
- Whether plant was a zinc or non-zinc plant.
- Type of pipe finish.
- Type of treatment given to the condensate.
- Type of heater drains.

Work is continuing on normalizing plant dose rates by taking into account the thickness and diameter of piping. Wall thickness seems to have a major effect. Work is also continuing on taking into account the number of days of hot operation. Here, too, differences are apparent.

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

- The zinc plants showed lower dose rates.
- Hydrogen water chemistry had an appreciable affect on dose rate build up, but no observable effect on decontamination.
- Several plants showed that decontamination was highly effective in reducing dose rates.
- Without operational changes recontamination occurs at the usual rate and approaches the steady state within 2 EFPY.

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- A combination of electropolis. ng and prefilming showed consistently lower dose rates. Sandblasting was found to be counter-productive.
- The dose rates at plants that had forward pumped heater drains were consistently higher whether they were zinc or non-zinc, although the dose rates at forward pumped heater drain plants with zinc injection were somewhat lower than non-zinc plants as a whole.
- F/D condensate treatment was found to be better than deep bed.

References: ") Kincaid, C.B., and H. L. Kenitzer, "BWR Radiation Field Trends," Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken. 2) "BWR Radiation Field Assessment," EPRI Report RP-2494-01.

Duration: from: 1990 to: 1991

Funding: N/A

Status: Completed

Last Update: April 17, 1992

STATUS OF ZINC INJECTION IN BOILING WATER REACTORS

Keywords: CONTAMINATION PREVENTION; ZINC INJECTION; COBALT; BWR; RADIATION FIELD; DOSE RATE; RECIRCULATION SYSTEM; DRY WELL DOSE RATE; ZINC ADDITION

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Objectives: The objectives of this program were:

1. To investigate the extent of radiation buildup in plants using zinc injection.
2. To re-examine the Zn-65 outage release concern.
3. To examine the availability of zinc depleted in Zn-64.
4. To eliminate the precursor of Zn-65.
5. To examine the impact of hydrogen water chemistry on zinc addition.

Comments: Currently nine BWRs have implemented zinc injection and four more plants are making provision for its installation. The current status is as follows:

- A passive system to inject zinc has been tested and is qualified for general application.
- Potential sources of zinc oxide depleted in Zn-64 (GEZINC) have been identified and it should be available soon.
- Dose rates at plants using normal water chemistry are excellent.
- Dose rates at plants using hydrogen water chemistry are higher than expected There is a need to understand the mechanism at work for HWC.
- Feedwater iron input must be kept below 5 ppb, with the target set for <2ppb.
- The ways to minimize the impact of Zn-65 is to minimize iron input and/or to use GEZINC.

Remarks/Potential for dose limitation: Zinc reduces corrosion film buildup of Co-60. However zinc does not reduce crud deposition buildup and hydrogen water chemistry adversely affects the buildup of radiation fields. Evaluation to understand the hydrogen water chemistry mechanism is in progress.

A passive system to inject zinc is now operational.

Zinc depleted in Zn-64 should be available by the end of 1991.

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References: Marble, W.J., "Zinc Injection Update," Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: N/A

Status: In progress

Last Update: June 26, 1991

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EXPERIENCE WITH ZINC INJECTION AT MILLSTONE 1

Keywords: CONTAMINATION PREVENTION; ZINC INJECTION; COBALT; BWR; RADIATION FIELD; DOSE RATE; RECIRCULATION SYSTEM; DRY WELL DOSE RATE; ZINC ADDITION

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Objectives: Millstone 1 is a 660 MWe GE designed Mark 1 BWR. It has deep bed condensate polishers with no provision for regeneration. It started commercial operation in 1970. It has been classified by GE as a non zinc plant.

The objectives of this program were:

1. To demonstrate the impact of zinc injection on a mature plant with well-established oxides (as opposed to the demonstration at the "new" Hope Creek plant).
2. To examine the extent of Zn-65 outage release.
3. To decontaminate the plant prior to implementation and then evaluate the overall impact of zinc addition, including extent of recontamination and reduction in dose to plant personnel.
4. To make recommendations to continue or to terminate after two fuel cycles with zinc injection.

Comments: The results were as follows:

1. There was an appreciable drop from the predicted dose rates. Using GE's four standard survey points dose rates were expected to be about 240 mR/h but were measured to be just over 100 mR/h.
2. The rate of recontamination on recirculation piping decreased by 30 to 35% and little or no recontamination was observed in the Reactor Water Clean Up System piping.
3. Actual job-related dose savings were difficult to assess. They were conservatively estimated to be 6% attributable to zinc injection.

Due to the experience of Hope Creek, measures were taken to mitigate the Zn-65 spiking problem. These included: no zinc injection two weeks prior to outage, maintaining RWR, and RWCU systems in service up to and including fuel off load. Only minor releases of Zn-65 were seen at about 300 - 400°F and during cavity level changes. The quantity of Zn-65 appeared to be related to the quantity of iron in the feedwater.

Remarks/Potential for dose limitation: After two cycles are complete, we hope to assess a number of parameters, including:

- Piping dose rate trends.
- Job-related dose avoidance.
- Extra waste loading and costs.
- Extra dose from Zn-65 contamination.
- Equipment and manpower costs.

We will also evaluate the extent of recontamination and make a full assessment of ALARA related cost/benefits. After that, recommendations will be made on whether to continue with or terminate zinc injection.

References: Hudson, M.J., and D.L. Wilkins, "Experience with Zinc Injection at Millstone 1," Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: N/A

Status: In progress

Last Update: June 26, 1991

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U.S.A. R-300

CONTROL OF RADIATION FIELDS AT BOILING WATER REACTORS BY REDUCING IRON INPUT

Keywords: CONTAMINATION PREVENTION; IRON INPUT; COBALT; BWR; RADIATION FIELD; DOSE RATE; COBALT-60; MANGANESE-54; CRUD DEPOSITION; CORROSION FILM

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Objectives: Shutdown radiation fields have been monitored at Susquehanna, a BWR plant with two 1050 MWe units. The object was to determine the source of the radiation fields and to identify the most effective approaches to reduce them. Isotopic gamma scans of recirculating and RWCU piping indicated that the radioactive deposits are principally due to Co-60 and Mn-54. By relating the radiochemistry of the reactor water to the isotopic deposits on the out-of-core piping, iron oxide crud migration was found to be a more important transport and deposition process than the incorporation of soluble isotopes into the corrosion films on out-of-core piping. Crud deposition was found to account for 60-70% of the shutdown radiation fields; corrosion film accounted for 30-40%. By reducing the activated crud deposited on the fuel rods, the transport of isotopic species to out-of-core piping can be reduced. The source of the iron oxide crud is the feedwater iron input to the reactor. The objectives of this program were to identify the sources of iron in the feedwater and find ways to eliminate these sources.

Comments: Analysis showed that the major sources of iron in the feedwater are:

- Corrosion/erosion of carbon steel steam extraction piping.
- High iron input (20-30 ppb) to condensate demineralizers.
- Poor iron removal efficiency by condensate demineralizers (70%).
- Incomplete and indeterminate cleaning of condensate demineralizer resin (65%).

The following actions are being pursued:

- Replacement of steam extraction piping with more corrosion/erosion resistant material than carbon steel
- Identification of areas of high erosion in the condenser and formulation of mitigation plans.
- Modification of condensate demineralizer resin vessels to provide complete resin transfer and eliminate heels.
- Improvement in the operation of ultrasonic resin cleaner and verification of the cleanliness of cleaned resin.
- Assessment of new resins and mixtures for the condensate demineralizer.

Remarks/Potential for dose limitation: As these actions are implemented, their impact on Susquehanna radiation fields is being monitored. Once these actions have been fully implemented, the personnel exposure due to radioactive crud deposition is expected to decrease by one-half.

References: Pacer, J.C., "Control of Radiation Fields at Susquehanna by Reducing Iron Input," Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: N/A

Status: in progress

Last Update: June 26, 1991

EFFECT OF PRECONDITIONING ON COBALT CORROSION RELEASE RATES

Keywords: CONTAMINATION PREVENTION; CORROSION; COBALT; CORROSION PRODUCT RELEASE; PRECONDITIONING; STEAM GENERATOR

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Objectives: The objective of this project was to obtain answers to the following questions:

1. Is the preconditioning of material surfaces able to reduce the release of metallic elements and radionuclides from the alloys in the coolant?
2. Do the hot functional and precritical test "passivate" the circuit surfaces?
3. What is the best preconditioning to apply to the tubes in case of a replacement of the steam generators?

Comments: Tests were carried out on mill annealed inconel 600, heat treated inconel 600, and heat treated inconel 690 tube specimen. Several preconditioning processes were carried out, including FRA, COR, RCT, EDTA, and KOH. The release rates were found to be important below a pH300C of 6.9, but the effect was slight between 7.0 and 7.5. It seems that an optimum value of pH exist for minimum release around 7.2. For mill annealed Inconel 600, there was greater release at the beginning of the test, there was a decrease in the rate of release as time progressed. Moreover, the release rate of identical test sections could differ to a greater extent than tubes with different types of preconditioning. For heat treated Inconel 600, comparison between preconditioned and unpassivated test sections showed that release rates for preconditioned samples start to decrease but, after 1500 hours of operation, the release rates become constant and start to approach the values measured for unpassivated samples. The preconditioning creates an oxide layer which dissolves during the first 1000 hours.

Remarks/Potential for dose limitation: The following conclusions were drawn:

1. Preconditioning has no effect on release rate after 1000 hours of operation.
2. At the beginning of the test, preconditioned test sections have a higher release rate than unpassivated test sections.
3. It appears that at the beginning of the test, a part of the oxide layer formed during the preconditioning operation is dissolved.
4. The hot functional test should be carried out with deaerated water in a hydrogen atmosphere to maintain the reducing environment.

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References: Beslu, P., S. Anthoni and N. Galliano, "Effect of Preconditioning on Cobalt Corrosion Release Rates", Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: N/A

Status: Complete

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U.S.A. R-302

RADIATION FIELD ISSUES IN SWITCHING TO HYDROGEN WATER CHEMISTRY

Keywords: COMPONENT RELIABILITY; BWR; RADIATION FIELD; DOSE RATE; CHEMISTRY; HYDROGEN WATER CHEMISTRY; IGSCC; CRUD DEPOSITION; ZINC INJECTION; ZINC ADDITION

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Objectives: Hydrogen water chemistry (HWC) was developed to combat the tendency for intra-granular stress corrosion cracking in BWR piping. 7 domestic and 6 foreign plants (some with multiple units) have operated under this chemistry regime. Five more U.S. plants and one foreign plant are planning to change to HWC.

However, an increase in shutdown dose rates has been reported by some HWC plants. The magnitude of the increase appears to be plant specific. Crud hot spots are also a problem in some of these plants. The dominant isotope is Co-60. The increase indose rates has also been observed in a Japanese steam generating heavy water reactor. However, no rise in dose rates has been observed in Swedish designed plants using HWC plants.

The objective of this study is to understand fully the mechanisms that cause the increases in dose rates and to develop appropriate remedial actions.

Comments: The following changes in water chemistry have been observed with the introduction of HWC:

- 2 to 3X increase in total reactor water Co-60.
- Spiking of insoluble crud (Co-60, Mn-54, Zn-65).
- Increase in soluble Co-60.
- Reduction in soluble Cu and reduction in total Cr-51 content.

For HWC radiation buildup, the following has been observed:

- Magnitude of piping dose rate increase affected by:
  - Zinc addition in reactor water.
  - H2 addition rate.
  - Co-60 concentration in reactor water.

- Increase in reactor water Co-60 concentration is potentially affected by:
  - H2 addition rates.
  - Crud deposit on fuel surfaces.
  - Cobalt containing alloys in reactor core.
- Increase in radiation buildup may be transient in nature.
- Effect appears to be less in zinc plants.

Remarks/Potential for dose limitation: A joint utility/EPRI/GE program is under discussion.

The present prognosis is as follows:

- The effect should be transient in nature but may be longer than one cycle.
- Deliberate cycling of the hydrogen addition rate probably exacerbates the problem; a sustained program of hydrogen addition is believed to minimize it.
- Plants with low cobalt core alloys may see lower amounts of Co-60 in the reactor water.
- Plants which utilize zinc addition should see a smaller amount of transient increase in the radiation buildup.

References: Ruiz, C.P., and R.L. Cowan, "Switching to HWC: Radiation Field Issues," Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991 **Funding:** N/A

Status: In progress **Last Update:** June 26 1991

QUALIFICATION OF ELECTROPOLISHING FOR REPLACEMENT STEAM GENERATORS

Keywords: CONTAMINATION PREVENTION; PWR; RADIATION FIELD; DOSE RATE; ELECTROPOLISHING; STEAM GENERATOR; IGSCC; CHANNEL HEAD; MECHANICAL SURFACE PREPARATION

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Objectives: A prev ous EPRI sponsored program found electropolishing to be harmless to the metallurgical pr rties of PWR steam generator construction alloys. However, only one set of electropol g conditions were used, and the program did not have the consensus necessary for industr wide acceptance.

The objective of this follow-up program was to establish acceptance criteria for surface modification of PWR channel head cladding, divider plate and connector welds. The approach taken was to (a) obtain protypic test samples, representing not only normal process but also weld repair and bead tie-ins for the overlays; (b) apply a wide range of surface smoothing parameters (mechanical and electropolishing); (c) analyze the results to establish acceptable process parameters related to metallurgi al and surface analysis findings.

Comments: The following summarizes the work done during the study:

- A set of parameters was established with wide process margins. This included current densities of 10 to 17 A/sq.in, a duration between 2 to 8 minutes, and a temperature range between 130 to 160°F.
- Material removed was between 8 to 20 mils; 80% was removed by mechanical preparation and 20% by electropolishing.
- Process worst case scenarios were examined, including weld repairs, bead tie-ins and low ferrite content.
- Tests conducted failed to reveal significant surface degradation. Surface contaminants will not be detrimental to balance of steam generator's components or primary water circuit. Note: Material was selected with the highest promise to reveal damage.

Remarks/Potential for dose limitation: Results indicate that, compared with nonpolished surfaces, mechanically polished and electropolished surfaces both show a significant reduction in radioactivity buildup. Specific metallurgical highlights include the following:

- No intergranular attack in 309L weld overlay, Inconel 600 or Inco 82 weld.

- Preferential ferrite removal kept at a minimum.
- Smooth surfaces achieved with wide process parameters.

In regards to surface residues, electrolyte removal was verified by EDS surface chemical and electron microprobe analysis.

This study established surface smoothing parameters that will not compromise the reactor performance of weld overlays in PWR steam operation components. The process parameters recommended can be used to prepare reliable procedures for direct field application to the surface of replacement channel heads, welds, weld overlays, and divider plates.

References: 1. "Electropolishing Qualification Program for PWR Steam Generator Channel Heads", EPRI NP-6617, Final Report, January 1990. 2. "Electropolishing Process Development for PWR Steam Generator Channel Heads", EPRI NP-6619, Final Report, April 1991. 3. Spalaris, C.N., and C.J. Wood, "Qualification of Electropolishing for Replacement Steam Generators," Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

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

Funding: N/A

Status: Completed

Last Update: April 16, 1992

FRENCH EXPERIENCE WITH ELECTROPOLISHING STEAM GENERATOR CHANNEL HEADS

Keywords: CONTAMINATION PREVENTION; PWR; RADIATION FIELD; DOSE RATES; ELECTROPOLISHING; STEAM GENERATOR; IGSCC; CHANNEL HEAD; SURFACE PREPARATION

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Objectives: A collaborative program between Framatome, EdF and CEA has been in place since 1980 to examine the potential of electropolishing as a means of reducing high radioactivity buildup on steam generator channel heads and the resulting high radiation exposures to occupational workers. The first tests were conducted in 1984 on Chinon B1 NPP. Between 1988 and 1991, 24 steam generator channel heads from 6 of the 1300 MWe series power plants have been electropolished on the plant. In 1989, the three DAMPIERRE 1 replacement SGs were electropolished by Framatome. The present study analyzes the results of this experience.

The procedure was to compare the deposited activity from Co-58 and Co-60 on a reference surface with the deposited activity on the treated surface. A reduction by a factor of 3 was observed for the electropolished surface. Various tools were developed during this work, for the divider plate, for the curved area of the clad bowl, for the manway, and for the primary pipe outline.

Comments: The following controls were exercised: (a) surface pH, (b) water conductivity, (c) electrolyte concentration (Fe, Ni, Cr), (d) in-situ molecular examination, (e) microscopic examination, (f) surface profiles. Also, during the on-line work, the number of passes, intensity, temperature and speed of sucker were noted. About 80% of the surface was electropolished. The results were very satisfactory.

The first field measurements, made with shielded TLDs at Nogent 1, showed a decrease in buildup at the partition plate by a factor of 3.5 and at the clad bowl by a factor of 2.5. The results at Nogent 2 were even better. Buildup decreased by 4.5 at the clad bowl and by 6.6 at the partition plate. The average steam generator channel head dose rate at Nogent 2 had been reduced to 2.2 R/h, a reduction of approximately 45%.

Remarks/Potential for dose limitation: Future plans include the electropolishing of all 4 steam generators at new Golfech 2, Chooz B1, Chooz B2 nuclear power plants, as well as a number of replacement steam generators of older plants.

Development work will include a new electrolyte which will not contain any sulphuric acid and a number of new tools to simplify the process and make it more efficacious.

BNL ALARA Center Data Base

References: Saurin, P., C. Weber, N. Engler, B. Lantes, A. Brissaud and G. Gouillardou "French Experience with Electropolishing," Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: N/A

Status: In progress

Last Update: June 26, 1991

SURFACE PRETREATMENT OF PRIMARY SYSTEM COMPONENTS TO REDUCE RADIATION BUILDUP

Keywords: CONTAMINATION PREVENTION; PWR; BWR; RADIATION FIELD, DOSE RATE; ELECTROPOLISHING; MECHANICAL POLISHING; STEAM GENERATOR; RECIRCULATION PIPING; IGSCC; CHANNEL HEAD; SURFACE PREPARATION; PRETREATMENT; PASSIVATION; PREFILMING

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Objectives: To investigate the efficacy of two primary techniques to reduce radiation buildup in PWR and BWR plants:

- Surface finishing to reduce true area through mechanical and electropolishing.
- Passivation to prevent corrosion and deposition through preoxidation of base material and pretreatment with coatings.

Comments: The latest application to replacement piping in BWRs has been at the Cooper, Peach Bottom-3 and BSEP-1 & 2 plants. For electropolishing (E/P) a Phosphoric/Sulfuric acid electrolyte was used and polishing densities ranged from 10 to 30 amp/dm2. RCT passivation was carried out at 300°C, atmospheric pressure, utilizing circulating, hot moist air. The exposure time was between 10 and 150 hours. The results showed low activity and low dose rates since the piping was replaced at all the above BWR plants. The application of surface pretreatment processes was also carried out at Chinon B-1, Tihange and Doel-2 PWR plants, each under slightly different conditions. At Chinon B-1, tests were carried out using electropolishing alone and also combined with either the RCT or Framatome passivation process (FRA). The Framatome process is carried out at 280°C, at a pressure of 153 atmospheres for an exposure time of 360 hours. From the amount of activated corrosion product deposition it was determined that E/P combined with RCT passivation is the most efficacious technique in reducing buildup, followed by E/P and FRA.

Remarks/Potential for dose limitation: For BWRs the electropolished and preoxidized alloy steel showed:

- Low radiation level buildup and low contact dose rates.
- The dose rates leveled off or were declining after 3 cycles.
- The costs were rapidly recovered in terms of reduced occupational exposures.

For PWRs, the combined plant result lead to the following conclusions:

- Electropolishing reduced the radiation buildup rate by a factor of 5.
- Addition of RCT passivation to the electropolished surface further reduced buildup rate by up to a factor of 2.
- Addition of the Framatome process yielded an improvement of 10 to 20%.
- Chromium rich film looks extremely promising in reducing radiation buildup.
- Palladium coating gave mixed results.

References: Asay, R.H., "Surface Pretreatment of Reactor Components to Reduce Radiation Buildup," Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991 **Funding:** N/A

Status: In progress **Last Update:** June 26, 1991

BNL ALARA Center Data Base

BELGIUM R-306

REDUCING RADIATION BUILDUP BY SURFACE COATING OF PRIMARY SYSTEM COMPONENTS

Keywords: CONTAMINATION PREVENTION; PWR; RADIATION FIELD; DOSE RATE; SURFACE TREATMENT; SURFACE COATING; ELECTROPOLISHING; MECHANICAL POLISHING; STEAM GENERATOR; IGSCC; CHANNEL HEAD; SURFACE PREPARATION; PRETREATMENT; PASSIVATION; PREFILMING

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Objectives: The Doel-2 PWR was selected to carry out in-plant tests to evaluate the effect of surface treatment and coating in reducing contamination rate and to inhibit activity buildup on new surfaces. The principal plant materials are: stainless steels 304L, 309L, 316L, CF8M cast material, and Inconel 600. In 1986, coupons were only exposed in the hot leg; in 1988 they were exposed in both the hot and cold legs. Various combinations of electropolishing, mechanical polishing, RCT passivation, chromium plating and palladium coating (as well as Quadrex SRI methods) were used.

Comments: The details of the Quadrex and SRI methods were as follows:

Quadrex method:

- Activated 304L and Inconel 600 in sulfuric acid (25%) for 80 seconds at 70°C.
- Electroless plated 304L and Inconel 600 in $PdCl_2$ solution.
- Exposed 304L for 10 minutes at 49-55°C.
- Exposed Inconel 600 for 20 minutes.

SRI method:

- Cleaned coupons with toluene solution.
- Activated 304L and Inconel 600 in sulfuric acid (25%) for 80 seconds at 70°C.
- Electroless plated 304L and Inconel 600 in $PdCl_2$ solution for 10 minutes at 52°C.

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

- Electropolished and electropolished/passivated Inconel 600 coupons showed reduction in Co-58 and Co-60 by a factor of 4 compared with 'as received' coupons in the same cycle.
- Electropolishing and passivation reduced buildup rate by factors of 2 to 3.
- Palladium coated coupons showed Co-58 and Co-60 activity levels 2 to 3 factors greater than electropolished and electropolished/passivated 304L and Inconel 600.
- Chromium plated and passivated 316L coupons showed Co-58 and Co-60 activity levels approximately an order of magnitude lower than electropolished and passivated 309L and CF8M coupons.

References: Roofthooft, R., "Surface Coating of Primary System Components to Reduce Radiation Buildup," Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: N/A

Status: In progress

Last Update: June 26, 1991

PWR PRIMARY WATER CHEMISTRY GUIDELINES - REVISION 2

Keywords: CONTAMINATION PREVENTION; PWR; RADIATION FIELD; DOSE RATE; WATER CHEMISTRY; GUIDELINES; LITHIUM; PH CONTROL; STEAM GENERATOR; PWSCC; CORROSION; FUEL CLADDING; PH CALCULATION METHODOLOGY

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Objectives: The original function of primary chemistry guidelines was to promote industry consistency in addressing the need to minimize radiation buildup. More recently incidents of PWSCC has prompted the guidelines to review the impact of chemistry on materials integrity, to ensure that the zest for radiation field control did not initiate a problem. These guidelines are intended to serve as a model only for the development of site specific programs. The technical basis of the guidelines regarding pH/lithium is as follows:

- Operation at pH(T)< 6.9 can lead to crud deposition on fuel surfaces, which can cause fuel cladding failures.
- Depending on other core parameters, operation at high lithium concentrations can integrate lithium in the zirconium oxide film, which can lead to accelerated oxidation rates and eventual fuel cladding failures.
- Operation at lithium concentrations greater than 2.2 ppm may lead to PWSCC of mill-annealed alloy 600 steam generator tubes.

Comments: Five guiding principles in order of priority are:

1. Operate at or above $pH(T_{average})= 6.9$ to minimize crud deposition on fuel and enhanced Zircaloy oxidation. Assess impact on PWSCC if operation above 2.2 ppm lithium is required. Consider a fuel surveillance program if plant-specific conditions require operation below $pH(T_{average})= 6.9$.
2. A plant-specific fuel and materials review should be carried out and a fuel surveillance program considered for operation above 2.2 ppm lithium to achieve $pH(T_{average}) > 6.9$.
3. Once lithium has been reduced to 2.2 ppm, select plant specific pH, between 6.9 and 7.4, based on impact on fuel, materials and radiation field control.
4. Maintain the specified pH at ± 0.15 ppm lithium until end of operating cycle, noting lithium variations have greater effect on pH at lower boron concentrations.

5. Attempt to minimize pH fluctuations during power operation. However, during power changes, some fluctuations in pH may be unavoidable.

Remarks/Potential for dose limitation: A new parameter, pH($T_{average}$) is a part of the new guidelines. The committee determined the appropriate temperature to be selected for each plant should be the average temperature of the water as it passed through the core. Thus $T_{average}$ = average of T_{inlet} and T_{outlet}. In order to ensure consistency within the industry in the method of calculating pH(T) and taking into account that the code selected was consistent with the published research data on the effects of pH on fuel and materials integrity, a review of the available industrial codes was carried out. It was found that the method of calculating pH(T) in each code available to the industry was slightly different. The committee finally selected the parameters used by Combustion Engineering in their SYSCHEM code. This code is commercially available and information about this code is available in Appendix A of the guidelines.

These guidelines provide utilities with a framework for developing a practical PWR water chemistry program during cold shutdown, startup, and normal operating conditions.

References: 1. "PWR Primary Water Chemistry Guidelines: Revision 2", EPRI NP-7077, Final Report, November 1990. 2. Brobst, G., "PWR Primary Water Chemistry Guidelines," Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: N/A

Status: Completed

Last Update: April 16, 1992

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U.S.A. R-308

REDUCTION OF RADIATION FIELDS BY ELEVATED PH CONTROL AT MILLSTONE-3

Keywords: CONTAMINATION PREVENTION; PWR; RADIATION FIELD; DOSE RATE; WATER CHEMISTRY; LITHIUM; PH CONTROL; STEAM GENERATOR; PWSCC; CORROSION; FUEL CLADDING

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Objectives: Early EPRI and Westinghouse studies indicated that elevated pH should be helpful in reducing radiation field buildup. The concept was successfully demonstrated at Ringhals 2,3 and 4 NPPs, but concerns arose about PWSCC of mill-annealed Inconel 600 SG tubing and the effect of elevated lithium on Zircaloy corrosion during extended fuel cycles. The EPRI/Westinghouse strategy was to demonstrate elevated pH at a U.S. plant with less susceptible SGs and to study PWSCC in the laboratory. Millstone-3 was selected because the 1150 MWe, four loop plant of Westinghouse is a relatively new plant, has Model F steam generators, with thermally treated I-600 tubing and hydraulically expanded tube to tubesheet connections, a void fraction similar to Ringhals 3 & 4, extended fuel cycle operation, good coolant control history and low SG channel radiation fields. The program elements included: maintain required Li/B ratio, carry out field surveys and gamma scans, carry out fuel and SG tube inspections and evaluate the data prior to restart.

Comments: Standard pH 6.9 chemistry was used during the first cycle. The second cycle was started with 3 ppm lithium and 1600 ppm boron and raised after two months to 3.5 ppm lithium. Good chemistry control was maintained at a pH of 7.4. Piping fields increased by 14%, when typically they would have increased by 30%, and channel head fields decreased 3%. There was no evidence of PWSCC but fuel sheath oxidation showed some data outside Westinghouse data base. During the third cycle, a lithium level of 3.5 ppm was maintained up to pH 7.2. Less time was spent with high lithium than during cycle 2 to minimize fuel sheath oxidation and yet realize ALARA benefits. However, some concerns, although minor, surfaced at this time about the effect of high lithium on PWSCC. The results after cycle 3 showed piping fields decreased by 8% and SG channel head fields were at an average value of 6 R/h. Fuel sheath oxidation was heavier than anticipated on high burn-up discharge assemblies. There was no evidence of PWSCC in tight U-bends or tube transitions in the steam generators.

Remarks/Potential for dose limitation: The following are the conclusions of this trial:

- The radiation fields remain low.
- There is no evidence of any adverse effect on Inconel 600.
- There is some concern about adverse effect of lithium on high burn-up Zircaloy oxidation, although this effect has not been fully confirmed as yet.

We are proposing to use a modified high pH coolant chemistry for cycle 4. Later, it is likely that we will revert to conventional coordinated lithium - boron chemistry and a pH of 6.9, mainly because of continued concerns about PWSCC and Zircaloy limitations at high burn-up.

References: Hudson, M.J.B., "Radiation Field Reduction by Elevated pH Control at Millstone 3," Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: N/A

Status: In progress

Last Update: June 17, 1991

LOOP EXPERIMENTS ON ZINC INJECTION UNDER PWR CONDITIONS

Keywords: CONTAMINATION PREVENTION; ZINC ADDITION; ZINC INJECTION; PWR; RADIATION BUILDUP; COBALT; 304 STAINLESS STEEL; ALLOY 600; COBALT RELEASE; OXIDE FILMS

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Objectives: It is now well known that zinc affects oxide films on BWR system materials, such as stainless steel. Cobalt-60 pickup and release rates are reduced and films are thinner and more protective. Most affected are Spinels. It was conjectured that there would be similar benefits for PWRs. It was therefore decided to investigate the effect of zinc addition on PWR materials under simulated PWR conditions and monitor such aspects as film growth on Alloy 600 and Type 304 Stainless Steel, cobalt release from Alloy 600 and cobalt-60 pickup by Alloy 600 and Stainless Steel.

Comments: The methods used were to create an out-reactor loop in which PWR conditions were reproduced as closely as possible. The temperature was 300°C, dissolved hydrogen was 18 cm^3/kg, dissolved zinc was between 0 and 40 μg/kg and coordinated boron/lithium chemistry was used to obtain pH300 in the 6.9 to 7.5 range. Radiotracing, comprising on-line monitoring of release, and on-line and coupon measurements of cobalt-60 pickup was used. Detailed examination of surfaces was carried out by means of SEM, AES and EDX techniques.

Remarks/Potential for dose limitation: For Alloy 600 it was found that zinc in simulated PWR coolant:

- produces thinner oxide films
- reduces cobalt-60 pickup by factors of 8 to 10
- had no apparent effect on corrosion release (deemed anomalous).

For Type 304 Stainless Steel zinc addition:

- produces thinner and more protective oxide
- reduces cobalt-60 pickup by factor of 8 to 10
- reduces pickup by inhibiting incorporation of cobalt in growing oxide.

Initial loop experiments measuring the effect of zinc additions on corrosion release from key PWR construction materials and subsequent cobalt-60 pickup by these materials indicate that this technique has substantial potential for PWR plants.

CANADA R-309

References: 1. "The Effect of Dissolved Zinc on the Transport of Corrosion Products in PWRs", EPRI NP-6975-D. Final Report, September 1990. 2. Lister, D., "Loop Experiments on Zinc Injection Under PWR Conditions", Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: N/A

Status: Completed

Last Update: April 16, 1992

CORROSION CONTROL AND DOSE RATE REDUCTION

Keywords: CONTAMINATION PREVENTION; ZINC ADDITION; PWR; PWSCC; COBALT; RADIATION BUILDUP; DOSE RATE

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Objectives: Westinghouse has conducted tests to evaluate the impact of zinc addition on PWSCC in simulated PWR conditions. The tests showed significant benefits from zinc addition. For example, the tests showed a significant delay in mean PWSCC initiation time, and microanalysis of oxide films confirmed that zinc was contained in the passivating film. Moreover, the possibility for dose rate reduction appears likely. A phase 2 program has been initiated and will be completed in 1992. The following tasks are to be carried out:

- Evaluation of the chemistry of zinc addition for PWR systems, including the determination of the solubility under selected conditions.
- Determination of equilibrium film composition.
- Stability of zinc containing films.
- Electrochemical characteristics.
- Microanalytical characterization.
- Crack growth rate as a function of zinc addition.
- Reverse U-bend testing.
- Activity reduction mechanisms.

Comments: Part of the phase 2 program will be devoted to obtaining a practical understanding of the mechanism and effects of zinc addition on radio-cobalt and zinc-65 transport and deposition. Tests and field applications will be followed and evaluated. Zinc data from nuclear power plants will also be evaluated and estimates of zinc addition on PWR exposure rates will be made.

Remarks/Potential for dose limitation: One aspect of the phase 2 program will be to identify system wide issues that must be addressed prior to field application of zinc on the primary side. An action plan will be developed to resolve any issues. Thus, system, radiological, and safety issues will be identified through contact with other Westinghouse divisions and industry sources. After these issues have been identified, the specific tasks required to address each issue will be selected and incorporated into a phase 3 proposal, should the results of phase 2 appear to be promising.

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U.S.A. R-310

References: Esposito, J.N., C.A. Bergmann and R.J. Jacko, "Zinc Addition for Corrosion Control and Dose Rate Reduction", Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: N/A

Status: In progress

Last Update: July 15, 1991

EFFECTS OF PH AND LI ON PWSCC INITIATION AND GROWTH

Keywords: COMPONENT RELIABILITY; WATER CHEMISTRY; STRESS CORROSION CRACKING; PWSCC; ALLOY 600; PWR; LITHIUM; PH

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Objectives: Increases in lithium concentration in primary water are considered as a means of reducing PWR radiation levels and many utilities in the U.S. have considered adopting this approach. However water chemistry can affect both general corrosion and primary water stress corrosion cracking. The objective of the present prc am was to determine whether significantly different PWSCC initiation and growth rates are observed in higher Li primary water chemistries.

Comments: Previous PWSCC tests had indicated that the influence of lithium on PWSCC behaviour of Alloy 600 was a function of temperature. For an increase in Li concentration from 2 to 7 ppm, Westinghouse results showed a rate increase by a factor of 3 at 360°C but no increase at 330°C. On th other hand, Swedish results, showed a factor of 2 increase at 330°C for an increase in Li concentration from 2.4 to 3.5 ppm. Because of these conflicting results a new test program was initiated.

Remarks/Potential for dose limitation: The test results may be summarized as follows:

- For high susceptibility RUBs under high stress no effect of Lithium was observed.
- For high susceptibility RUBs under intermediate stress the effect of lithium was insignificant.
- For high susceptibility RUBs under low stress (where it took longer for PWSCC to be initiated) there was more and faster initiation in 3.5 ppm Li solutions compared to 2.2 and 0.66 ppm lithium.
- For thermally treated, lower susceptibility RUBs of alloy 600, under high stress there was more and faster initiation of PWSCC in 3.5 ppm Li compared to 2.2 or 0.66 ppm Li. The environment that indicated the least PWSCC contained 315 ppm Boron and 2.2 ppm Lithium with a pH300C of 7.4. It was also found that the lithium concentration appears to make more of a difference in PWSCC than pH.

References: 1. "Effect of Lithium Hydroxide on Primary Water Stress Corrosion Cracking of Alloy 600 Tubing", EPRI NP-7396, Final Report, September 1991. 2. Jacko, R.J., "Effects of pH and Li on PWSCC Initiation and Growth", Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: N/A

Status: Completed

Last Update: April 16, 1992

RADIOACTIVITY PICK-UP BY CARBON STEEL AND STAINLESS STEEL IN SLIGHTLY OXIDIZING LITHIATED COOLANT

Keywords: CONTAMINATION PREVENTION; CARBON STEEL; STAINLESS STEEL; COOLANT; RADIATION BUILDUP

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Objectives: Though still low, Co-60 fields at the CANDU type Bruce-A and B nuclear power plants are rising faster than predicted. The increase seems to start when the reactor is uprated from 88% to 100% power. The uprating puts many fuel channels into boiling, which seems to produce oxidizing coolant. An appropriate question is whether radiolytic oxygen is responsible for the increased activity transport.

The program objectives were to determine:

- How oxide films on carbon steel and type 403 stainless steel change when oxygen is added to the coolant.
- How Co-60 pick-up on carbon steel and stainless steel is affected by oxygen.

Comments: The experimental method was as follows:

- Expose coupons and sintered magnetite and haematite in the out-reactor loop.
- Characterize oxide phase changes as reducing/oxidizing nature of the coolant changes.
- Monitor pick-up of Co-60 on-line and with coupons.

A scoping study showed that:

- Mossbauer effect is effective for analyzing conversions between Fe3O4 and Fe2O3.
- Co-60 pick-up on 403 stainless steel is greater than on carbon steel.

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

- Type 403 stainless steel picks up more activity than carbon steel under all conditions.
- Mossbauer effect is a useful technique for characterizing oxides.
- Oxidation of Fe3O4 in 60ug/kg of oxygen is rapid (50% conversion in 50 h).
- Reduction of Fe2O3 in 17 cm^3/kg hydrogen is slow (10% conversion in 130 to 300h).

References: Lister, D. and M.S. Godin, "Radioactivity Pick-up by Carbon Steel in Slightly Oxidizing Lithiated Coolant", Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: CANDU Owners Group

Status: End December 1991

Last Update: December 16, 1991

LESSONS LEARNED FROM RECENT BWR CHEMICAL DECONTAMINATION APPLICATIONS

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; BWR; PERSON-REM REDUCTION; PARTIAL SYSTEM DECONTAMINATION

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Objectives: To realize economical exposure reduction, reduce high radiation fields, combined with a large amount of work in the dry well area.

Other objectives were to increase effectiveness, reduce unscheduled delays, and allow for contingency measures.

Comments: The lessons learned could be divided into three categories:

Organization and Planning

- Provide detailed specifications.
- Review materials for compatibility.
- Check equipment layout thoroughly.
- Carry out a nuclear safety evaluation.
- Allow maximum flexibility in procedures.
- Ensure that adequate technical support is available.
- Have daily coordination meetings.
- Identify the decision maker on behalf of the utility for each shift.

Decontamination Equipment

- Consider secondary containment penetrations.
- Flexible metal suction hose may fatigue.
- Calibrate instrumentation and prepare standards early.
- Coordinate contaminated equipment transportation.
- Provide adequate spare parts.
- Check new equipment early.
- Utility supplied material must meet design criteria.

Field Implementation

- Evaporation will reduce volume.
- Oxides "plugging" cracks may dissolve and create leaks.
- Sumps may require trace element analysis.
- Always add acid to base.
- Coordinate with all outage activities.
- Cleanup process water adequately.
- Provide adequate ion exchange flow.
- Maintain proper pH control.
- Provide extra resin capacity for in-leakage.
- Maintain solvent level below jet pump slip joints.
- [illegible]fully walkdown initial valve alignment.
- LOMI is effective on HWC oxides.
- Don't leave faulted LOMI in the system.

Remarks/Potential for dose limitation:

References: Vandergriff, D.M., "Overview of Recent Utility Experience with Part System Decontamination", Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991 **Funding:** N/A

Status: In progress **Last Update:** June 26, 1991

DECONTAMINATION OF BEAVER VALLEY STEAM GENERATORS USING THE CAN-DEREM PROCESS

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; STEAM GENERATOR; CAN-DEREM; CAN-DECON; RWCU; RHR; COBALT; ANTIMONY; NICKEL; IRON; CHROMIUM; STAINLESS STEEL; INCONEL

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Objectives: The objectives of this project were:

1. To carry out an assessment of the CAN-DEREM and CAN-DECON processes for the decontamination of the steam generators of the Beaver Valley Nuclear Power Plant.
2. To assess the results of the Beaver Valley decontamination and other recent developments.

The recent developments in work on the CAN-DECON and CAN-DEREM processes include a lowering of the temperature during decontamination from 120 °C to 90 °C; a lowering of the concentrations and utilization of weak base ion exchange resin, resulting in lower wastes; a program to increase the effectiveness of the AP step in the process; removal of antimony.

Comments: CAN-DECON utilizes citric acid, oxalic acid and EDTA and has been successfully used in 14 full-system with fuel decontaminations of CANDUs as well as 19 BWR and 5 PWR subsystem decontaminations. CAN-DEREM, which utilizes only the citric acid and EDTA steps, is less corrosive.

Predecontamination assessment involved characterizing Beaver Valley artifacts to decide whether to use CAN-DECON or CAN-DEREM. Having decided on CAN-DEREM, to decide on the number of AP/CAN-DEREM cycles, decide on the optimum % by weight of AP and the optimum time for the AP and CAN-DEREM steps.

It was found that for the first step, CAN-DEREM dissolves most of the oxide within 8 hours, after which there is no further dissolution. CAN-DECON reaches the same effectiveness after 20 hours, but then continues to dissolve further layers of oxide. The effect on dissolution of the second CAN-DEREM step is even quicker.

Remarks/Potential for dose limitation: In summary, CAN-DECON/CAN-DEREM dilute regenerative technology has been well established, with 18 years of experience. The CAN-DEREM process has been used in 7 subsystems, including RHE, CVCS, RHR, RWCU and steam generators at both PWR and BWR plants.

At Beaver Valley it was found to be effective, with decontamination factors of between 4 to 9, it produced negligible corrosion, low volumes of waste, and could be carried out with relatively short decontamination times of between 8 and 10 hours.

The process is presently being qualified by Westinghouse Electric Corporation for full-system decontamination of PWRs.

References: Speranzini, R.A., R. Voit and M. Helms, "Decontamination of Beaver Valley Steam Generators Using the CAN-DEREM Process", Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991 **Funding:** N/A

Status: In progress **Last Update:** June 26, 1991

PWR FULL REACTOR COOLANT SYSTEM CHEMICAL DECONTAMINATION

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

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Objectives: The potential long-term benefits of full primary system chemical decontamination are significant for pressurized water reactors. However, a detailed qualification of the various decontamination processes available is required, both with the fuel removed and the fuel in place. This program is underway and includes both Westinghouse and Combustion Engineering designed reactors. It has been divided into three phases. The second phase of the program is now complete. The objectives of the second phase were:

1. To qualify the chemical decontamination, through the AP/CAN-DEREM and AP/LOMI processes, of the full primary system of Westinghouse and Combustion Engineering designed pressurized water reactors, with the fuel removed.
2. To complete a fuel decontamination qualification program.

Comments: The following tasks were completed:

1. Material corrosion and compatibility test program.
2. Fluid systems evaluations.
3. RCS equipment evaluations.
4. Waste management.
5. Radiological evaluations.
6. Nuclear safety evaluations.
7. Development of decontamination technology; cost and schedule estimates; design of equipment.

The main highlights were:

1. AP/CAN-DEREM and AP/LOMI were qualified for three applications.
2. No prefilming of specimens was carried out.
3. Particulate testing and wear and friction tests were completed for both processes.
4. Off-normal tests were conducted for the two processes.
5. Processes were qualified at boron concentrations ranging from 0 to 650 ppm.

Remarks/Potential for dose limitation: The following is a part of the fuel decontamination program:

1. General corrosion data for Zircaloy-4 and Zirlo were obtained from the current program.
2. Fuel decontamination demonstration is to be performed on four assemblies, two using CAN-DEREM and two using LOMI.
3. The four assemblies will then be reinstalled in the VC Summer nuclear plant for one more cycle and then inspected.

The main conclusions of the study so far are:

1. Results are extremely positive.
2. Minimal effects of chemical reagents on most primary system materials and components.
3. Certain pre- and post-decontamination inspections and modifications are recommended.
4. Potential benefits are significant.
5. Qualification without fuel is now complete; qualification with fuel will be completed in early 1993.

References: Miller, P., "PWR Full Primary System Decontamination Program", Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, Californ'a, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991 **Funding:** N/A

Status: In progress **Last Update:** November 18, 1991

BWR FULL-SYSTEM DECONTAMINATION

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; BWR; FULL-SYSTEM DECONTAMINATION; LOMI; AP/LOMI

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Objectives: The objectives of this program are to evaluate the feasibility of full-system decontamination of a BWR with specific emphasis on:

1. Long term materials performance.
2. Decontamination process and system interaction.
3. Radwaste.
4. Recontamination.
5. Economics.
6. Licensing.
7. Fuel evaluation.

Some of the specific full-system decontamination issues being addressed are:

1. Corrosion during the decontamination itself.
2. Post-decontamination corrosion.
3. The effects of decontamination on non-mettalics.

Comments: The qualification process was divided into a number of phases. The various phases of the program were:

- A description of the LOMI process as applicable to BWRs: during this work functional process guidelines were identified and sampling and analytical techniques developed.
- Evaluation of materials compatibility: conclusions were that LOMI is acceptable, AP/LOMI appears to be promising but NP/LOMI is unacceptable for full-system decontamination.
- Development of decontamination equipment and analysis of system interaction: e.g. decontamination engineering background and system boundaries were formulated; operating conditions for the decon of RPV, RHR, Recirc, RWCU, CRD were established.

- Previous experience was examined and cost benefit analysis was carried out: FSD and recontamination experience of relevant types of reactors which had undergone FSD was examined, recontamination mitigation techniques were evaluated, exposure and critical path savings versus waste and other costs were considered.
- Shielding and exposure related factors were examined.

Remarks/Potential for dose limitation: A great deal of the work has already been completed. A topical report is to summarize the following:

1. A generic description of the LOMI process as qualified for FSD of BWRs.
2. Review of previous BWR LOMI experience.
3. Technical support of all calculations and assumptions.

A plant specific report will deal with the following topics:

- Identification of differences between generic topical and plant specific differences. It will deal with such items as metallurgical differences, chemical matters, etc.
- Evaluation of the effect of such differences on potential applications.

References: Gordon, B.M., "BWR Full System Decontamination", Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1989 to: 1992

Funding: EPRI and Commonwealth Edison Company (CECO)

Status: In progress

Last Update: December 16, 1991

PWR COOLANT CHEMISTRY STUDIES IN SUPPORT OF DOSE REDUCTION USING IN-PILE LOOPS AT MIT

Keywords: CONTAMINATION PREVENTION; COMPONENT RELIABILITY; CRUD; PWR; BORON; LITHIUM; PH; MATERIAL; CRUD ANALYSIS; GAMMA SCAN

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Objectives: The objectives of this work are:

1. To design, construct and operate an in-pile facility which closely simulates the PWR Primary Coolant System with respect to Crud transport, temperatures, gradients, materials, etc.
2. To carry out a series of experiments on the effect of coolant pH on corrosion product activity buildup.

Comments: The experimental program consisted of the following:

1. Seven one-month-long in-pile runs preceded by 2 months of out-of-pile pretreatment under primary coolant conditions.
2. Principal runs examined effect of pH (LiOH) at constant Boron (800 ppm B) for pH300°C = 6.5, 7.0, 7.2, 7.5.
3. Loop sacrificed for post-run examination, such as gamma scans and crud analysis.

The major findings were as follows:

1. The solubility based picture of corrosion product transport was supported by the tests.
2. A value below 6.5 for pH300°C was found to be very detrimental. Values of 7.2 to 7.5 showed appreciably less activity buildup.
3. The pH and heat flux effect on surface crud morphology correlated with differences in activity buildup.

Remarks/Potential for dose limitation: Future work will include:

1. A series of three long (3,000 hr) runs to investigate the effect of pH on activity buildup.
2. Discussions of a zinc injection test and possible implementation.
3. Preliminary consideration of surface treatment evaluation.

U.S.A. R-317

References: 1. Driscoll, M.J., G.E. Kohse and O.K. Harling, "PWR Coolant Chemistry Studies in Support of Dose Reduction Using In-Pile Loops at MIT", Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken. 2. EPRI TR-100156, report, December 1991.

Duration: from: 1990 to: 1991

Funding N/A

Status: Completed

Last Update: April 17, 1992

U.S.A./JAPAN R-318

SOLUBILITY MEASUREMENT OF CRUD AND EVALUATION OF OPTIMUM PH

Keywords: CONTAMINATION PREVENTION; CRUD; PH; SOLUBILITY; NICKEL; COBALT; PWR

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Objectives: To derive the optimum pH for Pressurized Water Reactor from measurement of the solubility of crud.

Comments: It is well known that occupational radiation exposures may be significantly reduced by decreasing the amount of crud deposited in the primary circuit. It is also known that radioactive crud is produced by the activation of corrosion products which are then transferred to out-of-core surfaces. Since out-of-core dose rates are mainly determined by Co-58 and Co-60, the preferred value of the pH should be related to the solubility of the nickel and cobalt precursors. Therefore the solubility of nickel ferrite was measured, and the most effective pH value was derived from the solubility data.

Remarks/Potential for dose limitation: In order to reduce dose rates, it is important to minimize the transfer of crud from inside the core to out-of-core surfaces. From the various relationships derived, a relationship was established between the pH and the dose rate. It was determined that the optimum pH for minimum dose rate was around 7.3 at 285°C.

References: Shoda Y., et al., "Solubility Measurement of Crud and Evaluation of Optimum pH", Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: N/A

Status: In progress

Last Update: June 26, 1991

FULL REACTOR COOLANT SYSTEM (RCS) DECONTAMINATION NATIONAL DEMONSTRATION PLAN

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

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Objectives: Since 1988, the Electric Power Research Institute and 11 utility owners of pressurized water reactors have been sponsoring a program to determine the technical acceptability of dilute chemical solvent processes for primary system decontamination. Two processes, AP/CAN-DEREM and AP/LOMI, are being qualified for use in the entire reactor coolant system of a PWR. The goal of this program is to demonstrate, for the first time in the United States, full reactor coolant system decontamination of operating nuclear power plants.

Comments: Based on experience with subsystem decontamination and the findings of this study, a full-system decontamination without fuel should be effective over five cycles. This period could be extended if FSD is combined with improvements in plant chemistry and a reduction in cobalt bearing alloys. Some of the potential benefits are:

- Greater ability of the decontaminated plant to obtain skilled personnel.
- Extension in the ability of the utility to respond to crisis situations.
- Smaller, more experienced, and more productive crew sizes.
- Lower contamination levels leading to less cleanup manpower.
- Longer stay times for crews.
- Significant improvements in productivity, and lower operating costs.
- Potential decrease in the cost of nuclear liability insurance.
- An improvement in the public's perception of nuclear power, in general, and the utility, in particular.

Remarks/Potential for dose limitation: EPRI is attempting to form a consortium of utilities and other organizations to sponsor a national demonstration. Participants will receive proprietary results of the qualification process which will form an important basis for design for FSD. Participants will also be provided specifications, procedures, safety evaluations from the demonstration. These documents will be largely generic and help the participating utilities to considerably reduce their own costs for FSD.

EPRI will also form an advisory group to gain direct experience and information from the demonstration. Participants will be given the opportunity to obtain "hands on" training for their personnel through the EPRI "loan engineer" program. A training course on FSD will be given to educate personnel in many aspects of the complex technology.

References: 1) Parry, J.O., and S.A. Trovato, "Plans for a Full System Decontamination National Demonstration", Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken. 2) "Full Reactor System Decontamination: Long Term Radiological Evaluation," NATD/PA/155/90, Westinghouse Electric Corporation, June 1990. 3) Proceedings of the EPRI Seminar on PWR Radiation Field Control, Berkeley, California, March 1988, ed. C.J. Wood.

Duration: from: 1990 to: 1991

Funding: N/A

Status: In progress

Last Update: June 26, 1991

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FULL-SYSTEM DECONTAMINATION OF THE BR-3 PLANT

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

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Objectives: The BR-3 nuclear plant was a Westinghouse designed PWR, rated at 10.5 MWe. It was a 1.5 loop unit with stainless steel U-tube steam generators. The plant was in operation from October 1962 to June 1987. It was the first PWR to be dismantled in Western Europe and was selected as a pilot project for decommissioning by the Commission of European Communities. CEN/SCK, the owner of the plant, contracted Siemens AG KWU to carry out the full-system decontamination of the plant.

The incentives for full-system decontamination were:

- Plant decommissioning was to be carried out under new and lower radiation exposure limits.
- Major work was required.
- Several options would be available for decommissioning.

An objective of this project was to minimize radioactive waste as far as possible.

Comments: The CORD (Chemical Oxidation Reducing Decontamination) process, developed by Siemens AG KWU, was selected for the decontamination based on the following arguments:

- It has a higher oxidation potential.
- A lower chemical concentration is required (< 0.2 Wt %).
- Treatment time is shorter.
- Decontamination factors are higher.
- Radioactive wastes are smaller.
- No intermediate cleanup is required.
- No flushing is required.
- No cooling or heating up of the solutions are required during the first cycle (temperature of application is 95°C).
- Continuous operation is possible.

Partial decontamination experience with the process included 200 main coolant pumps, 4 regenerative heat exchangers, and 9 subsystem decontaminations.
Full-system decontamination was initiated after a plant specific qualification program.

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Remarks/Potential for dose limitation: The findings may be summarized as follows:

- Objectives of full-system decontamination were fully met at BR-3 and the efficacy of the CORD process for FSD was demonstrated.
- Activity release was higher than estimated; reasons for this are still speculative.
- Cleanup capacity was small, resulting in long purification times.
- A high radiation field reduction was achieved.
- Release of Fe/Cr/Ni was higher than estimated.
- Available resin capacity limited the application of further cycles of CORD.

References: 1) Wille, H. R. Riess and F. Motte, "Future Developments in Processing Decontamination Waste", Proceedings, EPRI Seminar on Radiation Field Control, Palo Alto, California, April 9-11, 1991, Editors C.J. Wood and H. Ocken, Research Report Center, Box 50490, Palo Alto, CA 94303. 2) Riess, R., "German Work on Full-system Decontamination and BR-3 Experience", Proceedings, EPRI Workshop on Full-System Decontamination, Charlotte, North Carolina, June 4-5, 1991, Editor C.J. Wood, Research Report Center, Box 50490, Palo Alto, CA 94303, U.S.A.

Duration: from: 1980 to: 1991

Funding: N/A

Status: Complete

Last Update: June 26, 1991

FUTURE DEVELOPMENTS IN PROCESSING DECONTAMINATION WASTE

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; RESIN OXIDATION; ELOMIX; WASTE HANDLING; WASTE VOLUME REDUCTION

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Objectives: Conventional decontamination waste handling processes are satisfactory for sub-system decontamination, but inadequate for full-system decontamination. They are based on evaporation/precipitation or ion exchange. The problems with the current processes usually are:

- Excessive volume, both of the waste and the handling equipment.
- Their chelant content.
- Their radiation stability.
- The problems with cementation.

The objectives of the present program are:

- To reduce waste volume by removing unwanted components from the waste.
- While doing so, to avoid changes to qualified decontamination processes.

Comments: Two techniques are under development: (a) ELOMIX, (b) Resin Oxidation.

The purpose of the ELOMIX process is:

- Reduce waste volumes from LOMI decontamination.
- Reduce chelant loadings in LOMI waste.
- Reduce chemical requirements.
- Separate radioactivity from organic substances.
- Allow storage and radioactive decay prior to burial.

The purpose of the resin oxidation process is:

- Use low temperature wet oxidation to destroy chelants and resin.
- Achieve better waste form, no chelant content, and volume reduction.

Remarks/Potential for dose limitation: Current status of the ELOMIX process is as follows:

- Extensive laboratory work has been completed.
- Successful on-site trial was carried out at Dresden.
- A scaled up second phase is under way.

Current status of the resin oxidation process is:

- 40 cubic feet of resin has been processed in the trials.
- NRC topical report tests are under way.
- The next stage will be active trials.

References: Bradb. y, D., "Future Developments in Processing Decontamination Waste", Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991 **Funding:** N/A

Status: In progress **Last Update:** December 17, 1991

REDUCTION OF CRITICAL PATH TIME FOR BWR RECIRCULATION SYSTEM DECONTAMINATIONS

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; BWR; RECIRCULATION SYSTEM; RWR; LOMI

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Objectives: To identify methods for reducing the critical path time required to perform chemical decontamination of the reactor water circulation system in a BWR by examining actual tasks that affect critical path during decontaminations and identifying time reducing approaches for each task.

The main tasks considered were:

Task 1. Draindown of systems and preparation of the Reactor Water Recirculation System (RWR) for the decontamination.

Task 2. Performance of the RWR decontamination.

Task 3. Performance of reagent removal.

Task 4. Return of RWR to normal status.

Comments: Among the suggested improvements were the following:

Task 1: Connect decon equipment to system prior to drain down and use decontamination equipment to expedite draindown.

Task 2: Adjust the flowpath to perform a single phase application without permanent system modifications; perform injection of LOMI chemicals simultaneously and tailor injection rates to the specifics of the system.

Task 3: Increase the ion exchange processing rate and use a drain and refill technique, rather than a recirculating one, for final polishing.

Task 4: Initiate as much restoration of the RWR as possible in parallel with the completion of task 3, reagent removal. Also loosen restrictions on drywell access as soon as possible after the initiation of task 3. Usually this can start after 3 or 4 purification half lives.

Remarks/Potential for dose limitation: The main conclusions of this program were:

- Significant critical path reductions can be achieved in performing RWR decontaminations.
- The level of reduction will vary according to the amount of effort employed.

- In absolute terms, the potential savings have been estimated at between 14 and 66 hours. As a percentage, these represent savings between 14% and 60% based upon an average current duration of 110 hours.

References: Beaman T.A., "Reduction of Critical Path Time for BWR Recirculation System Decontamination", Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: N/A

Status: Completed

Last Update: June 26, 1991

IMPROVEMENTS IN THE LOMI DECONTAMINATION PROCESS

Keywords: CONTAMINATION REMOVAL; LOMI; DECONTAMINATION

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Objectives: There have been a number of improvements and modifications to the LOMI decontamination process. These modifications have involved elimination of requirement for nitrogen purging of the annulus during RRS decons, leading to simpler operations; reductions in the reagent margin from 100% to 50%, and in the ratio of picolinic acid to vanadium from 6:1 to 4.5:1, resulting in reduced costs and less waste. Other improvements have been the introduction of Ionic A365 for cleanup and utilization of low formate vanadous formate to further reduce the waste volume.

The objective of the present program was to examine, evaluate and develop additional improvements and modifications to the LOMI process.

Comments: The improvements presently being developed are:

- Qualification for full-system decontamination in BWR and PWR plants with fuel removed. This should improve DFs in all areas.
- Further reduction in the picolinic acid to vanadium ratio to 3:1. This will further reduce cost and reduce wastes.
- Application of the process in the presence of boron, which has potential for full-system decontamination with the fuel in place.
- Evaluating other suppliers for vanadous formate, which should increase competitiveness, lower material costs and increase the reliability of the supply of the formate.
- Removal of non-ionizable impurities from the vanadous formate, which should reduce the TOC problem during the final cleanup.

Remarks/Potential for dose limitation: Many new developments are taking place in decontamination technology. Among these are:

For LOMI:

- Reduction of picolinic acid ratio to 3:1.
- Development of the ELOMIX process.
- Development of the resin oxidation process.

These developments should drastically reduce the wastes from the LOMI process.

For CAN-DECON/CAN-DEREM:

- Removal of oxalic acid.
- Use of A365 resin.
- Use of a sulphur free inhibitor.

Both processes are being qualified for full-system decontamination.

References: 1. Smee, J.L. and D. Bradbury, "Improvements in the LOMI Decontamination Process", Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken. 2. "Reformulation of the LOMI Chemical Decontamination Reagent", EPRI NP-7276, Final Report, May 1991.

Duration: from: 1990 to: 1991

Funding: N/A

Status: Completed

Last Update: April 16, 1992

RADIATION FIELDS TRENDS AND CONTROL AT FRENCH PWR'S

Keywords: CONTAMINATION PREVENTION; RADIATION FIELD; PWR; CHEMISTRY; COBALT REDUCTION; DOSE RATE

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Objectives: The objectives of this program are to monitor the dose rate and collective dose trends at French PWRs and develop improved dose rate and radiation field control measures.

Comments: The methodology consists of three principal steps:

1. Predict sources and dose rates - deposited activity is predicted by means of the PACTOLE code, dose rate calculations are carried out by means of MERCURE 4 and increasingly through PANTHERE VI codes.
2. Measure sources and dose rates - dose rates are measured systematically at all plants through the "Standard Radiation Monitoring Program". These measurements are made between 12 and 24 hours after plant shutdown. At some plants, corrosion product activity deposits are measured using gamma ray spectroscopy with a special device called "EMECC". The technique has been developed by CEA and qualified at EDF NPPs.
3. Analyze any discrepancies between the predictions and the actual measurements - the analysis is carried out by means of the TIGRE-RP software developed by EDF.

From this data, the influence of various parameters, such as primary coolant chemistry and shutdown procedures, may be evaluated on the development of radiation fields.

Remarks/Potential for dose limitation: At the present time, the average dose rate near the primary loops is 1 person-Sv/h for the six oldest plants, and 0.5 person-Sv/h for other units. The average dose rate at the center of steam generator channel heads is 90 person-Sv/h for older plants, and 60 person-Sv/h for others. Further dose rate reduction should be possible for future plants. Present efforts are directed towards reductions in collective dose and meeting the recommendations of the ICRP.

The same methodology will be applied for collective dose reduction as for dose rate reduction. The use of ALARA techniques will be reinforced. These techniques will range from simple ones, such as maintaining ALARA sheets, to sophisticated ones that have been tested during the replacement of the steam generators at Dampierre-1.

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References: Ridoux, P., and A. Brissaud, "Radiation Fields Trends and Control at French PWR's", Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: N/A

Status: In progress

Last Update: November 21, 1991

WELDABILITY OF NOREM FOR IN-SITU REPAIR & REPLACEMENT

Keywords: CONTAMINATION PREVENTION; NOREM; REPAIR; WELDING

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Objectives: The objectives of this program were as follows:

1. Develop a cobalt-free hardfacing alloy which can be used for in-situ repair/replacement of valve seats and other similar components.
2. Demonstrate the ability to deposit the NOREM alloy on applicable austenitic stainless and low alloy steel substrates.
3. Develop welding parameters which require minimal preheat and post weld heat controls.
4. Demonstrate an all position/in-situ application utilizing a commercially available GTAW-AU welding system.
5. Demonstrate the ability to repair a NOREM alloy deposit.

Comments: The main characteristics that have an influence on parameter development for NOREM are:

- Crack free deposit
- Bead appearance
- Hardness
- Deposit chemistry

These were all investigated during the study.

Material testing for the NOREM weldability program was carried out by examining the following:

- Galling wear
- Mechanical wear
- Chemical analysis

Remarks/Potential for dose limitation: The main findings of the program were:

- Welding filler metal wires conforming to NOREM B1 and B4 compositions can be produced in small diameter suitable for GTAW equipment (0.045).

- NOREM B1 can be deposited on stainless steel with no pre-heat yielding acceptable results.
- A two-layer NOREM B1 deposit yields galling wear results comparable to Stellite 6.
- NOREM can be deposited in-situ, in all positions utilizing commercial GTAW-AU equipment.

References: Vandergriff, D.M., M. Philips and S. Findlan, "Weldability of NOREM for In-Situ Repair & Replacement", Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: N/A

Status: In progress

Last Update: November 22, 1991

HIGH PH OPERATION AT SWEDISH PRESSURIZED WATER REACTORS

Keywords: CONTAMINATION PREVENTION; PH; PWR; CHEMISTRY; WATER CHEMISTRY; STEAM GENERATOR; COBALT; ALLOY 600; ALLOY 690

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Objectives: Heavy crudding of some fuel assemblies and increasing dose rates lead to the use of high pH chemistry (pH300°C of up to 7.4) at Ringhals 2 NPP in the early 1980s. High pH chemistry was also subsequently utilized at Ringhals 3 and 4, resulting in low dose rates at these plants. The somewhat adverse results of recent crack initiation studies in alloy 600, together with concerns about enhanced fuel cladding oxidation has lead to a slightly more conservative approach.

The objective of the present work was to arrive at an optimum value of pH300°C for the Ringhals 2, 3 and 4 primary coolant which would give the best results for dose rate reduction, mitigation of stress corrosion cracking in alloy 600 steam generator tubing and prevention of hydriding in guide tubes.

Comments: The original Ringhals 2 steam generators, which had mill-annealed alloy 600 tubes, have been replaced by new steam generators with the more robust thermally treated alloy 690 tubing and the power of Ringhals 2 has been uprated. Therefore, operating this plant with higher lithium concentrations than 2.2 ppm should have been possible. However, cladding material problems in a bad fuel batch, together with hydriding of guide tubes, has necessitated restricting lithium to 2.2 ppm. All three Ringhals plants now start their cycles at the maximum lithium concentration and a pH300°C of 6.9. They then proceed until a pH300°C of 7.24 is established.

The dose rates have been low at all three plants since the introduction of high pH chemistry, with the dose rates being lowest at Ringhals 4. Another factor contributing to the low dose rates is the substitution of zircaloy fuel grid spacers which are free of cobalt content.

Remarks/Potential for dose limitation: Our data support that high pH chemistry seems to promote low dose rates, and that the introduction of this type of chemistry early in the life of the plant is advantageous. High pH chemistry also seems to result in low fuel crudding.

BNL ALARA Center Data Base

SWEDEN R-326

References: Aronsson, P.O., P. Andersson, S. Duniec and O. Erixon, "High pH Operation at Swedish PWR's", Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: N/A

Status: In progress

Last Update: June 27, 1991

RADIATION FIELD CONTROL BY EARLY BORATION DURING SHUTDOWN AT BEAVER VALLEY POWER STATION

Keywords: CONTAMINATION PREVENTION; RADIATION BUILDUP; BORATION; RADIATION FIELD

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Objectives: Beaver Valley is a two unit power station. During shutdown for the seventh refueling outage at Unit 1 in 1989, plant problems forced operations personnel to borate to greater than 2,000 ppm boron with the Reactor Coolant System temperature above 300°F. A large release of soluble cobalt 58 and cobalt 60 occurred, resulting in reduced out-of-core radiation fields by as much as 20%. Moreover, over three times as much radiocobalt was removed as had been during previous shutdowns. The Co-58 to Co-60 ratio was also significantly different from the cobalt ratios observed during previous shutdowns and hydrogen peroxide additions, indicating that the source of the radiocobalt was outside the core.

The objective of this program is to evaluate the extent of the impact of early boration on radiation fields and to determine the optimum conditions for early boration.

Comments: The early boration resulted because there was an increase in the monitoring requirements due to a crudburst occurring during the cooldown of Unit 2, and a failure of instrumentation during the cooldown of Unit 1.

The early boration of Unit 1 lead to a large release of soluble activity, which was 100 times higher than previous shutdowns. This activity was preponderantly Co-58 activity. There was no crud burst.

Substantial reductions in out-of-core radiation fields were noted. For example, the NR heat exchanger hotspot dose rate came down from 7,000 mR/h to 50 mR/h; the NR heat exchanger general area dose rates were reduced to 25 mR/h from 1,200 mR/h, and the sample panel hotspot dose rate was decreased to 15 mR/h from 300 mR/h. Steam generator channel head dose rates were reduced by 20%.

Subsequent peroxide addition gave a small additional release.

Remarks/Potential for dose limitation: A new test of early boration is to be carried out with the following measures in place:

- Early boration.
- Lithium removal enhanced.

- Enhanced radiation field monitoring.
- Chemistry monitoring to include metals.

The Beaver Valley data will then be compared with data from other plants.

References: Linnenbom, V.J., F.P. Liptak and K.J. Winter, "Radiation Field Control by Early Boration during Shutdown at Beaver Valley Power Station", Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: N/A

Status: In progress

Last Update: December 19, 1991

HIGH PH OPERATION IN ABB COMBUSTION ENGINEERING PLANTS

Keywords: CONTAMINATION PREVENTION; PH; WATER CHEMISTRY; CHEMISTRY; DOSE RATE; PWR

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Objectives: Two pressurized water reactors of Combustion Engineering design, Calvert Cliffs and St. Lucie, have been selected for trials with high pH chemistry. The reasons for the trial include positive results of laboratory studies, considerable success in the nuclear industry with pH of 6.9, and the pioneering work of the Biblis and Ringhals nuclear plants with higher pH values.

Comments: Prior to the introduction of elevated lithium chemistry, the experience of Calvert Cliffs was as follows:

- Dose rates in steam generator channel heads were 5 R/h.
- Shutdown oxygenation crud burst was no longer seen.
- The Reactor Coolant System was very clean.

Elevated RCS lithium chemistry was implemented in Unit 1 near the end of the 9th cycle and during the middle of the 8th cycle in Unit 2.

There were some concerns regarding fuel oxides, since oxides on zircaloy fuel cladding have been generally on the high side of normal for several plants:

- Oxides on Calvert Cliffs fuel cladding were similar to what they were during operation with normal lithium.
- Calvert Cliffs Unit 2 had limited exposure to 3.5 ppm lithium.

Remarks/Potential for dose limitation: Summarizing Calvert Cliffs experience:

- Unit 2 channel head dose rates returned to end of cycle 6 levels.
- The dose avoided was estimated to be 10%.
- Experience could be compared to other mature plants on high lithium chemistry.
- There were no adverse effects.

Summarizing Saint Lucie experience:

- There was a slight reduction in dose rates in the channel head in both units.
- Experience was comparable to other mature plants.
- There were no adverse side effects.

References: Beineke, T.A., P. Crinigan and A. Gould, "High pH Operation in ABB C-E Plants", Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: N/A

Status: In progress

Last Update: June 27, 1991

U.S.A. R-329

REACTOR COOLANT SYSTEM SHUTDOWN CHEMISTRY AND NICKEL MANAGEMENT AT H.B. ROBINSON NUCLEAR PROJECT

Keywords: CONTAMINATION REMOVAL; CONTAMINATION PREVENTION; PWR; RADIATION FIELD; DOSE RATE; SHUTDOWN CHEMISTRY; NICKEL CONTROL; LITHIUM; PH CONTROL; STEAM GENERATOR

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Objectives: The objectives of this program were to achieve conditions necessary for the decomposition and solubilization of activated RCS corrosion products, thereby avoiding crud bursts (high suspended solids) that may lead to redeposition rather than removal by ion exchange. The control of specific chemistry conditions during plant shutdown for refueling are expected to result, over time, in a steady decline in the steam generator channel head dose rates.

Comments: Japanese experience had shown that, in addition to at-power chemistry control, additional chemistry control during plant shutdown would aid in removing nickel and activated corrosion products in the primary system. This shutdown chemistry consisted of: (a) lowering the pH early during plant shutdown and adjusting coolant temperature to increase the solubility of activated corrosion products, (b) operating one or more reactor coolant pumps to increase solubilization, (c) operating the letdown purification at maximum flow rate, and (d) extending the purification period to increase the activated corrosion product removal. The data also showed that a significant portion of the release occurred before the primary system was oxygenated. A steady decline in steam generator dose rates resulted from this program. Analysis by the consultant indicated there would be beneficial results from a two-phase program at H.B. Robinson. A reducing phase using hydrogen should cause nickel-ferite oxide film to decompose, releasing iron and nickel. The oxidizing phase would create a second release of soluble nickel, free from suspended iron.

Remarks/Potential for dose limitation: Analysis of RCS data trends and the letdown removal of the isotopes and elements tracked during the shutdown indicated:

1. A total of 288 curies of activity was removed. The relatively low amount of total activity removed was probably due to the beneficial effects of 3.35 ppm lithium program which was being simultaneously used for operational chemistry.

2. More activity was released, solubilized and removed during the reducing phase (71%) compared to the oxidizing phase (29%).
3. A sustained release and removal of nickel occurred during the reducing phase. A return and efficient cleanup of soluble iron also occurred during the reducing phase.
4. The majority of activity released (88%) was soluble. The total particiulate activity released was small (11%) during the reducing phase and 1% during oxidizing phase.
5. About 90% of the out-of-core activity was released and removed during the reducing phase.

References: Morgan, E.A., Fender, B.A., Neeley, W.M., "RCS Shutdown Chemistry During the 1990 Refueling Outage at H.B. Robinson," Proceedings, EPRI Conference on Radiation Field Control, Palo Alto, California, April 9-11, 1991, editors C.J. Wood and H. Ocken.

Duration: from: 1990 to: 1991

Funding: N/A

Status: Completed

Last Update: September 24, 1991

U.S.A. R-330

ZINC INJECTION AT MILLSTONE 1

Keywords: CONTAMINATION PREVENTION; ZINC INJECTION; ZINC ADDITION; MILLSTONE; CHEMISTRY

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Objectives: To control buildup of activated corrosion products such as Co-60 on reactor water recirculation piping.

Comments: The GEZIP application at Millstone 1 has led to the leveling out of the recirculation system pipe dose rates at values more than 50% less than predicted for recontamination without GEZIP. This success was offset by some problems. These concerns included:

1. The production and spread of Zn-65.
2. Zn-65 potential to get into the food chain.
3. Lack of sensitivity of in-plant monitorin, devices to Zn-65.
4. The extra radwaste curie loading from Zn-65.

Remarks/Potential for dose limitation:

References: None

Duration: from: 1987 to: 1991

Funding: N/A

Status: Ongoing

Last Update: September 19, 1991

THE EFFECT OF ZINC ON CORROSION AND DOSE RATE CONTROL

Keywords: CONTAMINATION PREVENTION; ZINC ADDITION; ZINC INJECTION; PWR COOLANT; DOSE REDUCTION; CORROSION CONTROL

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Objectives: To define the effect of zinc addition to a PWR primary coolant from the viewpoint of reduced PWSCC and corrosion product generation and transport.

Comments: Results to date show a beneficial effect of zinc addition because the PWSCC rate and general corrosion rates of materials are reduced in simulated laboratory studies.

Remarks/Potential for dose limitation: This appears to be significant by reducing dose rates and time needed to repair components.

References: J.N. Esposito, et al, "Zinc Addition for Enhanced Corrosion Resistance", paper presented at the Fifth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems & Water Reactors, Monterey, CA, August 25-29, 1991.

Duration: from: 1990 to: 1992

Funding: N/A

Status: In progress

Last Update: November 21, 1991

BNL ALARA Center Data Base

NETHERLANDS R-332

TRACKER: AN ABSOLUTE TUBE-POSITION DETECTION AND TUBE MARKING SYSTEM

Keywords: REMOTE SYSTEM; STEAM GENERATOR; REPAIR; MAINTENANCE; REMOTE MAINTENANCE

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Objectives: To develop an absolute tube position detection and tube-marking system called "TRACKER" for guidance to jumpers in nuclear steam generators during emergency shutdowns to minimize radiation exposure and avoid plug or tool positioning errors. Other objectives are:

- reduction of radiation exposure to jumpers engaged in emergency shutdown maintenance of steam generators, who have to touch and count tube ends to identify tube positions.

- minimizing the risk of extended down time due to plug positioning errors.

- offering an independent tool for verification of tool and plug locations to Q.A.

- offering a tool for remotely controlled visual inspection of the S.G. tubesheet with data and image storage capability.

Comments: A real-time CCD camera image of the tubesheet is linked to the S.G. tubesheet geometry stored in the computer memory. A remotely controlled light spot inside the S.G. provides the operator with the link to the I.D. of the illuminated tube. The same light spot guides the jumper to the tube wanted. Images can be recorded continuously as snapshot and can be printed.

Remarks/Potential for dose limitation: This system has significant potential for reducing radiation exposure during shutdown maintenance of steam generators.

References: None

Duration: from: 1989 to: 1992

Funding: $ 250,000

Status: Final stage, prototype ready

Last Update: April 13, 1992

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H-139	IDENTIFY ALL COBALT CONTRIBUTORS IN PNPS
H-140	EVALUATE HOT SPOTS ASSOCIATED WITH SPENT FUEL POOL SYSTEM
H-141	SURROGATE LASER DISC PLANT TOUR SYSTEM
H-142	MAINTAIN RADIOLOGICAL EVALUATION FACTORS
H-143	REPLACE FEEDWATER CONTROL VALVE TRIM WITH NON-COBALT DESIGN
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H-145	FUEL IMPROVEMENTS TO REDUCE COBALT SOURCE
H-146	ESTABLISH CHEMICAL DECONTAMINATION STRATEGY
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H-148	REACTOR CONTROL BLADE MANAGEMENT CONSIDERING ALARA
H-149	EVALUATION, POSSIBLE REDUCTION IN OPERATION AND TESTING OF CONTROL ROD DRIVES TO REDUCE COBALT INPUT
H-150	PROJECT MINDOS
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H-153	REACTOR CAVITY DECONTAMINATION AT V. C. SUMMER
H-154	USE OF RESPIRATORS AND DOSE EXPANSION
H-155	OPTIMIZING WORKER PROTECTION: A PRACTICAL APPLICATION OF RISK ANALYSIS

ID	TITLE
H-156	ADVANCED RADIATION WORKER TRAINING PROGRAM AND LABORATORY
H-157	ALARA ASPECTS OF THE CALVERT CLIFFS PRESURRIZER REPAIR PROJECT
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INNOVATIVE APPROACHES AT TMI-2

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; REMOTE SYSTEM; TMI CLEANUP

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Objectives: The aim of this project is to explore new ways to perform nuclear plant operations and maintenance tasks.

Comments: A centralized Coordination Center enables GPU Nuclear to maintain close supervision of work in the reactor building and to minimize the number of people entering. It has served to reduce exposure and to enhance the supervision of tasks and industrial safety.

Annual employee training includes an evaluation of a radiation worker's ability to dress properly and to conduct work in a radiological area. Preparation for doing actual tasks often includes practicing planned radiological work in a non-radiological area. TMI-2 radiation workers are provided with a Personnel Access Facility where support personnel assist workers in donning protective clothing and supervise entries into radiological areas. A computerized heat stress index, personal cooling devices, employee training and administrative controls limit potentially dangerous heat stress in workers.

The use of battery-powered respirators allows greater worker comfort and efficiency.

Remarks/Potential for dose limitation: Making use of remote supervision to monitor and supervise the work in the highly contaminated reactor containment building has been one successful technique to reduce occupational exposures. Extensive use of training mock-ups, setting up of a dedicated area for dressing in protective apparel, use of battery powered respirators and the heat stress control program to improve the comfort and increase the efficiency of radiation workers, have also contributed to the reduction in exposure.

References: J. E. Hildebrand, "TMI-2 Shows the Benefits of an Innovative Approach", Nuclear Engineering International, October 1988, Vol. 33, No. 411, pp. 14-15.

Duration: from: 1988 to: 1990

Funding: N/A

Status: In progress

Last Update: September 1, 1989

IDENTIFY ALL COBALT CONTRIBUTORS IN PNPS

Keywords: CONTAMINATION PREVENTION; COBALT, IMPURITIES REDUCTION; PUMP; VALVE; COBALT REDUCTION

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Objectives: EPRI and other industry studies have identified about 200 to 300 components of a typical BWR that contribute significant amounts of cobalt to the reactor systems to be irradiated to cobalt-60 as it passes through the reactor. This cobalt-60 is the source of most after-shutdown dose. The contributing components include pumps, valves, pipes, and reactor parts. Overall, they contribute about 231 effective grams of cobalt to the system each year. This task identifies each of the cobalt contributors in PNPS in order to provide the basis for cobalt reduction actions.

Comments: Includes list of all cobalt contributors with estimates of cobalt contribution and effective cobalt contribution grams from now until EOIL (End Of Initial License term).

Remarks/Potential for dose limitation: None directly from this task. Ultimately, assuming an 80% reduction of effective cobalt contribution, would be about 10,368 person-rem reduction from 21,600 to 11,232 person-rem EOIL.

References: McDonald, B., Boston Edison Company, Pilgrim Nuclear Power Station, Plymouth, MA, LTP (Long Term Plan) Item 266, Task No. 266-1-1.

Duration: from: 1988 to: 1992

Funding: N/A

Status: In progress

Last Update: January 6, 1992

BNL ALARA Center Data Base

EVALUATE HOT SPOTS ASSOCIATED WITH SPENT FUEL POOL SYSTEM

Keywords: CONTAMINATION REMOVAL; RADIATION SHIELDING; COMPONENT DECONTAMINATION; COBALT REDUCTION; SHIELDING; SOURCE REDUCTION; DOSE ESTIMATES; HOT SPOTS

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Objectives: Several local hot spots exist in the spent fuel pool systems. These are currently ameliorated with temporary shielding, but dose rates as high as 100 person-rem/hr exist in transit paths. Conclusive data are not available to ascertain the origin of source nuclides. This task characterizes the hot spots associated with this system, estimates the resultant dose, and evaluates alternative source reduction actions.

Comments: Report with recommendations and plan of source reduction action.

Remarks/Potential for dose limitation:

References: McDonald, B., Boston Edison Company, Pilgrim Nuclear Power Station, Plymouth, MA, LTP Item 266, Task No. 266-2-1.

Duration: from: 1988 to: 1989

Funding: N/A

Status: Completed

Last Update: January 6, 1992

SURROGATE LASER DISC PLANT TOUR SYSTEM

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; MAINTENANCE; INSPECTION; OPERATIONS; OPERATIONAL PRACTICE; PHOTOGRAPHIC LIBRARY

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Objectives: To provide a remote visual photograph of critical areas of the station to aide in planning design changes, maintenance and operations, as well as aide in training of personnel and the public. The computer controlled laser disc will be capable of sorting individual photographs of areas, components or systems from the entire collection. In addition, joy stick operation will allow viewing of a series of photos to provide tour capability, as well as zoom and pan capabilities.

Comments: 65,000 slides, 6 units (each includes: video disc, joy stick, Microsoft Mouse, 3 video printers, 1 IBM-AT, Math Co-processor, TARGA-BOARD, 19" color video terminal, operational manual).

Remarks/Potential for dose limitation: Estimate -- 60 Rem during outage -- 10-15 Rem during operation. Also, depending on extent of utilization.

References: None

Duration: from: 1988 to: 1989

Funding: $157K

Status: Completed

Last Update: September 1, 1989

MAINTAIN RADIOLOGICAL EVALUATION FACTORS

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; DOSE; DOSE ESTIMATE; DOSE EXPENDITURE; SOURCE STRENGTH; SOURCE REDUCTION

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Objectives: Evaluations of the projected dose effects of changes in plant facility, operation, or maintenance are complex. No model is readily available to facilitate realistic evaluations. As an alternative, radiological evaluation factors will be developed and periodically updated to normalize all source reduction evaluations. Updating will be at least yearly.

Comments: A document containing the key radiological evaluation factors for use by all NUORG people working on source reduction.

Remarks/Potential for dose limitation: None directly.

References: Vasquez, G., Boston Edison Company, Pilgrim Nuclear Power Station, Plymouth, MA, LTP Item 266, Task No. 266-9-1.

Duration: from: 1988 to: 1992

Funding: N/A

Status: In progress

Last Update: January 6, 1992

REPLACE FEEDWATER CONTROL VALVE TRIM WITH NON-COBALT DESIGN

Keywords: CONTAMINATION PREVENTION; FEEDWATER; VALVE; LOW COBALT ALLOY; COBALT-FREE ALLOY; COBALT; COBALT REDUCTION

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Objectives: Feedwater control valves (main and by-pass) have been found in EPRI studies to contribute about 15 gm/yr of cobalt. Those valves are often rebuilt during major plant outages. Non-cobalt alloy designs of internals are available for such control valves. This task would determine the availability of non-cobalt replacement internals, evaluate the assurance of satisfactory performance, and provide the necessary parts and software to make the change of internals at the next occasion for maintenance.

Comments: Redesign parts, materials and procedures to effect change.

Remarks/Potential for dose limitation: 836 person-rem from 21,600 to 20,764 person-rem EOIL (End Of Initial License term), (Annualized 44 person-rem/yr). Calculate 12,960 x 285/4420 = 836.

References: Riggs, W.,Boston Edison Company, Pilgrim Nuclear Power Station, Plymouth, MA, LTP Item 266, Task No. 266-8-1.

Duration: from: 1988 to: 1989

Funding: N/A

Status: Complete

Last Update: January 6, 1992

RECIRCULATION PUMP COBALT ELIMINATION

Keywords: CONTAMINATION PREVENTION; RECIRCULATION PUMP SEAL; LOW COBALT ALLOY; COBALT-FREE ALLOY; COBALT; COBALT REDUCTION

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Objectives: The recirculation pumps have wear surfaces hardfaced with cobalt alloys. It is expected that both recirculation pumps will be removed for inspection and overhaul sometime during RFO 8 or RFO 9. In preparation for simultaneous reduction of cobalt in these pumps, the actions to evaluate and prepare for use of non-cobalt overhaul parts will be done by this task. This task will also provide an estimate of the cobalt contribution from recirculation pumps as a basis for decisions.

Comments: Recommendation for specified non-cobalt, proven parts for pump overhaul.

Remarks/Potential for dose limitation: Less than 10 person-rem, from 20,600 to 20,590 person-rem EOIL (End Of Initial License term). Estimate to be refined based on task information about cobalt contribution from recirculation pumps.

References: Riggs, W., Boston Edison Company, Pilgrim Nuclear Power Station, Plymouth, MA, LTP Item 266, Task No. 266-8-2.

Duration: from: 1988 to: 1990

Funding: N/A

Status: Complete

Last Update: January 6, 1992

FUEL IMPROVEMENTS TO REDUCE COBALT SOURCE

Keywords: CONTAMINATION PREVENTION; FUEL; FUEL CONTAMINATION; FUEL MATERIAL; COBALT REDUCTION; COBALT-FREE ALLOY; LOW COBALT ALLOY; ELECTROPOLISHING; COBALT; PRECONDITIONING

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Objectives: Fuel contributes to cobalt sources in two ways:

1. Structural materials contain trace concentrations of cobalt.
2. Fuel surfaces can increase the deposition of crud and its in-core residence time for greater production of cobalt-60.

This task would examine potential improvements such as reduced trace concentrations of cobalt, improved surface finish on fuel tubing (electropolishing?), or pretreatment of fuel tubing.

Comments: Recommendations for fuel procurement specifications.

Remarks/Potential for dose limitation: 235 person-rem from 21,600 to 21,365 person-rem EOIL (End Of Initial License term), (Annualized 12.3 person-rem/yr). Assumes 50% reduction of cobalt contribution.

References: Clough, R., Boston Edison Company, Pilgrim Nuclear Power Station, Plymouth, MA, LTP Item 266, Task No. 266-8-3.

Duration: from: 1988 to: 1989

Funding: N/A

Status: Complete

Last Update: January 6, 1992

ESTABLISH CHEMICAL DECONTAMINATION STRATEGY

Keywords: CONTAMINATION REMOVAL; SURFACE DECONTAMINATION; CHEMICAL DECONTAMINATION; DECONTAMINATION

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Objectives: Chemical decontamination methods for piping systems are commercially available. One was effectively used in 1984 at PNPS to reduce dose on the recirc pipe replacement. Chemical decontamination source reductions are short term unless done in conjunction with other source reduction actions. As part of PNPS long term planning, a strategy for optimizing the dose reduction benefits of chemical decontamination would be developed and maintained under this task.

Comments: A report including a recommended chemical decontamination strategy and action plan.

Remarks/Potential for dose limitation: Depends upon systems and timing of decon; it could be in range of 2000 to 5000 person-rem to EOIL (End Of Initial License term).

References: Clancy, W., Boston Edison Company, Pilgrim Nuclear Power Station, Plymouth, MA, LTP Item 266, Task No. 266-6-2.

Duration: from: 1988 to: 1992

Funding: N/A

Status: In progress

Last Update: January 6, 1992

EVALUATE ZINC ADDITION TO REACTOR FEEDWATER (GEZIP)

Keywords: CONTAMINATION PREVENTION; FEEDWATER; ZINC ADDITION; ZINC INJECTION; WATER CHEMISTRY

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Objectives: GE has determined that the zinc contributed to reactor feedwater from condenser tube corrosion has the effect of reducing after shutdown radiation dose rates 30 to 50%, compared to plants that do not have zinc in the condenser tubes. PNPS does not have zinc in its condenser tubes. By injecting zinc into the feedwater, the 30 to 50% reduction should be achievable in PNPS (Pilgrim Nuclear Power Station). This task evaluates zinc addition (GEZIP) for application to PNPS.

Comments: Included are recommendations on whether to apply zinc addition in PNPS, and a plan for implementation of zinc addition.

Remarks/Potential for dose limitation: 1909 person-rem from 21,600 to 19,691 person-rem EOIL (End Of Initial License term), (Annualized 100 person-rem/yr). Assume 40% of Co-60 dose reduced for 4 years by which time 80% effective cobalt reduction has occurred.

References: Clancy, W., Boston Edison Company, Pilgrim Nuclear Power Station, Plymouth, MA, LTP Item 266, Task No. 266-6-1.

Duration: from: 1988 to: 1992

Funding: N/A

Status: In progress

Last Update: January 6, 1992

REACTOR CONTROL BLADE MANAGEMENT CONSIDERING ALARA

Keywords: CONTAMINATION PREVENTION; OPERATIONAL AND MAINTENANCE TECHNIQUES; CONTROL ROD BLADE PINS AND ROLLERS

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Objectives: PNPS has 145 control blades, including 134 of the original design. The blade assembly becomes a radiation source:

1. By contribution of corrosion and wear products from high cobalt bearing materials and stainless steel structural parts. EPRI data indicate an estimated contribution of about 20 gm/yr, or, considering the in-flux residence time of this cobalt, an effective cobalt input rate of about 100 gm/yr.
2. By handling, storage, and ultimate disposal of each blade consumed. The original blade life varied depending on the control mode and blade location, the fastest available with about twice the original life and with substantially less cobalt content. This task evaluates the contribution of control blades to PNPS radiation exposure, analyzes the costs and benefits of alternate blade management programs, and recommends the appropriate actions to accomplish ALARA blade management.

Comments: Report includes a proposed Blade Management Program for the remainder of license life, and a plan for necessary design changes, analyses, procurement and related preparations to effect recommended blade management program.

Remarks/Potential for dose limitation:

References: Aboltin, J., Boston Edison Company, Pilgrim Nuclear Power Station, Plymouth, MA, LTP Item 266, Task No. 266-5-1.

Duration: from: 1989 to: 1990

Funding: N/A

Status: In progress

Last Update: May 16, 1991

BNL ALARA Center Data Base

U.S.A. H-149

EVALUATION, POSSIBLE REDUCTION IN OPERATION AND TESTING OF CONTROL ROD DRIVES TO REDUCE COBALT INPUT

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; COMPONENT RELIABILITY; CONTROL ROD DRIVE; INSPECTION; INSPECTION AND SURVEILLANCE; OPERATION; OPERATIONAL PRACTICE; COBALT; COBALT REDUCTION

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Objectives: The CRD's (Control Rod Drives) and their hydraulic system contain cobalt alloys that contribute wear products to the reactor system each time they are operated. Excessive operation for testing, surveillance or other purpose could input unnecessary cobalt. It is expected that proven non-cobalt designs may not be available for some time. Thus, assuming PNPS will continue to operate with cobalt frequency and conditions of CRD operations, determine whether changes are feasible to reduce cobalt input.

Comments: A report with recommendation and plan of action.

Remarks/Potential for dose limitation: 157 person-rem from 21,600 to 21,443 person-rem EOIL (End Of Initial License term). (Annualized 8.3 person-rem/yr). Assume CRD operation can be reduced 50%.

References: Clancy, W., Boston Edison Company, Pilgrim Nuclear Power Station, Plymouth, MA, LTP Item 266, Task No. 266-4-1.

Duration: from: 1988 to: 1989

Funding: N/A

Status: Complete

Last Update: January 6, 1992

PROJECT MINDOS

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; CONTAMINATION PREVENTION; CHEMISTRY; MATERIAL; FUEL; DOSE REDUCTION

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Objectives: To reduce radiation exposures at Oskarshamn NPP; to do so through a number of sub-projects, which are as follows:

- Radiation Protection
- Chemistry/Materials/Fuel
- Operation/Refuelling outage
- Information/Attitudes/Motivation.

Comments: Although the collective doses at the Oskarshamn NPP are still reasonably low (5.5 person-Sv for 1988), they have been increasing during the last three years. Instead of the usual approaches to radiation protection a number of other techniques, which are well known but difficult to utilize, were tried. One approach was to involve personnel in all the various groups within the organization and have them participate in a project which is concerned with the most efficacious factors that affect collective dose. In addition to generating useful ideas and new solutions, the project has increased mutual understanding and cooperation for the ALARA program.

Remarks/Potential for dose limitation: The ultimate goal of this project is to maintain the collective dose at Oskarshamn NPP below 3.6 person-Sv/year.

References: None

Duration: from: 1989 to: 1990

Funding: N/A

Status: In progress

Last Update: September 15, 1990

BNL ALARA Center Data Base

STUDY ON THE ALARA POLICY IN KOREA

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; DOCUMENT ANALYSIS; ALARA; OPTIMIZATION; COST-BENEFIT ANALYSIS; DOSE REDUCTION; OCCUPATIONAL RADIATION EXPOSURE

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Objectives:

1. To collect documents and operating experiences in relation to radiation protection and the implementation of the ALARA principle to both domestic and foreign nuclear power plants.
2. To construct a data base for future study on the optimization of radiation protection in Korean nuclear power plants.

Comments: More than 10 years have passed since commercial nuclear power plants started their operation in Korea. However, there has never been a systematic study on the status of radiation protection and the implementation of the ALARA principle to Korean nuclear power plants. The results of this study can be used as a data base for future study on the optimization of radiation protection by both the regulatory body and utilities in Korea.

Remarks/Potential for dose limitation:

References: None

Duration: from: 1990 to: 1993

Funding: N/A

Status: In progress

Last Update: August 29, 1990

REDUCTION OF TIME, EXPOSURE, AND COST THROUGH PLANT DECONTAMINATION

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; AREA DECONTAMINATION; PROTECTIVE APPAREL; PROTECTIVE CLOTHING; RESPIRATOR; COST-BENEFIT ANALYSIS

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Objectives: The objectives of this project are:

1. To reduce plant contaminated area from 27,000 sq.ft. to 8,000 sq.ft.
2. To maintain this level of contamination through subsequent outages.
3. To investigate the reasons why areas get contaminated.
4. To carry out a cost benefit study to analyze the level of dollar savings from reduced contamination in the plant.
5. To show the financial and human factor benefits from reducing contaminated square footage.
6. To provide justification for staffing of personnel to obtain the reduced contaminated area.

Comments: The costs and savings during this project were carefully monitored and tabulated as:

1. area decontaminated.
2. cost of entries during decon.
3. cost of entries without decon.
4. cost of the decon operation

The cost of maintaining the areas as contamination free and the monthly savings from entries were also maintained.

It was found that considerable savings accrued, principally because the high cost of protective clothing was avoided.

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Remarks/Potential for dose limitation: It was found that the principal reasons areas become contaminated is due to poor work practices. Vent and drain processes by the operations group was one major source. This was followed by poor valve work and then system testing.

Dose savings can be achieved by allowing planners to access areas and reduce time in area since protective clothing is not used. Also, reduction of dose received from highly contaminated areas.

References: Vander Velde G., "Reduction of Time Exposure and Cost Through Plant Decontamination", Proceedings, Westinghouse REM Seminar, Pittsburgh, Pennsylvania, October 1990.

Duration: from: 1988 to: 1991

Funding: $105,000

Status: Ongoing

Last Update: December 7, 1990

REACTOR CAVITY DECONTAMINATION AT V. C. SUMMER

Keywords: CONTAMINATION REMOVAL; CONTAMINATION PREVENTION; REACTOR CAVITY; DECONTAMINATION; STRIPPABLE COATING; ISOLOCK 300

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Objectives: The 1990 refueling outage schedule indicated the necessity of filling and draining the reactor cavity twice. The possibility of radioacuve airborne contamination and the migration of discrete radioactive particles was increased by the outage schedule requirements. The decision to use a strippable coating was made in order to reduce these radiological hazards.

A non-immersible coating would have been applied and removed twice. This would have resulted in a potential radioactive airborne hazard during refueling activities. Additional man-hours would be required to remove the coatings a second time and there would be an appreciable increase in the volume of radwaste.

For these reasons the Health Physics Staff recommended the use of the immersible coating ISOLOCK 300.

Comments: The application and removal process required 148 hours. Radiological surveys performed after coating removal indicated contamination levels of 10,000 dpm/100 cm^2 and 50,000 dpm/100 cm^2 on the walls and floors. The cavity decontamination was completed with 6.5 person-rem of exposure, a reduction of 38% compared to the previous outage. The reactor head work was completed with 33 person-rem, a 20% reduction. This was attributed to an increase in worker efficiency due to a decline in the need for protective clothing and respiratory protection. The total personnel contaminations decreased from 264 to 120 compared to the previous outage, and the calculated dose to the skin as a result of these contaminations was down from 31 to 12 rem.

Remarks/Potential for dose limitation: Post-outage critique identified several recommendations:

1. The size of the work crew should be sufficient to keep the time required to mask the cavity and remove the coating to a minimum.
2. The coating cure time is a function of cavity wall temperature and the relative humidity. An allowance of about 24 hours should be used.
3. Access to polar crane directly affects the evolution of the masking, painting and stripping. A new Cav-Span crane system will accelerate masking and stripping.

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U.S.A. H-153

References: Laney, C.F., "Reactor Cavity Decontamination at V.C. Summer", Proceedings, Westinghouse REM Seminar, Pittsburgh, Pennsylvania, October 1990.

Duration: from: 1990 to: 1990

Funding: N/A

Status: Completed

Last Update: December 7, 1990

USE OF RESPIRATORS AND DOSE EXPANSION

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; RESPIRATOR; EXTERNAL EXPOSURE; INTERNAL EXPOSURE; WORKER EFFICIENCY

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Objectives:

1. To minimize the cost of the respirator program.
2. To investigate the reasons for overutilization of respirators.
3. To evaluate the cost per person-rem saved.
4. To determine the additional dose to the respirator wearer from the decrease in work efficiency and the resultant increase in exposure times.

Comments: During the studied outage a total of 6,869 respirators were issued at a cost of $105,000. Air sampling data were compared to respirator use logs to determine the dose reduction efficiency from respirator utilization, with the following results:

- 188 were worn in areas = or >1 MPC.
- 350 were worn in areas = or > 0.25 but less than 1 MPC.
- 747 were worn in areas = or > 0.10 but less than 0.25 MPC.
- 2,486 were worn in areas = or > 0.01 MPC but less than 0.1 MPC
- 444 were worn in areas with positive air sample results but less than 0.01 MPC.
- 77 were worn in clean areas with no positive air sample results and no history of airborne contamination.

It was also noted that work performance was slowed by between 10 to 200% from use of various breathing air devices.

Remarks/Potential for dose limitation: It was concluded that the use of a filter respirator cannot be justified unless the external dose rate, including the contribution from noble gases, is numerically less than ten times the total MPC fraction (excluding the noble gases).

References: Greenwood, R.A. and J. O'Dou, "Use of Respirators and Dose Expansion", Proceedings, Westinghouse REM Seminar, Pittsburgh, Pennsylvania, October 1990.

Duration: from: 1990 to: 1990

Funding: N/A

Status: Completed

Last Update: December 7, 1990

OPTIMIZING WORKER PROTECTION: A PRACTICAL APPLICATION OF RISK ANALYSIS

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; RADIATION PROTECTION; OPTIMIZATION; COST-BENEFIT ANALYSIS; INDUSTRIAL SAFETY; OCCUPATIONAL RISK; RISK ANALYSIS

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Objectives: To optimize the occupational risk to radiation workers, taking into account all types of risks. The potential risks may be from internal or external radiation hazards, from non-radiological industrial hazards, and from stochastic, as well as non-stochastic, processes.

Comments: In the approach suggested, the following steps should be followed in sequence:

1. Identify and evaluate all sources of risk. These include ionizing and non-ionizing radiation, hazardous materials, industrial safety aspects, work processes and other factors which are specific to workers, such as age, sex, physical condition, hereditary factors, allergies, etc.
2. Analyze protection options.
3. Optimize worker protection.
4. Minimize the total risk.

Remarks/Potential for dose limitation: A number of examples are presented. Each example allows a number of options. The technique allows one to select the best possible option from those available.

References: Williams, Michael C., "Optimizing Radiation Worker Protection: The Practical Application of Risk Analysis," Health Physics, December 1990, Vol. 59, No. 6, pp. 925-929.

Duration: from: 1990 to: 1991

Funding: N/A

Status: Initiated

Last Update: January 8, 1991

ADVANCED RADIATION WORKER TRAINING PROGRAM AND LABORATORY

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; TRAINING; EXTERNAL EXPOSURE; INTERNAL EXPOSURE; WORKER PROTECTION; RADIATION PROTECTION

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Objectives: The Advanced Radiation Worker concept was evolved at Davis-Besse station to improve everyday radiological work practices. The program involved training workers in various aspects of radiological safety. The areas included in this program were:

1. Acquaint workers with the need for basic administrative controls, such as radiological work permits.
2. Acquaint workers with the risks of work in a radiation environment.
3. Provide training on procedures, such as: how to enter and leave contaminated areas, to dress and undress in protective apparel.

The program entailed 8 hours of classroom instruction and practical demonstrations.

Comments: The goals of the program were to:

1. decrease the rate of personnel contaminations.
2. lower the station collective dose.
3. decrease the radwaste generated.
4. improve communication between radiation workers and radiological control personnel.

Remarks/Potential for dose limitation: Although it required considerable explanation of the programs objectives to obtain worker support for the program, the results were positive:

1. There was a significant decrease in the number and causes of contamination.
2. Communications between the work groups and the radiological control group have improved.
3. There is a better understanding of the reasons for the procedures instituted by the radiological control group.
4. There is a better appreciation and understanding of the previously unidentified problems of the work groups by the radiological control personnel.

References: Haney, W.G. and L.D. Kleit, "Advanced Radiation Worker Training Program", Proceedings, Westinghouse REM Seminar, Pittsburgh, Pennsylvania, October 1990.

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

Funding: N/A

Status: In progress

Last Update: December 10, 1990

ALARA ASPECTS OF THE CALVERT CLIFFS PRESSURIZER REPAIR PROJECT

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; CONTAMINATION REMOVAL; RADIATION PROTECTION; SHIELDING; DECONTAMINATION; HYDROLAZING; PRESSURIZER

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Objectives: To reduce the high dose rates in the pressurizer area and the radiation dose to workers during the pressurizer heater repair project.

The initial doses on the pressurizer bottom were up to 300 mR/h Gamma. The contact dose on the level tap line was between 2,500 and 3,000 mR/h Gamma.

Comments: The following measures were taken:

1. The hot spots on instrument lines were shielded, reducing dose rates from 3,000 mR/h to 700 mR/h. The ventilation duct beneath the grating was also shielded, reducing dose rates from 100 to 20 mR/h.
2. A herculite tent was used to enclose the contaminated area. Two 1,00 cfm Hepa filters were installed and drew suction from the work area. This reduced the airborne contamination and impr ed cooling.
3. The pressurizer sample line was removed to reduce dose rates from this source and improve access.
4. Prejob briefings were conducted at the start of every shift.
5. A video camcorder was utilized to tape the job in progress. It was later reviewed by the work group and the ALARA personnel. It proved an invaluable tool because workers in the next shift could see where mistakes had occurred and rectify them.
6. Extensive mock-up training was performed for all work phases.

Remarks/Potential for dose limitation: When the work program was expanded to remove and replace all heater sleeves, it was decided to make additional efforts to reduce dose rates. These included:

1. Decontamination of pressurizer internals by means of hydrolazing in order to reduce dose rates at the bottom of the pressurizer and exposure rates from the shine coming from the sleeve openings.
2. Putting shield plugs in the open sleeves and holes.
3. Shield the pressurizer surge line to reduce general area dose rates from this source.

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U.S.A. H-157

As a result of these efforts, the dose rate at the bottom of the pressurizer was reduced from 300 to 100 person-rem/h. General area readings were also reduced to 60 person-rem/h.

References: Hutson, S.G., "Calvert Cliffs Pressurizer Repair Project", Proceedings, Westinghouse REM Seminar, Pittsburgh, Pennsylvania, October 1990.

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

Funding: N/A

Status: Completed

Last Update: December 10, 1990

ACE - ALARA CENTER'S DOSE-REDUCTION INFORMATION SYSTEM

Keywords: COMPONENT RELIABILITY; OPERATIONAL AND MAINTENANCE TECHNIQUES; REMOTE SYSTEM; CONTAMINATION REMOVAL; RADIATION SHIELDING; CONTAMINATION PREVENTION; RADIATION PROTECTION; SOURCE REDUCTION; ALARA

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Objectives: The ALARA Center maintains a number of databases on dose reduction and ALARA for the Nuclear Regulatory Commission. These databases cover ongoing research, innovative health physics techniques, processes and practices related to dose reduction, a bibliography containing 1,613 abstracts of articles on ALARA related topics, and short notes on current topics of interest in ALARA. This material is published periodically and made available to the nuclear industry. Up-to-date information from these databases is available through telephone link, using a personal computer and modem, and through FAX machines. The ACE on-line system is designed to be very user friendly. ACE stands for ALARA Center Exchange.

Comments: The ACE system allows the user to:

1. Search each database through either keywords or simple English.
2. Sort data alphabetically, numerically or by date.
3. Print information to the remote printer.
4. Electronically capture screens for off-line playback and print out.
5. Record the whole session for later off-line playback.
6. View and retrieve through FAX machines: charts, graphs, photographs of equipment, journal articles, and other documents.

Remarks/Potential for dose limitation: A package is required in order to access ACE. The package contains the system manual and a diskette containing the communication software. The software installs itself automatically and is completely ready for connecting to ACE.

The access packages may be obtained by calling Maria Beckman at (516)282-3228.

U.S.A. H-158

References: Khan, T.A., "ACE - ALARA Center's Dose-reduction Information System", Proceedings, Westinghouse REM Seminar, Pittsburgh, Pennsylvania, October 1990, U.S.A.

Duration: from: 1989 to: 1992

Funding: N/A

Status: In progress

Last Update: December 10, 1991

AN EFFECTIVE ALARA AWARENESS PROGRAM

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; ALARA AWARENESS; RADIATION PROTECTION; ALARA; OPTIMIZATION

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Objectives: Following the January 1, 1989 issue of the Department of Ene.gy revised order 5480.11, the Westinghouse Hanford Company's ALARA Program Office began an aggressive ALARA Awareness campaign. The goal was to ensure the awareness of all facets of the ALARA program including radiological and non-radiological exposure minimization for humans and the environment.

Comments: The objectives of the program were achieved through consistent application of visibility, recognition and documentation. For visibility, the following actions were taken:

1. An ALARA logo contest was conducted. The selected logo reflects the expanded concept of radiological and non-radiological exposure minimization.
2. Dedicated ALARA display modules were installed in major facilities.
3. A quarterly ALARA Newsletter was started. Its objective was to highlight current issues and concerns, and publicize innovative ideas and good practices.
4. The ALARA Awareness Administrator provides information and assistance to other Health and Safety programs.
5. An ALARA awareness week was conducted. Pamphlets and videos were distributed and contests on ALARA were conducted for Hanford workers.

The documentation of the ALARA Awareness Program was initiated by a video and pamphlets.

Remarks/Potential for dose limitation: Special criteria were provided to the cost saving committee, clearly defining when it is appropriate to claim dollars saved per person-rem. The company will only approve dollars saved if it fits within the developed criteria.

Performance indicators, charts, and graphs are prepared on a monthly basis for senior management to review. These performance indicators provide the proper direction to management about current issues and concerns.

References: Aldridge, T.L. and D.A. Hammond, "An ALARA Awareness Program", Westinghouse REM Seminar, Pittsburgh, Pennsylvania, October 1990, U.S.A.

Duration: from: 1989 to: 1991

Funding: N/A

Status: In progress

Last Update: December 11, 1990

AN ALARA TRAINING PROGRAM FOR DESIGN ENGINEERS

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; RADIATION PROTECTION; ALARA; OPTIMIZATION; TRAINING

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Objectives: The Westinghouse Hanford Company conducts an ALARA training program for design engineers. The program was conceived to comply with the requirements of DOE order 5480.11 which states that "ALARA shall be documented and tracked."

The four elements of the program are:

1. Develop a program goal.
2. Ensure program documentation and tracking.
3. Establish a strong program structure.
4. Develop and apply program tools.

Comments: Some components of the program are a set of worksheets and a checklist. Each worksheet is a fully developed technical training package. It is described in reference (1).

Section 1 of the checklist has the following criteria:

1. Minimal contamination and hazardous waste production.
2. Surface coatings for ease of decontamination.
3. Remote handling equipment provided.
4. Traffic patterns minimize passage through contaminated and high exposure areas.

Section 2 pertains to system design as follows:

1. Design and locate to minimize maintenance time.
2. Locate in lowest radiation fields.
3. Design for removal to low dose rate areas.
4. Locate away from hazardous substances and conditions.
5. Filter ventilation system to remove contamination.
6. Balance air to reduce spread of contamination.
7. Design contamination control and exposure reduction.

Remarks/Potential for dose limitation: The following enhancements to the training program have been recommended:

1. Additional internal and external references to the check list statements.
2. Development of nonradiological checklists.

3. Enhancement of nonradiological training.
4. Provision of guidance in internal manuals for the documents requiring checklists.
5. Exchange of information both nationally and internationally.
6. Incorporation of other sources of ALARA good practices.

References: 1) Aldridge, T.L., "ALARA Program Manual", WHC-CM-4-11, Rev. 2, Washington Hanford Company, Richland, Washington, 1990. 2) Aldridge, T.L. and D.O. Hess, "ALARA Training Program for Design Engineers", Westinghouse REM Seminar, Pittsburgh, Pennsylvania, October 1990, U.S.A.

Duration: **from:** 1989 **to:** 1991

Funding: N/A

Status: In progress

Last Update: December 11, 1990

BNL ALARA Center Data Base

SYSTEM DECONTAMINATION OF RWCU SYSTEM

Keywords: CONTAMINATION REMOVAL; DECONTAMINATION; BWR; DOSE REDUCTION; REACTOR WATER CLEANUP SYSTEM

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Objectives: To decontaminate part of the RWCU system in order to reduce the dose rates and to keep the personnel doses as low as reasonably achievable.

Comments: The decontamination method used was the CORD method (Chemical Oxidation-Reduction Decontamination) developed by Siemens KWU, Germany. The total time for the decontamination was 80 hours.

Remarks/Potential for dose limitation: An average decontamination factor was calculated to be 4.3 and the dose rates were reduced, in average, by 77%. The collective dose for the decontamination was 50 mpersonSv (5 person-rem).

References: Solstrand, C.,"System Decontamination of RWCU System at Oskarshamn-1", Proceedings 12th Westinghouse REM Seminar, Pittsburgh, 1990, Westinghouse Electric Corporation, PO Box 355, Pittsburgh, PA, 15230, U.S.A.

Duration: from: 1990 to: 1990

Funding: N/A

Status: Completed

Last Update: December 12, 1990

RESISTANCE TEMPERATURE DETECTOR BYPASS SYSTEM ELIMINATION

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; RTD BYPASS SYSTEM; SOURCE REDUCTION; EXPOSURE REDUCTION; RADIATION PROTECTION; MAINTENANCE

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Objectives: To replace the existing RTD bypass direct immersion system with the latest, well mounted RTDs. This would not only reduce the reactor down time but also avoid considerable maintenance in high-dose areas.

Comments: In addition to a reduction in the source term, it was considered that the dose rates inside the loop cavities would decrease from this project. The reduction in general area dose rates would reduce doses resulting from reactor coolant pump, steam generator, and other maintenance activities.

A total of nine Radiation Work Permits were written for the project. The work included the following:

1. Scaffolding support for RTD (5 Rem).
2. Interference and piping removal (33 Rem).
3. Shielding support (4 Rem).
4. Scoop installation (3 Rem).
5. New cable/conduit installation (18 Rem).
6. Technical support (6 Rem).
7. Decon support (3 Rem).
8. Radiography for X-over lines (0.8 Rem).
9. Cable and conduit in auxiliary (1 Rem).

Remarks/Potential for dose limitation: Based on previous experience from other power plants, it was estimated that a collective dose of 124 person-rem would be required to carry out this project. The ALARA committee set itself a goal of 112 person-rem, which was 10% below the estimated collective dose. The actual total accumulated exposure for the nine radiation work permits was 103 person-rem.

BNL ALARA Center Data Base

U.S.A. H-162

References: Mothena, P.A., "RTD Bypass System Elimination at V.C. Summer", Proceedings, Westinghouse REM Seminar, Pittsburgh, Pennsylvania, October 1990, Westinghouse Electric Corporation, P.O. Box 355, Pittsburgh, PA 15230, U.S.A.

Duration: from: 1990 to: 1990

Funding: N/A

Status: Completed

Last Update: January 8, 1990

400 R/HR HOT SPOT REMOVAL AT COOPER NUCLEAR STATION

Keywords: CONTAMINATION REMOVAL; HOT SPOT; BWR; RADIATION PROTECTION; COOPER NUCLEAR STATION

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Objectives: The objective was to replace the bottom vessel drain line which had trapped several hot spots and had been a source of elevated dose to maintenance personnel. The drain line had a hot spot giving contact dose rate of 400 R/h.

Comments: After careful planning and mock-up training, the actual task was undertaken. Insulation was removed and shielding was installed. By careful shielding, the contact dose rates were reduced to 5 R/h and at 18" to 400 person-rem/h. Task workers were outfitted with multiple badges, rainsuits and respirators, for the hot spot removal. They were staged in the low dose rate area outside the drywell equipment hatch. Health physics personnel removed the insulation at the predetermined cut location. Interference lines were then cut away. The pipe was next clamped and cutting began on the hot tee and elbow. When the cuts had been made the cutting rig was moved back 4 feet and the hot piece, with the shielding still attached, was placed in a stainless steel basket. The basket was transported by a specially fitted forklift truck and a crane to the spent fuel cask pad area in the fuel pool.

Remarks/Potential for dose limitation: The total dose received during the hot spot removal was 650 person-rem. The dose for the entire project, including shielding and removal, was 1.6 Rem. Engineering cooperation was essential throughout the project to ensure that in addition to removing the hot spot, potential crud traps were also removed. The project demonstrates the importance of careful pre-job planning and the early involvement of ALARA personnel in the design change process.

References: Carson, T. and J. Whisler, "400 R/hr Hot Spot Removal on the Two-inch Bottom Vessel Drain Line", Proceedings, 12th Westinghouse REM Seminar, Pittsburgh, Pennsylvania, October 1990, Westinghouse Electric Corporation, PO Box 355, Pittsburgh, Pennsylvania, 15230, U.S.A.

Duration: from: 1990 to: 1990

Funding: N/A

Status: Completed

Last Update: January 8, 1991

INNOVATIVE SHIELDING

Keywords: RADIATION SHIELDING; SHIELDING; DOSE REDUCTION; SOURCE REDUCTION; RADIATION PROTECTION

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Objectives: To investigate the most appropriate type of shielding for an atypical refueling canal configuration; to investigate the comparative costs and man-rem benefits of various shielding materials; to investigate the possible applications of various types of water shields.

Comments: The plenum shield design involved a number of considerations. The following factors were examined:

1. Type of material.
2. Weight of the shielding.
3. Maneuverability.
4. Availability.
5. Set-up, tear-down requirements.
6. Costs.

The cost of shielding assembly was determined as follows:

• 5 water shields and associated equipment		$50,000 (hoses, fittings, etc.)
• 2 I-beams and rigging		$10,000
• 240 blankets of lead (leased)		$ 3,000
	TOTAL	$63,000

Remarks/Potential for dose limitation: The following advantages were seen in utilizing water shields:

- Water shields have very little weight compared to lead or concrete.
- Set up time is very low (between 10 and 15 minutes).
- The shields are designed to fit tightly in a given space with very little streaming possible.
- The watershields were very easy to contaminate and posed no mixed waste problem.

A number of other applications for these shields are planned.

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U.S.A. H-164

References: Priest, J.M. "Innovative Shielding" Proceedings, 12th Westinghouse REM Seminar, Pittsburgh, Pennsylvania, October 1990, Westinghouse Electric Corporation, PO Box 355, Pittsburgh, Pennsylvania, 15230, U.S.A.

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

Funding: N/A

Status: Completed

Last Update: December 14, 1990

REMOVAL OF CONTROL ROD DRIVE THROUGH ROBOTICS

Keywords: REMOTE SYSTEM; ROBOTICS; CONTROL ROD DRIVE; BWR; RADIATION PROTECTION

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Objectives: To evaluate the removal of control rod drives by means of Toshiba's remote control-rod-drive handling system.

The key components of the system are:

- Remote handling machine - RHM
- Disassembly and cleaning machine - DCM

Comments: Only one maintenance person was required during removal and installation. The RHM contained the water that normally spills on the floor during manual CRD removal. Thus, considerable contamination was avoided. The upper filter of the CRD was automatically shielded during transfer, reducing dose from this source. Dose was reduced by 50% per dive from the industry average for manual CRD removal.

The DCM flushes the upper filter, considerably reducing the dose rate from the filter, and avoiding the need for respirators.

General area dose rates were reduced to 2-10 person-rem/h during cleaning and disassembly. The complete disassembly was done underwater.

Remarks/Potential for dose limitation: Personnel are undergoing training in the utilization of the robotic system. After the training is complete, an additional 30% reduction in dose to personnel is expected. At present, the average dose to change a control rod drive at Hope Creek is 340 person-rem.

References: Wallender, T., "Control Rod Drive Removal Through Robotics" Proceedings, 12th Westinghouse REM Seminar, Pittsburgh, Pennsylvania, October 1990, Westinghouse Electric Corporation, PO Box 355, Pittsburgh, Pennsylvania, 15230, U.S.A.

Duration: from: 1990 to: 1990

Funding: N/A

Status: Completed

Last Update: December 14, 1990

DATA ACQUISITION ON PWR CONTAMINATION

Keywords: CONTAMINATION PREVENTION; CONTAMINATION REMOVAL; DATA ACQUISITION; PWR; CONTAMINATION; STEAM GENERATOR; PIPE; CORROSION PRODUCT; COBALT

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Project Manager:

Phone:

Objectives:

1. To quantify the sources of radiation.
2. To quantify the resultant dose rates.
3. To identify the spatial distribution of contamination.
4. To identify the spectral distribution of the contamination sources.
5. To identity the origin of exposure to personnel.

Comments: The program involves a series of methods, computer codes, and techniques. They include:

1. The PACTOLE code, which is used to simulate corrosion product contamination.
2. The PROFIP code, which is used to simulate fission product contamination.
3. A systematic program of periodic dose rate measurements at specified points is employed, for example, in the steam generator channel heads and piping.
4. Spectral measurement using a portable gamma ray spectroscopy system (EMECC) is also used.

By means of these techniques, a fairly complete assessment of the extent and type of activity on, for example, the bowl, partition plate and various parts of the associated piping, is possible.

Remarks/Potential for dose limitation: The technique will be utilized to investigate hot spots, also.

References: Anthoni, S.M., "French Experience on PWR Contamination", Westinghouse REM Seminar, Pittsburgh, Pennsylvania, October 1990, Westinghouse Electric Corporation, PO Box 355, Pittsburgh, PA 15230, U.S.A.

Duration: from: 1990 to: 1991

Funding: N/A

Status: In progress

Last Update: December 27, 1990

PANTHERE RP: A TOOL FOR EVALUATING DOSE RATES

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; CONTAMINATION PREVENTION; RADIATION SHIELDING; DOSE RATE; RADIATION PROTECTION; CAD

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Objectives: A technique from Computer Aided Design (CAD) has been utilized to predict dose rates at nuclear power plants. The objectives are to:

1. Predict key parameters such as sources and dose rates.
2. Measure actual values of predicted parameters.
3. Analyze the differences between predicted and measured values to improve the techniques of prediction and of measurement.

Comments: The CAD technique, called PANTHERE RP, for "Prediction and Theoretical Analysis of Exposures at Reactors for Radiation Protection", has been developed to:

- Perform dose rate mapping for complex layouts.
- Give information about the contribution to radiation fields from various parts of the installation for any specific point.
- Provide information about the influence of source composition, e.g. isotopes involved, evolution with time, etc.

PANTHERE RP starts from a knowledge of the radiation sources and a 3 dimensional representation of the geometry. It allows the calculation of dose rates at any given point.

Remarks/Potential for dose limitation: PANTHERE RP allows the calculation of dose rates and their representation as tables or in graphical form. It is hoped to utilize the system in the future at maintenance work sites to predict collective dose.

References: Ridoux, P., and A. Brissaud, "PANTHERE RP: A Tool for Prediction of Dose Rates" Proceedings, 12th Westinghouse REM Seminar, Pittsburgh, Pennsylvania, October 1990, Westinghouse Electric Corporation, PO Box 355, Pittsburgh, Pennsylvania, 15230, U.S.A.

Duration: from: 1990 to: 1991

Funding: N/A

Status: In progress

Last Update: December 27, 1990

THE INGREDIENTS OF A UTILITY'S DOSE REDUCTION PROGRAM

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; DOSE REDUCTION; RADIATION PROTECTION; ALARA; PROCEDURE; ZINC INJECTION

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Project Manager:

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Objectives: To identify and implement dose reduction actions and programs. This program should be based on the following components:

1. Establishment of a corporate ALARA committee to include a vice president, managers of the nuclear departments, and the corporate health physicist.
2. Communication of the expectations of senior management and corporate ALARA committee on collective dose for various nuclear plants to plant personnel and management.
3. Setting up of company dose reduction goals based on senior management's expectations.
4. Establishment of a dose-reduction action plan.

Comments: One primary action item was the source-term reduction program. It included the following actions at BWRs:

- Chemical decontamination of reactor piping.
- Zinc injection.
- Removal of stellite from control rod blades.

The following at PWRs:

- Use of elevated pH chemistry.
- Cobalt identification and reduction.
- Control and coordination of Rx coolant filtration.

Remarks/Potential for dose limitation: During the course of the dose reduction program the following actions were also taken:

- Improvement in the ALARA engineering process.
- Improvement in outage planning.
- Increased use of skilled craftsmen for high-dose work.

- Incorporation of ALARA incentives into nuclear contracts.
- Restriction of access to radiation control areas.
- Increased utilization of robotics.
- Increased emphasis on advanced radiation worker training.

References: Mayer, B., "Carolina Power & Light Company Dose Reduction Program" Proceedings, 12th Westinghouse REM Seminar, Pittsburgh, Pennsylvania, October 1990, Westinghouse Electric Corporation, PO Box 355, Pittsburgh, Pennsylvania, 15230, U.S.A.

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

Funding: N/A

Status: In progress

Last Update: December 28, 1990

METHODS USED TO ACHIEVE OUTAGE GOALS AT DIABLO CANYON

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; DOSE REDUCTION; RADIATION PROTECTION; ALARA; PROCEDURE

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Objectives: Some of the main objectives of this program were:

- To reduce duration of outages
- To reduce collective dose during outages and bring it in line with INPO's recommendations

Comments: As a result of this program, the duration of outages was gradually reduced from a high of 122 days during the first outage of unit 1, to 57 days for the third outage of unit 2. The collective exposures during outages were also reduced from a high of 436 person-rem for the second outage of unit 1, to 266 person-rem for the third outage of unit 2.

Remarks/Potential for dose limitation: Some of the most important lessons learned were:

- Maintain an outage control center and give it sufficient authority.
- Develop and make use of high-impact teams.
- Start with scope definition.
- Carry out task analysis, before, during, and after tasks.
- Make careful preparations.
- Critique the implementation of the tasks.
- The significance of a good briefing by foremen prior to commencement of each task cannot be over stated.

References: Irving, T., "Methods Used to Achieve Outage Goals at DCPP", Proceedings, 12th Westinghouse REM Seminar, Pittsburgh, Pennsylvania, October 1990, Westinghouse Electric Corporation, PO Box 355, Pittsburgh, Pennsylvania, 15230, U.S.A.

Duration: from: 1990 to: 1991

Funding: N/A

Status: In progress

Last Update: December 28, 1990

RADIATION EXPOSURE REDUCTION PROGRAM AT MITSUBISHI HEAVY INDUSTRIES

Keywords: CONTAMINATION PREVENTION; DOSE REDUCTION; RADIATION PROTECTION; CHEMISTRY; FILTER; PRECONDITIONING; PASSIVATION; ROBOTICS; AUTOMATION; PWR

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Objectives: To develop programs to reduce occupational radiation exposure during design, maintenance and inspection of Mitsubishi PWR plants. The programs are to be based on:

- Radiation source improvement measures
- Equipment improvement measures
- Radiation Exposure management

Comments: The radiation source reduction measures include:

- Improvement of materials by development of alternates to stellite.
- Better preconditioning by development of corrosion resistant film and use of HFT water chemistry control.
- Improvement of coolant chemistry, using high pH during operations and removal of ionic NI/Co from outer oxide layer.
- Improved techniques for CRUD removal, by improved filtration techniques and removing CRUD producing material during construction.
- On site investigation of CRUD behavior, through radiation monitoring, crud sampling, and other techniques.

Radiation exposure management measures are based on:

- Reduction of working time in high radiation areas.
- Reduction of radiation levels by employing temporary shields.
- Optimum radiation exposure management during operations by reduction of working time, and development of better tools, methods and management techniques.

Remarks/Potential for dose limitation: The following improvements in equipment have been realized:

- Integrated R/V Upper Head (CRDM cable bridge and ring duct elimination).
- Improvement of cavity seal ring clamp.

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The following automatic equipment has been developed:

1. Handling device for S/G manhole cover.
2. Stud hole brushing device for S/G manholes.
3. Telescopic type S/G sludge lancing device.
4. RCP seal handling device.
5. Fastening/Loosening device for valve flange bolts.
6. UT device for piping.
7. R/V tensioning system (QC tensioner plus automatic tensioner positioning system).
8. Self-mobile R/V stud hole brushing device.
9. Inspection device for R/V flange seat face.
10. UT device for R/V stud bolts.
11. Cleaning device for R/V stud bolts.

References: Hemmi, Y., "Radiation Exposure Reduction program at MHI", Proceedings, 12th Westinghouse REM Seminar, Pittsburgh, Pennsylvania, October 1990, Westinghouse Electric Corporation, PO Box 355, Pittsburgh, Pennsylvania, 15230, U.S.A.

Duration: from: 1990 to: 1991

Funding: N/A

Status: In progress

Last Update: December 31, 1990

CLAMSHELL NOZZLE/PIPE SHIELDING

Keywords: RADIATION SHIELDING; SHIELDING; NOZZLE SHIELDING; INSPECTION

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Objectives: To provide a means of quickly and effectively shielding high dose RPV nozzles for inspection activities.

Comments: Using lead blankets to shield RPV nozzles was time consuming and increased the dose to the shielding personnel. Preformed "clam shell" shielding rings were developed to expedite the process. Four-inch-wide rings are hinged and latched around the nozzle piping. The rings interlock to minimize streaming effects. These rings are stainless steel shells filled with lead. The shielding effect of these rings is identical to lead blankets, but weigh less.

Remarks/Potential for dose limitation: During a recent outage, the shielding rings were used. It took an average of 10-15 minutes to install the rings per nozzle. Previously, it would take 30-45 minutes. The nozzle dose rates are typically between 5 and 20 R/hr.

References: None

Duration: from: 1990 to: 1991

Funding: N/A

Status: Completed and In Use

Last Update: May 31, 1991

FEEDWATER NOZZLE THERMAL SLEEVE HYDROLAZING

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; HYDROLAZING; FEEDWATER NOZZLE; INSPECTION; DOSE REDUCTION

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Objectives: To reduce radiation doses to nozzle inspection technicians in the drywell by flushing high-dose "crud" from the thermal sleeve region of the feedwater nozzles; to maintain reactor vessel water clarity to minimize impact on refuel floor activities.

Comments: The double thermal sleeve/piston seal configuration of a BWR 4's feedwater nozzle creates a trap for contaminated particles (crud) to settle out. NUREG 0619 requires that nozzles be ultrasonically examined every other fuel cycle. A hydrolazing fixture with several lances is positioned in the feedwater nozzle by technicians on the refueling bridge. The displaced crud is captured by a vacuum shroud (powered by a submersible pump) and directed to the fuel pool cleanup system.

Remarks/Potential for dose limitation: Comparing pre- and post-hydrolazing dose rates of the feedwater nozzles in the drywell show a reduction of contact dose rates by a minimum of 50%. Most recent efforts during a refuel outage showed a drop from 22 R/hr to 2 R/hr on some nozzles.

References: None

Duration: from: 1987 to: 1991

Funding: N/A

Status: In progress

Last Update: May 31, 1991

SNUBBER POSITIONING FIXTURE

Keywords: OPERATIONAL AND MAINTENANCE TECHNIQUES; SNUBBER; RESTORATION; DOSE REDUCTION

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Objectives: To provide safety and dose reduction during removal and installation of large sized mechanical snubbers.

Comments: Past practice was to have three men manually lift and turn the large sized mechanical snubbers back into place. This caused a number of back injuries, pinched fingers, and larger dose to the snubber crews.

Remarks/Potential for dose limitation: Dose savings are based on the fact that the new snubber tool saves an average of 1 manhour per snubber for removal and installation.

References: None

Duration: from: 1990 to: 1991

Funding: $ 5,000

Status: In progress

Last Update: May 31, 1991

REMOVAL OF FINE CHROME PARTICULATE FROM SPENT FUEL POOLS BY MEANS OF A RADIAL LAMELLA

Keywords: CONTAMINATION REMOVAL; COBALT REDUCTION; FUEL

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Objectives: By means of a static stack of pagoda-shaped, nesting lamella plates, remove fine chrome particles of sub-micron size from spent fuel pools to improve visibility and limit dose. Utilizes no filter media, has no moving parts and particles can be collected in a removable sealed container. Unit can be completely decontaminated.

Comments: Although this device has been demonstrated on fine particles of less density, it would work better with chrome since it is a Stokes Law device. It would be necessary to conduct a Phase I Equivalency test before demonstrating it in a spent fuel pool. Phase I would utilize particles of equivalent size and density in equivalent water.

Would require teaming with a nuclear power facility.

Radial lamella elements are available for Phase I testing. Design and fabrication of removable sealed container is necessary.

Remarks/Potential for dose limitation: Since this system works 24-hours per day and is self-cleaning, the dose remaining in the pool should be limited to the chrome that is actually shedding at any particular time.

References: Fromson, R.E., "Advanced Clarification Based on Lamella Technology", Proceedings of the American Filtration Society, National Meeting, Chicago, IL, May 12, 1992.

Duration: from: 1992 to: 1993

Funding: $ 150,000

Status: Proposed

Last Update: April 16, 1992

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

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

This report presents information on 118 new or updated projects, covering a wide range of activities. Projects including steam generator degradation, decontamination, robotics, improvements in reactor materials, and inspection techniques, among others, are described in the research section of the report. The section on health physics technology includes some simple and very cost-effective projects to reduce radiation expsoures.

Included in this volume is a detailed description of how to access the BNL data bases which store this information. All project abstracts from this report, as well as many other useful documents, can be accessed, with permission, through our on-line system, ACE. A computer equipped with a modem, or a fax machine, is all that is required to connect to ACE. Many features of ACE, including software, hardware, and communications specifics, are explained in this report.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Nuclear Power Plants - Radiation Protection, ALARA, Radiation Protection - Research Programs Radiation Doses, Occupational Exposure, Indexes, Reactor Safety, Reactor Technology, Steam Generators, Nuclear Facilities, Decontamination, Reactor Materials Inspection, Robots

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