CONF-940505--

NUREG/CP-0143
BNL-NUREG-52440

Proceedings of the Third International Workshop on the Implementation of ALARA at Nuclear Power Plants

Held at
Hauppauge, Long Island, New York

Compiled by T.A. Khan

Sponsored by
U.S. Nuclear Regulatory Commission
Brookhaven National Laboratory

Proceedings Prepared by
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Manuscript Completed: December 1994
Date Published: March 1995

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Sponsored by
Division of Regulatory Application
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington , DC 20555-0001
NRC Job Code A3259

Brookhaven National Laboratory
Upton, NY 11973-5000

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Proceedings prepared by
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ABSTRACT

This report contains the papers presented and the discussions that took place at the Third International Workshop on ALARA Implementation at Nuclear Power Plants, held in Hauppauge, Long Island, New York from May 8 - 11, 1994. The purpose of the workshop was to bring together scientists, engineers, health physicists, regulators, managers and other persons who are involved with occupational dose control and ALARA issues. 175 persons from eleven countries attended the workshop. The countries represented were: Canada, Finland, France, Germany, Japan, Korea, Mexico, the Netherlands, Spain, Sweden, the United Kingdom and the United States.

The workshop was organized into twelve sessions and three panel discussions. The topics for these were as follows:

SESSION 1 - CONTROLLING RADIATION FIELDS

SESSION 2 - PANEL DISCUSSION ON RECENT RECOMMENDATIONS ON DOSE LIMITATION

SESSION 3 - PRESENTATIONS AND PANEL DISCUSSION ON ALARA IN NEW REACTORS

SESSION 4 - PATHWAYS TO ALARA

SESSION 5 - PANEL DISCUSSION ON ECONOMICS VERSUS EXCELLENCE

SESSION 6 - SHORT PRESENTATIONS ON ALARA IMPLEMENTATION

SESSION 7A - PWR AND CANDU PRESENTATIONS

SESSION 7B - BWR AND GAS-COOLED PRESENTATIONS

SESSION 8A - PWR AND CANDU PRESENTATIONS

SESSION 8B - BWR AND GAS-COOLED PRESENTATIONS

SESSION 9 - DECOMMISSIONING OF NUCLEAR POWER PLANTS

SESSION 10 - DECONTAMINATION OF NUCLEAR POWER PLANTS

SESSION 11 - ROBOTICS AND REMOTE HANDLING

The workshop was sponsored jointly by the U.S. Nuclear Regulatory Commission and the Brookhaven National Laboratory's ALARA Center.

EXECUTIVE SUMMARY

Introduction

The Brookhaven National Laboratory's ALARA Center and the U.S. Nuclear Regulatory Commission (NRC) periodically sponsor workshops on the implementation of ALARA at nuclear facilities. The third workshop in this series took place in Long Island from May 7 to 11, 1994. The gathering was truly international. The 175 participants from 11 countries included some of the world's foremost experts in their area. There were representatives from international regulatory bodies, safety institutes, power plant venders, utilities, contractors, consultants, and insurers. Organizations such as the National Council on Radiation Protection and Measurements (NCRP), Nuclear Energy Agency (NEA), Electric Power Research Institute (EPRI), Institute of Nuclear Power Operations (INPO), and Nuclear Energy Institute (NEI) were also represented. This wide and diverse attendance enriched the conference, and many different aspects of ALARA and radiation protection were discussed. Some of the main findings that emerged from the various sessions are presented below.

Opening Remarks and Session on Controlling Radiation Fields

The workshop was opened by Dr. Donald A. Cool of the NRC, who said that the ALARA Center grew out of the need to ensure that radiation exposures in the nuclear power industry are as low as reasonably achievable without the need for additional regulations. This need continues. In the session that followed some of the main points were:

There is now general consensus that pH control is one of the most cost-effective techniques available in reducing radiation fields in pressurized water reactors (PWRs). The question that has only partially been answered is the effect of the required elevated lithium concentration on fuel cladding corrosion. This question will become even more significant as utilities move to longer fuel cycles, requiring more lithium.

Swedish experts recommended that it is not only important to use enhanced pH to reduce radiation fields in PWRs but also to control it within a very tight band.

A Japanese paper suggested that maintaining pH in a narrow band could be accomplished by using automation in the control of pH. Automatic control of the pH has been introduced in Japan and is also available for other PWRs.

Zinc injection was shown to be a very successful and low cost technique to reduce radiation fields in boiling water reactors (BWRs). Preliminary data show good results for PWRs also. Moreover, the projected reduction in costs of depleted zinc will make it even more cost-effective.

Panel Discussion on Recent Recommendations on Dose Limitation

The panel was chaired by Charles B. Meinhold of the NCRP who explained the reasoning behind the new recommendations and then asked for a discussion on the subject. Some of the points that emerged were:

Questions were raised about the dichotomy between safety and dose control. Some participants thought that reductions in dose limits may have some adverse implications on safety. For example, less surveillance and inspections to save dose may result in reduced safety. This problem may require further study.

Comments were made on the importance of informing the public about such matters as the significance of exposures, about ALARA, combined risks, and how dose limits are set.

Dr. Mary Measures of the Canadian AECB raised the question of informed consent for women rather than regulations to protect the fetus. She thought that such a regulation would be very difficult to enforce, and Canadian women considered that it would have a discriminatory effect in that it would lessen their job opportunities.

Some participants thought that as new and stricter dose limits are imposed new anxiety is created in radiation workers. The perception is created that they were not being adequately protected in the past. It was thought that this again was an area where more could be done to inform the workers about how dose limits are set and how other risks compare with radiological risks.

Christopher Wood of EPRI proposed a simply written question and answer manual which could enhance worker understanding. A similar booklet may also be useful for public information.

Session and Panel Discussion on ALARA in New Reactors

The session on ALARA in New Reactors brought out a number of very important points:

If the new U.S. reactor designs apply the experience gained so far, annual collective dose per plant could drop very drastically from present values of hundreds of person-rem to perhaps a few tens of person-rem for even very large new reactors. However, the present design targets have so far been set rather high conservatively.

The U.S. NRC has shown foresight in leaving sufficient flexibility in the standard designs to allow venders to profit from the new lessons learned in the area of dose reduction in reactor design.

The reactor venders are making very good use of ALARA information and data in the design of advanced reactors. This lets the intent of the NRC, to have safer and more benign new reactors, to be fulfilled and yet allows the NRC to be less prescriptive in its rules and guidance.

The new German plants are setting the pace in dose reduction. Yearly collective dose equivalents in these plants are around 20 person-rem. The primary reason for this is the changes made by the Germans at the design stage. A secondary reason is the use of modern chemistry to reduce radiation fields.

Swedish, French, and U.S. plant doses are low, but the data show that there are some signs that plant doses may once again start to increase due to various reasons mainly involved with plant aging. Thus, ALARA surveillance, oversight and advice will be required to ensure that everything reasonable is being done to protect radiation workers.

The United Kingdom is expecting a yearly collective dose equivalent for Sizewell B of around 200 person-rem, having benefited from U.S. and German experience. Their next PWR, Sizewell C, will incorporate even more recent experience and doses have been conservatively estimated for this plant to be about 35 person-rem.

Panel Discussion on Economics versus Excellence

There was lively discussion during this panel. Some of the points made were:

Harvey Cybul of INPO (and Daniel Malone of the PWR ALARA Committee in a separate session) discussed the extreme importance of reducing cost in nuclear generation in order for the nuclear industry to survive.

Several speakers stressed the importance of doing more with less.

There was general consensus that better work planning was perhaps the most cost-effective way to reduce dose.

Alan Homyk gave an example where a 20% reduction in dose was realized due to better work planning techniques.

Sessions on Decontamination and Decommissioning

In the area of decontamination, the Canadian, German and U.S. presentations all illustrated the importance of proceeding as rapidly as possible with full-system decontamination with the fuel in place. Techniques are now available in the U.S., Canada, the U.K., and Germany which have been proven to be safe from a technological viewpoint. Only a few final wrinkles need to be ironed out. It was also illustrated during the workshop that one utility has agreed to conduct a pilot project for this task. This is very courageous since there is some risk still involved for components costing hundreds of millions of dollars. Other main points that emerged were:

Decommissioning was shown to be possible and can be performed at reasonable cost. It was shown that new plants designed for decreased radiation exposures will also be a lot easier to decommission.

One paper illustrated the importance of exercising great care in radiation protection for even old plants undergoing decommissioning, because a near major incidence occurred in the spent fuel pool at this very old plant. There was for a time the possibility of uncovering active fuel in the spent fuel pool.

U.S. plants are aging and decontamination and decommissioning are going to become major tasks. This implies that radiation protection and ALARA in these areas is going to become more and more important.

Miscellaneous Aspects

There were many other interesting presentations in the areas of operation, maintenance, robotics, remote handling, economics, ALARA criteria from the viewpoint of insurance, and the effect of respirators on worker performance. For example, in the session on *Pathways to ALARA* the new ALARA policy of Electricité de France was described. An assessment of the benefits and impacts on the U.S. nuclear industry of the hypothesized lower occupational dose limits was presented. There were presentations on the ALARA experiences of European installations, the NEA's Information System on Occupational Exposures (ISOE) and on the economics of radiation protection. A paper from Rolls Royce of the U.K. described six steps to a successful dose reduction strategy. In other sessions there were reports by the Chairpersons of the BWR and PWR Radiation Protection ALARA Committees on the work of their committees. The details are in this volume.

Conclusion

The conference showed that there has been a massive change in the extent of radiation doses that workers are receiving from nuclear plants since the time of the first workshop held in 1984. In all the countries represented at the workshop the doses have dropped very considerably. This has been largely due to the strong stress on ALARA in most countries, to the efforts of the industry, the leadership of the advisory agencies such as the ICRP and the NCRP and the insightful approach adopted by the regulatory bodies. It has resulted in a much more benign radiological environment for occupational workers. Moreover, research and development has so far advanced reactor design technology that the next generation of reactors are going to require very low annual collective exposures to service and maintain them. The slight upward trend in collective doses in some countries where the doses have hitherto been very low are some cause for concern. In order to keep improving the radiological climate for occupational workers and to ensure that decommissioning of the older reactors is safe and results in doses to workers that are as low as reasonably achievable, constant vigilance is still going to be necessary.

FOREWORD

The Third International Workshop on ALARA Implementation at Nuclear Power Plants was hosted by Brookhaven National Laboratory in May 1994. The workshop was attended by 175 participants from 11 countries, including representation from regulatory agencies and other organizations such as NCRP, EPRI, INPO, NEI, and NEA.

Topics discussed included control of radiation fields, ALARA in new reactor designs, and the economics of dose reduction and decommissioning. Discussion was extensive and several new findings were presented at the conference. PH control and zinc injection were identified by participants as especially cost-effective dose-reduction techniques. Considerable attention is being paid to dose control in the design of evolutionary and advanced reactors and significant reduction in operating dose costs are expected. A consensus is building that decontamination of present systems with fuel in place is feasible and could be very effective for U.S. designs. It was noted that reduced dose limits and the new more complex protection regulations will require improved training of workers and public information.

Both individual and collective doses have been declining in the U.S. for the past 10 years. There was some discussion during the conference to the effect that with increasingly tight budgets, even more cost-effective dose-reduction measures will have to be found. The general consensus was that we have not reached a point when dose reduction efforts can be relaxed.

The information, results, approaches, and/or methods described in this NUREG are provided for information only. Publication of this report does not necessarily constitute NRC approval or agreement with the information contained herein.

Donald A. Cool

Donald A. Cool, Chief
Radiation Protection and Health
Effects Branch
Division of Regulatory Applications
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CONTENTS

ACKNOWLEDGMENTS

This workshop was made possible by the continuing support of the U.S. Nuclear Regulatory Commission. The Brookhaven National Laboratory's ALARA Center wishes to thank our project manager, Alan Roecklein, for his support and encouragement. We thank the session chairpersons for helping to run the workshop. We would like to acknowledge with appreciation the advice and support of John Baum and Charles Meinhold of Brookhaven National Laboratory in organizing the workshop.

Lastly, we would like to express a special note of appreciation to the members of the small workshop organizing committee for their remarkable effort. Among these were Maria Beckman, the ALARA Center secretary, Grace Webster, Clifford Yu, James Xie, and Justo Estrada.

Tasneem A. Khan
Conference Coordinator

WORKSHOP OPENING ADDRESS

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Good morning, and welcome to Long Island, New York, and to the Third International Workshop on ALARA Implementation at Nuclear Power Plants. This is the third such workshop we have had. The previous two were held at Brookhaven National Laboratory. The first one was about ten years ago. At that time, the U.S. Nuclear Regulatory Commission was concerned that the doses at U.S. nuclear power plants were much higher than at most other countries throughout the world. They began funding three small projects at Brookhaven to look at the questions of how the U.S. compares to the rest of the world, to identify the high-dose jobs, dose-reduction techniques and so on, that could be used to reduce doses, and to determine if the doses at plants in the U.S. are as low as reasonably achievable.

In order to focus industry's attention on these efforts, we suggested the formation of what we call the "BNL ALARA Center," which has been functioning since that time as an information-gathering, analysis, and dissemination center focused on this question of dose control at nuclear power plants. Over the last six years, it has also served a similar function for DOE facilities. The NRC effort has been supported by the Radiation Protection and Health Effects Branch of the Nuclear Regulatory Commission. We have as co-chair of this session, Dr. Donald Cool, who is the Chief of this branch, which is in the Office of Nuclear Regulatory Research. Later on in this meeting, we will hear from our colleagues from the Department of Energy, who supply support for that part of the effort.

We are very grateful to all of you in the audience who have helped over these past ten years to reduce doses at the various sites. As a result of your efforts, the doses in the U.S. have come down about a factor of two per plant, while power generation per plant has increased by a factor of two, so there is a net increase of about a factor of four, which is certainly very commendable, and we think, hopefully, that this information exchange has contributed somewhat to that.

I am not going to tell you much about the ALARA Center -- most of you are familiar with it. There is a little brochure in the inside cover of your preliminary proceedings binder that describes the ALARA Center in more detail for those of you who are not familiar with our activities. We also have a booth outside which has a computer and fax machine set up so that we can demonstrate the various databases that can be accessed in our system. Demonstrations will include the simple one where you can fax information back to yourself with a fax machine without the need for a computer.

The purpose of the workshop is, of course, to continue this process of information exchange, and we'd like to encourage you to make new friends here, meet old acquaintances, and through this interaction, we hope you will optimize the process at the international level. We are interested not only in optimization at the plant level but also internationally, and the ALARA Center will continue to support that activity. We hope that you all will make use of it.

We would like to encourage anyone who has a question or a comment to come to the microphones in either the center or the side aisles. The sessions will be recorded, and you will need to identify yourself so that we can contact you if there are any questions about the question that you raised or the comment that was given. We applied for certification credits for those who are certified members of the American Academy of Health Physics. You will be given 16 credits for this meeting if you need them for your recertification. So with that brief introduction, I'd like to turn the meeting over to Dr. Cool.

THE FUTURE OF ALARA

Dr. Donald A. Cool
U.S. Nuclear Regulatory Commission
Division of Regulatory Applications
Washington, D.C. 20555

Good morning. On behalf of the Nuclear Regulatory Commission and my branch, the Radiation Protection and Health Effects Branch, I want to welcome you to this Third International Workshop on the Implementation of ALARA at Nuclear Power Plants. I am very appreciative of Brookhaven and the ALARA Center for hosting this for us and the work that they do for us associated with keeping track of all of the activities going on in the radiation protection community, nuclear power plants, and now branching into other areas in terms of doses, dose-reduction techniques that are available, work activities, and the variety of things that go into ALARA.

As the sponsors of this conference, we are tremendously pleased with the participation that we are seeing, both nationally and internationally. The program this week represents our continued emphasis on reducing exposures to individuals, to populations. It illustrates the application of ALARA to future reactor designs. There are people considering what they may build and what the next generation of facilities will look like, and also for designs of what we are going to do over the course of time as we begin to take plants off line and to begin to actually decommission facilities and what ALARA will mean in the context of decommissioning.

January 1 of this year, for those of you who are familiar with history, may not go down as another day that will live in infamy, but it was an important date in terms of the radiation protection community in the United States. It was on that date, just a few months ago, that licensees in the United States were required to implement the revised 10 CFR 20, which are our basic standards for protection against ionizing radiation. That revision incorporated the recommendations of the ICRP, International Commission on Radiological Protection, as presented in publication 26, and finally managed to bring the United States out of the Stone Age into perhaps the Medieval times with respect to the radiation protection philosophy.

Probably the most significant changes, from the perspective of the regulators and the licensees, are the reduction in the "limits" of occupational exposure by a factor of about three. The requirements actually sum internal and external exposures to get what we call in the United States a total effective dose equivalent and a requirement for ALARA as part of radiation protection programs. Yet these changes are not really all that significant from the perspective of the nuclear power plants. Doses at the nuclear power facilities have been a lot lower than the new numbers for a number of years. Why? The answer is ALARA. As you can see, the collective dose, and John Baum has already mentioned this, from the nuclear power plants in the U.S. has declined in the post-TMI era from a high in the 1983 time frame of a little bit over 560 Sv collective dose to less than 300 Sv in 1992. This decrease has been achieved even as the total number of facilities coming on line has increased. It is really a significant change in the total amount.

As a class, the boiling water reactors, the BWRs, have had a little bit farther to come and have made a tremendous amount of progress. In 1980 the average collective dose was just over 11 Sv. In 1992 it was down to less than 4.

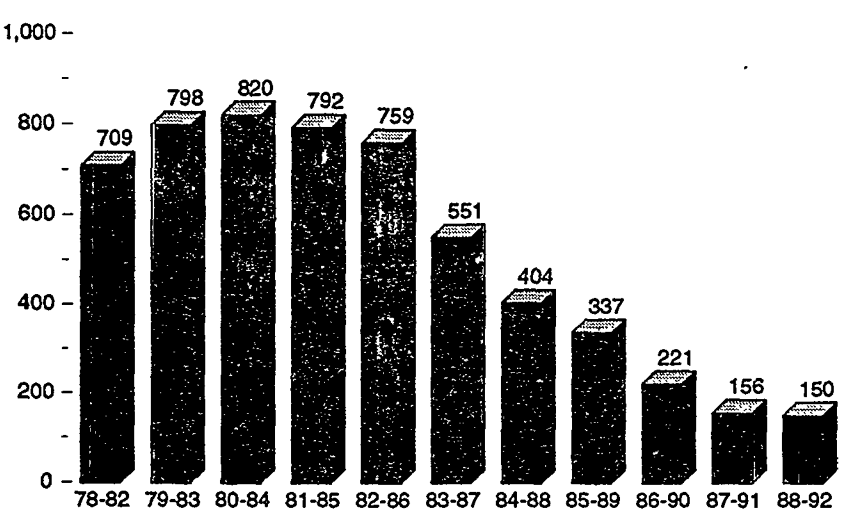
PWRs have also made tremendous progress. Their collective dose decreasing from a high of about 6.5 Sv in 1981 to just over 2 Sv in 1992. However, collective doses are not necessarily the whole story here. In addition to that, the average measurable doses for the individuals has declined, going from a high of slightly under 7 mSv in 1980 to less than 3 mSv in 1992. These reductions in dose are a tribute to the effort that the industry has undertaken, despite the fact that ALARA or the optimization process has, until very recently, only been in the U.S. regulatory term a "should," which means a lot of arm twisting, a lot of good will, and a lot of voluntary efforts to reduce exposures.

At a time when engineering efforts at some facilities, such as steam generator replacements, pipe replacements, different kinds of outages, are becoming almost common place, these kinds of improvements are really kind of remarkable. Use of ALARA reviews, job planning, and other kinds of activities are responsible for a large share of the credit. But the easy fixes have pretty much been accomplished. Further reductions in doses to individuals and groups are going to be harder to achieve in the future. However, as you are going to hear over the next three days, there are a number of techniques, a number of activities -- robotics and other kinds of advances -- that do hold promise for the future.

The U.S. NRC, and, in fact, the entire world, is now faced with the changes in radiation risk factors over the last 8 or 9 years, and resulting recommendations of the ICRP published in publication 60, and in similar reports from the United States' version, the National Council on Radiation Protection and Measurements. As you may know, incorporation of these recommendations into the regs could entail new, even lower occupational limits in order to effectively implement a long-term objective of less than 1 Sv of dose to an individual over the course of their working lifetime. But, once again, the data show that the reality of exposures in the workplace are already significantly less than the recommended limits.

This chart shows the number of individuals whose 5 year dose exceeds the 100 mSv ICRP 60 recommendation for a 5-year averaging period. This is actually a rolling sort of average, so it changes by one each year. As you can see, this number has gone from a high of about 820 or so individuals in the years immediately following the TMI incident to only about 150 for the latest time period, 1992. The U.S., as with most countries, requires submitting data following the year, so we have the 1993 data which is currently being submitted to the Commission. The reporting period ends in April, so we should have that analysis available in the August type of time frame for 1993. I expect that this number, the number of individuals who have a 5-year average over 100 mSv, will continue to decline. In fact, out of the over 200,000 monitored individuals in the U.S. in nuclear power, only 482 had doses greater than the 20 mSv average value in 1992.

Individuals with Doses >100 mSv in 5 yrs



ALARA in the 80s meant reducing wasted dose, which resulted in both lower collective doses and lower individual doses. ALARA in the 90s, as we continue to move forward, is likely going to mean something closer to optimizing doses to individuals and groups. We are going to have situations where we are going to have higher dose areas. Plants get older, and as you begin to decommission facilities, there will no doubt need to be trade-offs that have to be carefully analyzed. It may, in fact, no longer be possible in all cases to have both, as they say in the United States, "have our cake and eat it too." That is, reduce the individual and collective dose simultaneously.

As we enter the 21st Century, we are being faced with new challenges. Decommissioning of those facilities that were constructed during the boom years of the 60s and the 70s -- the current generation of plants -- is going to require us to learn how to reconcile exposures to different generations as well as different groups of individuals. Questions and issues are going to be raised and have already been raised. Reducing doses to workers versus reducing doses to the public of future generations. Or as another example, is it more dose effective or environmentally sound, if you will, to move radioactive material from one place to another or to leave materials at sites which may perhaps never be completely restored to their preexisting condition, or which for some reason such as the available infrastructure of power generation, continue to be used in an industrial setting and, therefore, might not need to be taken back to what, in the United States, is sometimes referred to as "green field," that is, returning it to entirely the way it was before anything was built on the site.

Another question will be the scope of consideration of risks to be included in the ALARA process. We have had the relative luxury, up until now, of being able to look at occupational doses, pretty much just within the context of whose on the site, or public doses in the context of what leaves the site as effluents. Now we're going to have to broaden that to consider things such as transportation of materials as we begin to decommission facilities and consider whether or not to move materials off the site.

These types of considerations have not been part of the typical analysis that has been conducted in the past. And then there is the issue of timing. Should we, for example, allow, or perhaps even require, twenty or fifty years of component decay before we start dismantling the facilities. If we do that, it means that workers who ultimately do the decommissioning will be working in an unfamiliar environment. In fact, it will be a whole other generation of workers -- workers who are not part of this culture familiar with these facilities. At question will even be what ALARA itself means in the context of the multiattribute analysis environment that we are now doing decisions in the U.S. and around the world in a climate of competing demands and uncertainties. All of the safety implications are going to need to be explored and certainly not all of them can be quantified.

ALARA has saved the industry a great deal over the years. Lower doses has meant lower risk of litigation, fewer workers, reduced overhead -- in general, a better operating environment. The real challenge for the next century is to help the public understand what ALARA really means. It's no longer going to be enough to optimize exposures or even reduce worker or public risk. We must now begin to remove the mystery that seems to be so prevalent outside the radiation protection community about this "thing" we call ALARA. ALARA has the potential to restore public faith in the use of nuclear materials, but only if it is understood as the tool that everyone wants. Namely, reducing risks and taking everyone's interests into account. This has really been brought to mind so clearly in the United States as the NRC has been pursuing an enhanced rule-making process for the radiological criteria for decommissioning. We have spent a tremendous amount of time over the past year going around the country in a series of workshops to get early public input into the kinds of considerations that ought to be placed into these criteria. It was a tremendous eye-opening experience, both the range of viewpoints and what some of those viewpoints were. And they would not be as you might characterize them, all of the environmentalist citizen groups being very negative. Some of them were very, very positive. Some of them were very concerned in a global sense about having material spread around versus leaving it in places where they knew where it was.

As long as ALARA is perceived as a chance to do less than could be done, it will not be accepted. This fact has been made clear, as I said, in our dealings with environmental and community groups. In waste management, in decommissioning, in license renewal, in construction of new facilities -- the job of nuclear professionals, you and I, should be to begin to recover the trust that has been lacking in all the nuclear and anti-nuclear activities taking place. Without public support, this industry is likely to die. Acceptance is the key to whether or not it will continue. We need to use the successes of ALARA to help us engender public trust to allow continued responsible use of radioactive materials in the 21st Century.

Once again, I welcome you to this international workshop and look forward to some of the things that we will be hearing over the next three days.

SESSION 1

CONTROLLING RADIATION FIELDS

Co-chairs:

Donald A. Cool
John W. Baum

RECENT DEVELOPMENTS IN RADIATION FIELD CONTROL TECHNOLOGY

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INTRODUCTION

The U.S. nuclear power industry has been remarkably successful in reducing worker radiation exposures over the past ten years. There has been over a fourfold reduction in the person-rem incurred for each MW.year of electric power generated: from 1.8 in 1980, to only 0.39 person-rems in 1991 and 1992. Preliminary data for 1993 are even lower: approximately 0.37 person-rem/MW.year. Despite this substantial improvement, challenges for the industry remain. Individual exposure limits have been tightened in ICRP 60 and there will be increased requirements for special maintenance work as plants age, suggesting that vigorous efforts will be required to meet the industry goals for 1995.

Reducing out-of-core radiation fields offers the best chance of continuing the downward trend in exposures. To assist utilities select the most economic technology for their specific plants, EPRI has published a manual capturing worldwide operating experience with radiation-field control techniques (TR-100265). No one method will suffice, but implementing suitable combinations from this collection will enable utilities to achieve their exposure goals. Radiation reduction is generally cost-effective: outages are shorter, manpower requirements are reduced and work quality is improved. Despite the up front costs, the benefits over the following 1-3 years typically outweigh the expenses.

RADIATION EXPOSURE SOURCES

Occupational exposures are the product of the time spent in the radiation field and the radiation intensity (dose rate). The former is determined by the amount of work to be done, the efficiency with which the task is carried out, and the extent to which remote technology is utilized. Radiation fields result primarily from activated cobalt isotopes; the dose-rate is determined by the amount of cobalt used in valves and materials of construction, control of water chemistry to limit transport and activation of the cobalt, condition of out-of-core surfaces in the primary system, which determines how much cobalt is deposited, and the extent to which decontamination is utilized.

Cobalt isotopes are the main cause of exposure; radioisotopes from failed fuel are but a minor contributor. Cobalt is activated to Co-60, the dominating gamma emitter. The widespread use of cobalt-base hardfacing alloys in U. S. plants and the higher cobalt impurity levels in construction materials are key reasons why U. S. plants have higher radiation fields than Swedish or modern German plants. The first goal of the radiation protection manager is, therefore, to replace cobalt hardfacing alloys whenever possible and to specify low cobalt impurity levels in ordering replacement components.

Cobalt-58, produced by the activation of nickel in corrosion products released by stainless steel and nickel-base alloys, is the second most important exposure source. The impact of both cobalt and nickel sources can be minimized by selecting water chemistry to minimize the release, transport, and activation of wear and corrosion products. In limited areas, such as the chemical volume control system (CVCS) in PWRs, extra-fine filters can be helpful. However, their benefit is more local than circuit-wide.

Activated corrosion products become a problem when they deposit on out-of-core surfaces, particularly in areas such as PWR channel heads and around valves, where inspection and maintenance work is performed. Preconditioning the surfaces of replacement components helps reduce activity pickup. If all else fails, chemical decontamination can typically remove 90% of the deposited corrosion products.

TECHNOLOGY TO HELP REACH FUTURE GOALS

Radiation control technology can be divided conveniently into three categories: established or mature techniques, recently-developed techniques that are now available for plant demonstrations, and the developments that are promised for the future.

Established Techniques

Cobalt Reduction Guidelines

A close look at valve duty in nuclear plants has pinpointed conditions where the use of the cobalt-base Stellites™ as a hardfacing alloy is not warranted. The latest results are described in the *Cobalt Reduction Guidelines, Revision 1*, published in 1993 (TR-103296). Implementation of these findings affords utilities an opportunity to reduce personnel exposures.

The recommendations of the guidelines have been implemented by Niagara Mohawk Power personnel who ordered 150 replacement globe valves with precipitation-hardened stainless steel seating surfaces. These valves are used in manifolds that provide differential pressure measurements. New York Power Authority has replaced major cobalt contributors, such as charging pump check valves, with cobalt-free valves that are performing well.

Niagara Mohawk also has purchased and installed at Nine Mile Point Unit 1 replacement control blades and local power range monitors fabricated from stainless steel containing very low levels of cobalt (150 ppm and 250 ppm, respectively). This should lead to reduced fields and reduced low level waste disposal costs.

PWR Primary Chemistry Control

Released Co-59 must be activated in the reactor core by its incorporation into the corrosion products that deposit on fuel rods. A wide range of experiments showed that the amount of activated Co-60 is reduced if the lithium concentration is increased in the primary coolant so that the pH exceeds 6.9. However, laboratory investigations show that very high lithium concentrations can increase the corrosion rate of Zircaloy fuel rod cladding and the susceptibility of Inconel 600 steam generator tubing to intergranular stress corrosion cracking. Thus, the benefits of lower radiation fields must be carefully weighed against possible degradation of critical reactor components. The *PWR Primary Water Chemistry Guidelines: Revision 2* provide a way to avoid the pitfalls (NP-7077). The key point is to operate at or above pH = 6.9 and as close as possible to pH = 7.4 to minimize corrosion-product deposition and accelerated Zircaloy corrosion.

Three new PWRs, Vogtle 1, Comanche Peak 1, and Seabrook have used a modified coolant chemistry regime from startup. Here the pH is maintained at 6.9 early in the cycle until a lithium concentration of 2.2 ppm is reached, which is maintained until a pH of 7.4 is achieved. After about 1 EFPY of operation the average SG channel head dose rate at these units is 4.2 R/h, which compares to a value of 6.4 R/h at other similar PWRs that have operated using pH 6.9 chemistry.

Control of shutdown chemistry using peroxide addition and early boration to minimize activity transients is discussed in *PWR Primary Shutdown and Startup Chemistry Guidelines* (TR-101884).

BWR Zinc Injection

When BWR radiation fields measurements were categorized according to the type of condensate treatment system and the alloy used in the condenser tubing, it was found that soluble zinc inhibited the corrosion of stainless steel and reduced the incorporation of Co-60. The lead utility in applying this technology was Public Service Electricity and Gas, which injected zinc from startup at its Hope Creek unit in 1986. The results of fuel examinations after three fuel cycles on zinc show that the zirconium oxide corrosion thickness on the fuel is in line with other BWR fuel experience. Some units have seen increased fields due to Zn-65 and technology to deal with this problem will be discussed later. Currently ten U. S. units are injecting zinc.

Zinc injection can help minimize the increase in shutdown radiation fields observed in some plants when hydrogen injection is implemented to control intergranular stress corrosion cracking. The *BWR Water Chemistry Guidelines, 1993 Revision* (TR-103515) discusses the options of water chemistry, including zinc injection.

Electropolishing Replacement Components

Laboratory, loop, and plant tests have shown that ex-core components incorporate less radioactivity if the surface is smooth. The earliest application of electropolishing to reactor components was on replacement BWR recirculation piping that was installed at Northern States Power's Monticello plant and Omaha Public Power District's Cooper plant. Subsequently, all BWR replacement recirculation piping has been electropolished.

Replacement PWR steam generator channel heads make use of three structural alloys. Programs to qualify electropolishing investigated prototypical materials and processes used by steam generator fabricators. No adverse results were found, and results on test coupons exposed in European PWRs indicated that electropolishing of representative weld overlay alloys would reduce radioactivity pickup by about a factor of three. In the U. S. channel heads have been electropolished at Northeast Utilities' Millstone-2 and at Consumer Power's Palisades unit. Northeast Utilities expects to avoid at least 18 man-rem per outage.

Part System Decontamination

Decontamination of recirculation systems at BWR plants has become almost routine at many utilities. The number of recirculation system decontaminations has doubled in the past two years, mainly as a result of the increase in fields observed in units that have implemented hydrogen water chemistry as a measure to mitigate IGSCC. The LOMI process has been used for all recent recirculation piping decontaminations. The CAN-DEREM and CITROX processes have been used for other BWR systems and PWR components. Decontamination developments are reviewed in another paper at this workshop.

Recently-Developed Techniques

High-Performance Cobalt-Free Hardfacing Alloys

Field tests of a new iron-base hardfacing alloy in key nuclear plant valves have been initiated. PWR utilities, including Consolidated Edison, Union Electric, and Houston Lighting and Power, are using NOREM trim in small gate or globe valves with isolating functions in the chemical and volume control system. The BWR utility, Boston Edison, is using NOREM in a large 12" gate valve that is being used to regulate feedwater flow. Successful performance will provide further confirmation of the extensive laboratory and loop test data showing the EPRI iron-base alloys, designated NOREM, have wear resistance matching the cobalt-base Stellite alloys. Licensees have produced weld consumables in the form of powder and wire, and valve vendors have developed welding procedures for these product forms. In addition to reducing the cobalt inventory, evaluations by the EPRI NDE Center showed that the NOREM alloy wire can be deposited by gas tungsten arc welding on carbon and stainless steel substrates without preheating. This advantage should facilitate valve refurbishing operations in the field, further contributing to exposure reduction.

Replacing Cobalt Pins and Rollers in BWR Control Blades

The first plant demonstration of equipment to replace the upper pins and rollers in BWR control blades took place at Commonwealth Edison's La Salle site in mid-June. These cobalt-bearing sources are a significant dose source because they operate in a high radiation field. The ability to remove these radiation sources in blades with remaining neutronic life is an attractive alternative to their premature discharge. It is expected that up to eight blades could be modified daily using a single work station. Similar equipment designed by GE has been demonstrated recently at KKM Muehlberg. TVA's Browns Ferry Unit 3 plans to change out all blades using this equipment in 1994.

BWR Zinc Injection Using Depleted Zinc-64

The main technical disadvantage associated with zinc injection is the formation of Zn-65 as a result of the activation of naturally-occurring Zn-64. Plant evaluations of zinc injection using zinc depleted in Zn-64 have started. Although depleted zinc is relatively expensive, the technology promises to be cost effective. Plant evaluations are being carried out at New York Power Authority's Fitzpatrick plant and Northern States Power's Monticello plant, both of which have experienced higher radiation fields since adopting hydrogen water chemistry, and other plants, including Millstone 1, operating on normal water chemistry.

PWR Enriched Boric Acid

An alternative way of increasing PWR primary system pH is to use boric acid enriched in B-10. Naturally occurring boric acid contains about 20% B-10, so the same nucleonic effect would be achieved with less boric acid. This, in turn, allows the desired pH to be obtained with less lithium.

As with depleted zinc, the main impediment to the use of enriched boric acid is economics. It is estimated that because of the high start-up costs, benefits will be realized only after several years of operation. However, it appears that enriched boric acid would be economically feasible for PWRs with boron recycle systems, especially at those plants operating on extended fuel cycles that require high boric acid concentrations after refueling.

Full System Decontamination

Current technology requires isolation of the part of the system that is to be decontaminated, such as recirculation piping in BWRs or steam generator channel heads in PWRs. Decontamination of the complete coolant system would provide a number of advantages, including improved decontamination factors, reduced recontamination rates, and lower background fields. Two qualification programs, one for PWR and one for BWRs, have recently been completed by groups of utilities and EPRI. No unresolved safety issues were found, and the economics appears to be feasible for both PWR and BWR applications, particularly for steam generator replacement in PWRs and the removal of in-vessel corrosion products in BWRs, which would otherwise be redistributed to out-of-core surfaces after implementing hydrogen water chemistry. Detailed engineering evaluations are now under way for full system decontamination of Consolidated Edison's Indian Point 2 PWR, the objective being to carry out the first demonstration in 1995.

Future Developments

Chromium Coatings

As part of the steam generator replacement project at Millstone 2, Northeast Utilities has installed one of the manway seal plates coated with a thin layer of electroplated chromium followed by pre oxidation in moist air. Activity measurements will be made in 1994. An RHR pipe at Diablo Canyon has also been treated. Chromium coated RWCU pipe sections will be installed at the Peach Bottom BWR units in 1994. The impetus to test this coating is data obtained from small specimens that were similarly coated and attached to the manway seal plates in the Doel 2 PWR. Dose rate measurements show substantial reduction over specimens that had been electropolished and pre oxidized. Thus, this metal coating provides an opportunity to reduce dose rates and hence exposures in the vicinity of channel heads during outages. Other possible applications of chromium coatings include the carbon steel piping that is being replaced in some BWR reactor water cleanup systems.

Zinc Injection for PWRs

Tests in an out-of-reactor loop at Chalk River showed that zinc reduced Co-60 deposition under PWR chemistry conditions, just as it did under BWR chemistry. Other workers followed up on earlier corrosion test results which suggested that the presence of a few ppb zinc reduced intergranular stress corrosion cracking (IGSCC) under BWR normal water chemistry. It appears that zinc additions delay the onset of PWSCC in Alloy 600 and may even reduce crack growth rates. This could be an important benefit of zinc injection in view of the increasing incidence of PWSCC in steam generator tubing and recent observations of degradation of Alloy 600 pressure vessel penetrations. In fact, mitigation of penetration cracking is now the main motivation for a major industry program now underway in the United States.

The concern about PWR zinc injection is that unlike the case with BWRs, no plant data is available the indicates the effect of zinc on the corrosion of Zircaloy fuel rod cladding. PWR applications require similar data, and EPRI and other utility groups have initiated two courses of action: accelerated loop tests at Halden where the effects of both heat flux and neutrons on Zircaloy corrosion can be assessed and a plant demonstration at Farley PWR including fuel surveillance after each cycle, as was done for the elevated lithium evaluation at Millstone 3.

The Farley plant demonstration will use natural zinc. However, the data obtained from Farley should enable a cost/benefit assessment to be carried out on the use of depleted zinc-64 for radiation field control.

Full System Decontamination Including Fuel

Ideally, fuel system decontaminations would be carried out before refueling, as this would significantly reduce recontamination rates and save critical path time. Encouraged by the results from the successful test on Quad Cities BWR fuel, four fuel assemblies from the V.C. Summer PWR were decontaminated during the 1991 outage. Two of these assemblies were reinserted for an additional cycle of exposure and were then examined. No adverse effects were found with either AP/CANDEREM or AP/LOMI. These promising results have prompted industry groups to consider a PWR full system decontamination with the fuel in place, as a follow-on to the 1995 Indian Point-2 fuelout demonstrations.

CONCLUSIONS

Technology developments in radiation field control continue to occur at a rapid pace, with an accelerating rate of implementation at nuclear power plants as utilities move to meet 1995 exposure goals.

Author Biography

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PAPER 1-1
DISCUSSION

Unknown: In some different presentations that I've heard in the past few conferences, and repeated here, we talk about exposures having bottomed out. I was wondering if anyone has been doing any research or compiling any data because I think, that although it's true that we've bottomed out, I still think that we are forgetting about some certain basic things about why it's bottomed out. What I would like to know is are we still looking at exposures due to poor work practices. They're still out there believe it or not. We're still doing some poor things in planning, poor things in scheduling and such. How much exposure is due to what may be termed as "unreasonable regulation" and that type of thing? My feeling is that there is still about 20% reduction due to those things.

Wood: EPRI is not doing any work in that area, but maybe John can answer that.

Baum: Of course, we collect information from the plants as we can, but I don't have any particular response to that question. Does anyone else from the PWR or BWR Owner's Group, for example, who meet frequently on these questions, have an answer?

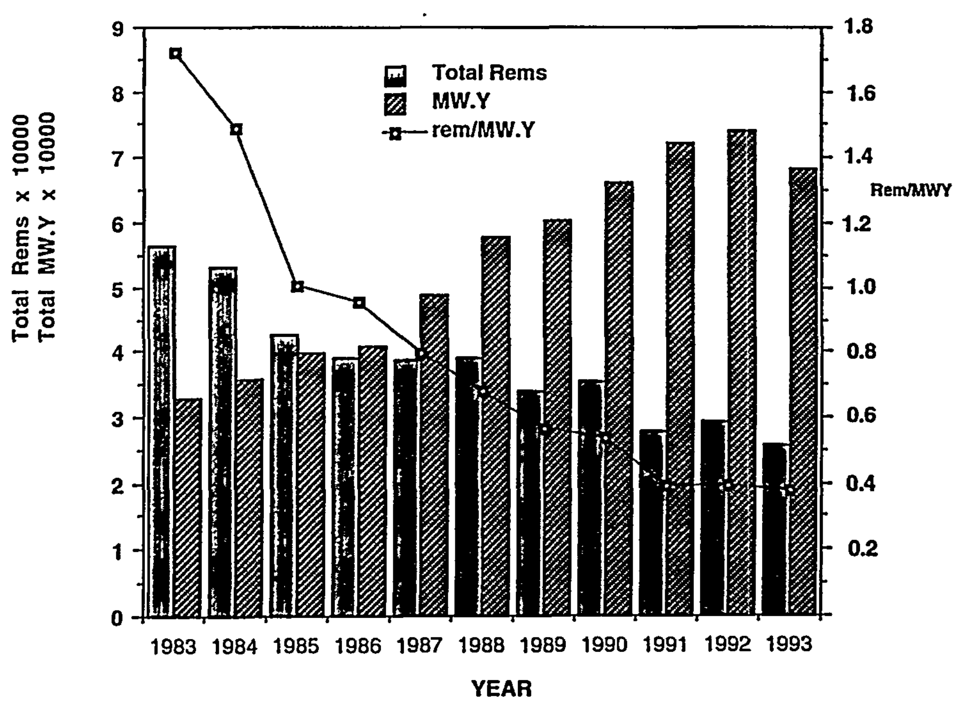
Cybul: From INPO's perspective, I agree 100% that there is a lot of room for improvement and much yet to be gained in the work practices area. We're putting a lot of emphasis into that. I think we will do that, but we still need to do the technical side.

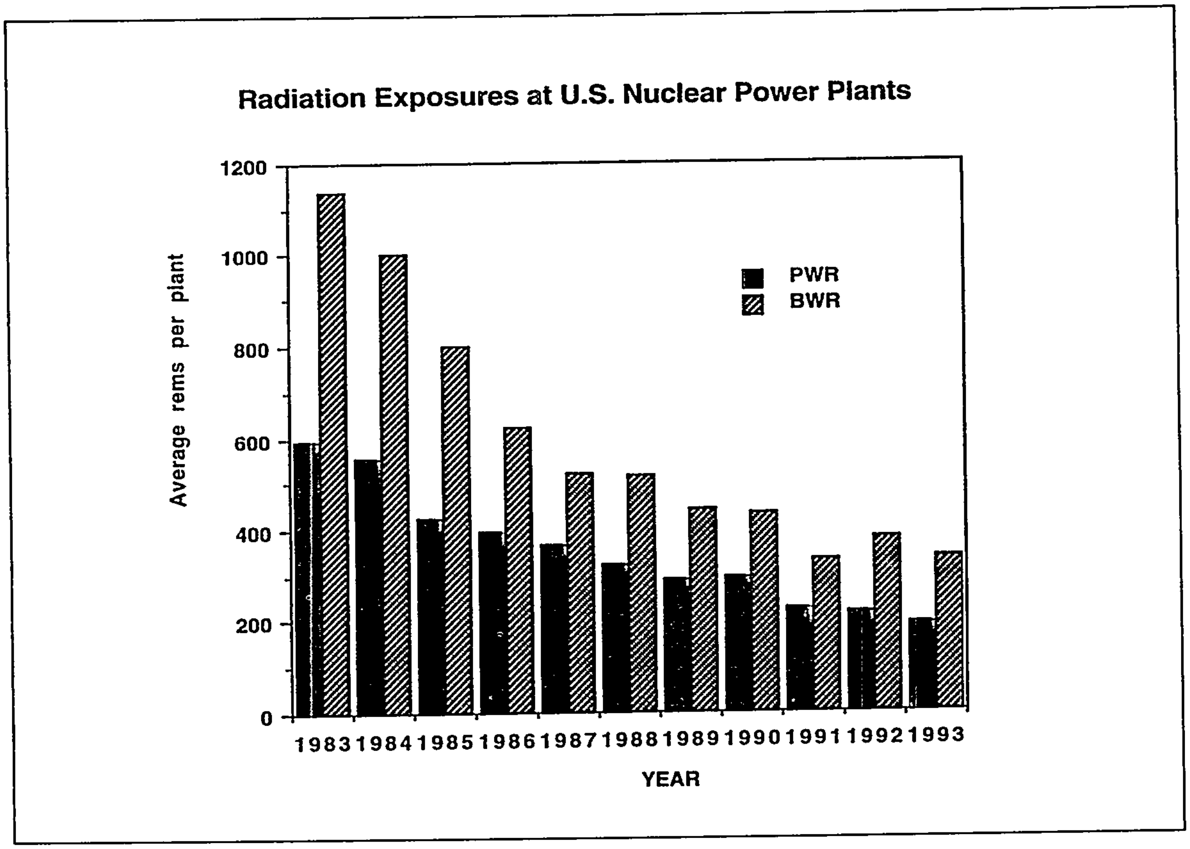
Baum: The Nuclear Energy Agency held a workshop on that subject about a year ago and there has been a publication on that which you might find helpful.

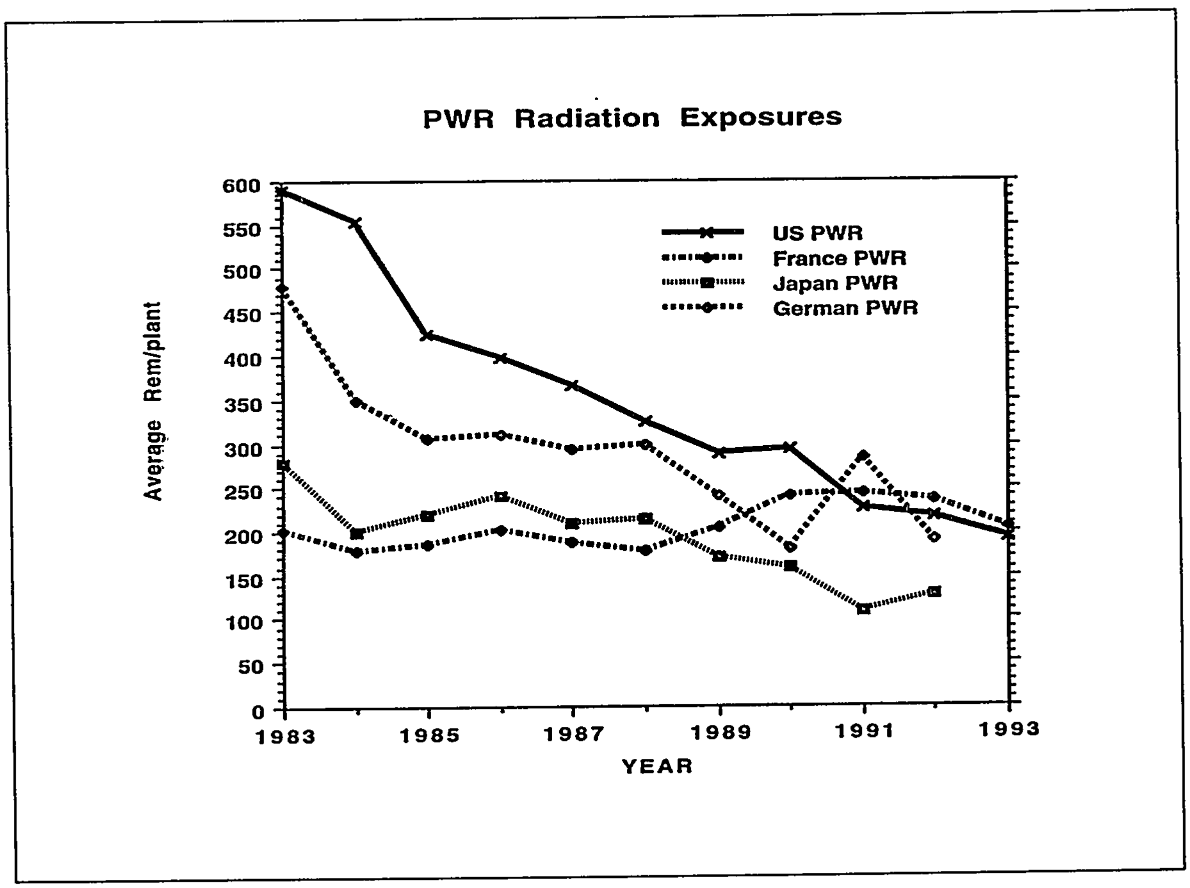
Radiation Exposure Trends

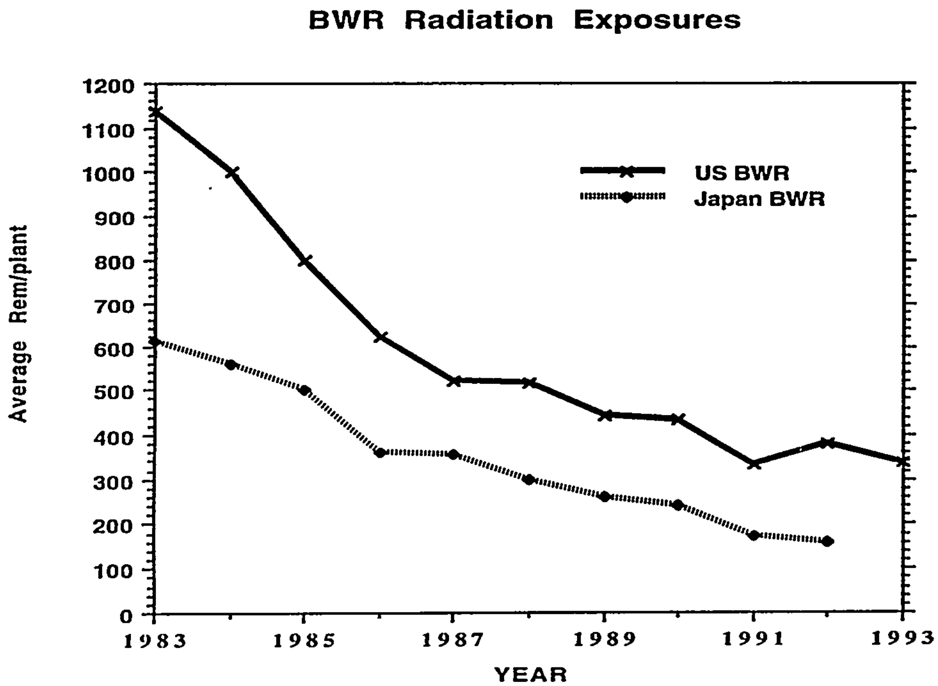
- Factor of 4 reduction in person rem per MW.year power generated over past 10 years
- Plant exposures continue to decline; comparable to France, Germany and Japan - Sweden continues to set the standard
- Significant reduction in numbers of workers exposed to radiation, especially those receiving more than 2 rems
- 1992-3 information suggests that exposures may have bottomed out

U.S. Nuclear Power Plant Occupational Exposures and Electric Generation




Radiation Exposures at U.S. Nuclear Power Plants
Average rems per plant
1200
1000
800
600
400
200
0
PWR
BWR
1983 1984 1985 1986 1987 1988 1989 1990 1991 1992 1993
YEAR


PWR Radiation Exposures
Average Rem/plant
600
550
500
450
400
350
300
250
200
150
100
50
0
US PWR
France PWR
Japan PWR
German PWR
1983
1985
1987
1989
1991
1993
YEAR



Recent Trends and Future Challenges

- Plant performance has improved significantly in past 5 years
  - Capacity factor
  - Outages due to corrosion-related problems
  - Radiation exposures and numbers of workers exposed
- Recommended exposure limits tighter (ICRP-60, NCRP-91)
  - 10 rem / 5 years, cumulative rems < age in years

More Challenges

- Corrosion-related problems increasing as plants age
  - Steam generator replacements
  - BWR core internals repairs
- Emphasis on reducing O&M Costs
  - Tighter budgets, reduced staffing levels

(note, however that past staffing increases have been in non-exposed workers)

Reducing the Source Term

- NOREM cobalt-free hardfacing alloy
  - Valve endurance tests under PWR and BWR conditions completed; outperforms Stellite
  - Welding repair procedures developed by EPRI NDEC, GE and Welding Services
  - Four nuclear plants installing NOREM valves in 1993
  - NOREM weldability demo held at NDE Center, Charlotte, August 1993

Reducing the Source Term (cont)

- BWR control blade pin/roller replacement

ABB Combustion Engineering approach demonstrated at CECo's LaSalle plant in 1992 (Report No TR-101837)

GE equipment demonstrated at KKM Muehleberg

Results show that key in-core sources can be removed in cost-effective manner

Reducing the Source Term (cont)

- Revision 1 to Cobalt Reduction Guidelines:

Specifies hardfacing requirements based on valve's duty cycle

Identifies current utility use of alternative hardfacing alloys

Documents cobalt-free hardfacings offered by valve vendors

Except for 89-10 valves, default position will be to specify cobalt-free valves, unless exception is document

Update cobalt impurity specifications for materials used in replacement components

Issued December 1993: TR-103296

Controlling Recontamination

- Electropolishing demonstrated on replacement steam generators for Millstone-2. EPRI Report TR-100559
- Improved process qualified and applied at Watts Bar
- Including electropolishing in the specification for replacement SGs results in minimal additional cost, and could result in major exposure savings should man entry be necessary in the future
- BWR replacement piping recontamination is best controlled by a combination of electropolishing and 560F air oxidation

Controlling Recontamination (cont)

- Stabilized chromium passivation for PWRs - *new product*
  - Doel PWR steam generator man-way covers: stainless steel recontamination reduced by factor of 5 - 10
  - Applied to Millstone-2 man-way cover seal plates and RHR piping at Diablo Canyon
  - Economical and effective way of controlling contamination of replacement components
  - Available from RCT

Reducing Corrosion Product Activation and Transport PWR Primary Chemistry Control

- PWR primary chemistry shutdown guidelines: TR-101884
  - Guideline prepared by industry committee
  - Objective was to provide standard procedure, covering cooldown, boration and oxygenation
  - Product lists set of chronological principles for utilities to use in defining plant-specific program
  - Cause of unexpected activity releases in mid-cycle shutdowns determined: pH increase on connecting RHR system. Solution: isolate RHR before lithiating at startup

Reducing Corrosion Product Activation and Transport PWR Primary Chemistry Control

- Revision 2 of Guidelines (1990) still applies - latest data on effects of lithium do not change recommendations

  Zircaloy corrosion: 3.5ppm lithium may be 10-15% worse than 2.2ppm (Millstone/North Anna comparison)

  Inconel 600: 3.5ppm worse than 2.2ppm, but 2.2ppmbetter than 1 ppm (Japanese data)

  pH 7.4 reduces radiation buildup by 20% (see graph)
- Conclusion: Modified chemistry (pH6.9 - 7.4) best choice

Reducing Corrosion Product Activation and Transport
New PWR Techniques

- Enriched Boric Acid: use of enriched B-10 avoids the problems of Li / B / pH optimization but high up front cost
  - to be used at German Phillipsburg PWR
- PWR Zinc Injection: originally proposed as radiation control measure, zinc is now being evaluated on aggressive timescale as inhibitor of Inconel cracking - vessel penetrations and PWSCC of steam generator tubes
  - coordinated qualification and demonstration program with vendor owners groups initiated by EPRI

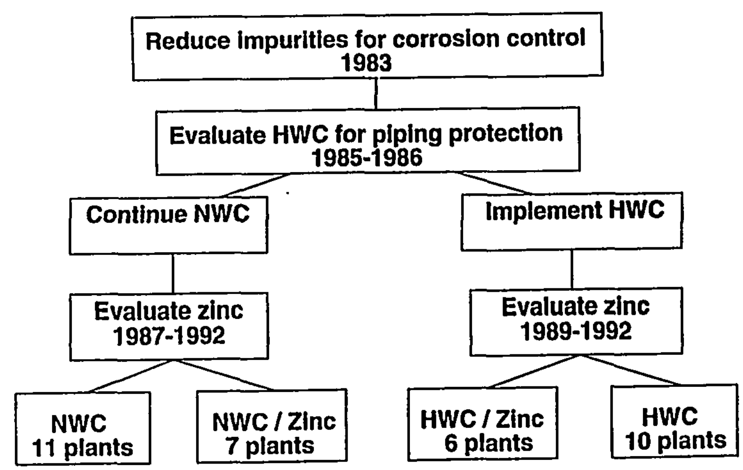
Reducing Corrosion Product Activation and Transport
BWR Primary Chemistry Control

- Hydrogen water chemistry causes transient increase in radiation fields
- Utilities need HWC to reduce IGSCC and IASCC
- Exposures will be worse (both N-16 and Co-60) for protection of core internals
- Zinc-65 is major contributor with zinc injection
- Depleted zinc-64 tests at Leibstat, Fitzpatrick, Monticello, Millstone, hatch, others

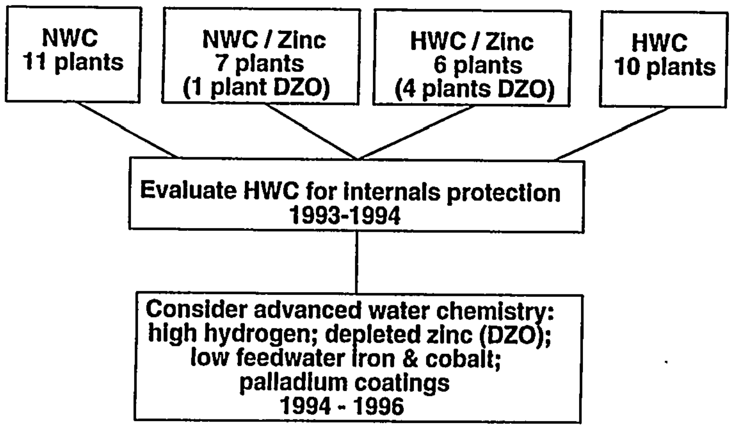
BWR Normal and Hydrogen Water Chemistry Guidelines: 1993 Revision

- Published February 1994: TR-103515
- Committee produced one Guideline, covering both normal and hydrogen water chemistry
- Key issue is high HWC to protect reactor internals
- Changes discussed: chromate recommendations, zinc injection, silica, soft shutdown, HWC issues
- In-core HWC tests will not be completed till 1994; possibility of updating 1993 Revision in 1995
- As with recent PWR guidelines, range of options offered; guidance on costs to be provided for first time

Evolution of BWR Chemistry 1983 - 1993



Evolution of BWR Chemistry 1993 - 1996



EPRI Tailored Collaboration Projects

- PWR full system decontamination engineering design and demonstration
- BWR control blade cobalt replacement (complete)
- BWR depleted zinc for radiation control
- PWR channel head electropolishing (complete)
- BWR feedwater purity control
- BWR hydrogen water chemistry radiation control

BWR Radiation Field Control Measures

- In use - immediate impact

  Chemical decontamination, together with replacement of control blade pins & rollers and zinc injection

  Install cobalt-free feedwater flow control valves

  Valve maintenance procedures to remove Co debris
- In use - slower impact

  Pins and roller replacement and zinc injection without decontamination

  Electropolish / precondition replacement components

  Cobalt replacement guidelines

New BWR Radiation Field Control Measures

- Now available - rapid impact

  In-situ pins and rollers replacement

  Depleted zinc-64 injection, avoiding excessive zinc-65

  Full system decontamination including vessel

  NOREM cobalt-free hardfacings for valves (e.g. MSIVs)

PWR Radiation Control Measures

- In use - immediate impact

  Chemical decontamination, together with elevated pH primary chemistry (2.2ppmLi, pH7.4)

  Zircaloy fuel grids

  Valve maintenance procedures to remove Co debris
- In use - slower impact

  Elevated pH and zircaloy fuel grids without decontamination

  Electropolish replacement steam generators

  Cobalt replacement guidelines, NOREM valves

New PWR Radiation Control Measures

- Available now - immediate impact

  Full system decontamination

  Enriched boric acid for primary system chemistry
- Under development

  Full system decontamination including fuel

  Chromium preconditioning of steam generators

  PWR zinc injection

Summary

- Industry exposure trends look great through 1991, but early 1992 data suggests exposures may have bottomed out; increasing pressure anticipated in the future resulting from repair needs
- Goal is technology that reduces exposures <u>and</u> O&M costs
- Technologies described in 1991 Manual (TR-100265)
- Latest developments and utility experiences discussed at EPRI Radiation Field Control Seminar, held in Seattle, August 16-18, 1993.

New Information from French Conference April 25-27, 1994

- Concern in Japan and Sweden about increasing BWR radiation fields on switching to extended fuel cycles
- German PWRs with cobalt removed and modified pH continue to have very low exposures
- French PWRs showed little benefit from pH7.1-7.2 but plan to use it anyway at remaining plants
- CANDU tests show benefit of zinc in reducing field buildup (similar results to our PWR loop tests)
- UK analysis concluded no acceleration of zircaloy corrosion with 3.35 ppm lithium in Millstone-3 tests

OPTIMUM WATER CHEMISTRY IN RADIATION FIELD BUILDUP CONTROL

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ABSTRACT

Nuclear utilities continue to face the challenge of reducing exposure of plant maintenance personnel. GE Nuclear Energy has developed the concept of Optimum Water Chemistry (OWC) to reduce the radiation field buildup and minimize the radioactive waste production. It is believed that reduction of radioactive sources and improvement of the water chemistry quality should significantly reduce both the radiation exposure and radwaste production. The most important source of radioactivity is cobalt and replacement of cobalt containing alloy in the core region as well as in the entire primary system is considered the first priority to achieve the goal of low exposure and minimized waste production. A plant specific computerized cobalt transport model has been developed to evaluate various options in a BWR system under specific conditions. Reduction of iron input and maintaining low ionic impurities in the coolant have been identified as two major tasks for operators. Addition of depleted zinc is a proven technique to reduce Co-60 in reactor water and on out-of-core piping surfaces. The effect of HWC on Co-60 transport in the primary system will also be discussed.

INTRODUCTION

LWR water chemistry parameters are directly or indirectly related to the plant's operational performance and for a significant amount of Operation and Maintenance (O&M) costs. Obvious impacts are the operational costs associated with water treatment, monitoring and associated radwaste generation. Less obvious is the important role water chemistry plays in the magnitude of drywell shutdown dose rates, fuel corrosion performance and materials degradation. To improve the operational excellence of the BWR and to minimize the impact of water chemistry on O&M costs, General Electric has developed the concept of Optimum Water Chemistry (OWC).[1] The "best practices" and latest technology findings from the U.S., Asia and Europe are integrated into the suggested OWC Specification.

It is believed that reduction of radioactive sources and improvement of the water chemistry quality should significantly reduce both the radiation exposure and radwaste production. A number of known technologies and options are available to reactor operators, including cobalt source reduction, iron reduction, depleted zinc addition, control of ionic and organic impurities in reactor water, and decontamination, etc. A plant specific computerized cobalt transport model has been developed to evaluate various options in a BWR system under specific conditions. Some key parameters in OWC specification and the effect of hydrogen water chemistry (HWC) on radiation buildup control will be discussed in this paper.

OPTIMUM CHEMISTRY GOALS AND PROPOSED KEY CHEMISTRY PARAMETERS IN BWR COOLANT

The goals of optimum coolant chemistry in BWRs and proposed key chemistry parameters are given in Tables 1 and 2, respectively. Each of these goals and proposed limits has been demonstrated to be achievable in an operating BWR in Asia, Europe and the United States. However, no one reactor has yet achieved all of the optimum parameters, simultaneously. Major objectives of this paper are to discuss how the proposed chemistry parameters are related to optimum chemistry goals and to outline a strategy to meet those proposed chemistry limits.

Table 1. Optimum Water Chemistry Goals in BWRs

Parameter	Goals
IGSCC	No new crack initiation or growth (<0.01 in/yr)
Annual collective radiation exposure	<100 man-Rem/reactor
Annual radwaste volume	$<110\ m^3$
Fuel clad corrosion	No fuel failure due to water chemistry effect

Table 2. Proposed Key Chemistry Parameters in BWR Coolant

Parameter	Feedwater	Reactor Water
Iron	0.1 to 0.5 ppb	*
Cobalt	< 2.0 ppt	*
Copper	<50 ppt	<0.5 ppb
Nickel	<30 ppt	*
Sulfate	<50 ppt	<5 ppb
Chloride	<50 ppt	<5 ppb
Co-60	**	<2 Bq/g
Conductivity	**	<0.08 μS/cm
Electrochemical corrosion potential	**	Value that achieves goals of -No new IGSCC crack initiation -Minimum crack growth rate <0.01 in/yr

*Unspecified, controlled by feedwater limits
**Unspecified, controlled by reactor water limits

SHUTDOWN RADIATION FIELD BUILDUP CONTROL

The primary source of radiation field buildup on out-of-core surface is Co-60, with the exception of a few GEZIP plants where Zn-65 is also an important contributor to the recirculation piping radiation field. The activity transport process is a complex chemical reaction which can be affected by many water chemistry parameters. A semi-empirical phenomenological model has been developed to describe and calculate the corrosion product transport in the BWR primary system.[2] This model is presented in a block diagram shown in Figure 1. Model calculations are often very useful to estimate the relative contribution of each cobalt source in the system to the radiation field buildup. The effects of iron and other chemistry parameters can also be evaluated. It is well understood that radiation field buildup in many locations in the primary system may not occur by a similar mechanism nor at the same rate. In order to achieve the goal of reducing the radiation field to a very low level,

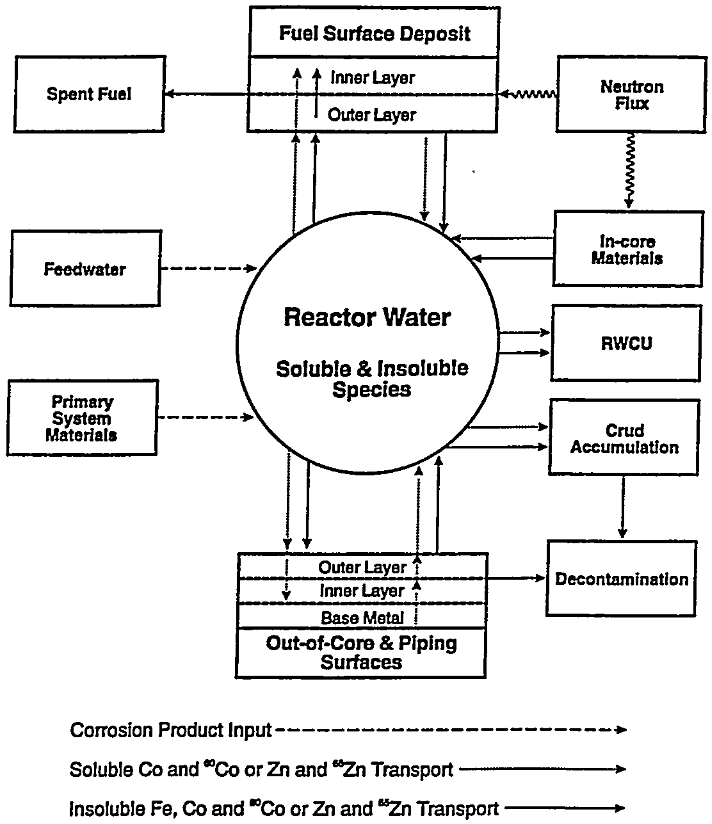

Figure 1. Schematic block diagram of corrosion product transport model

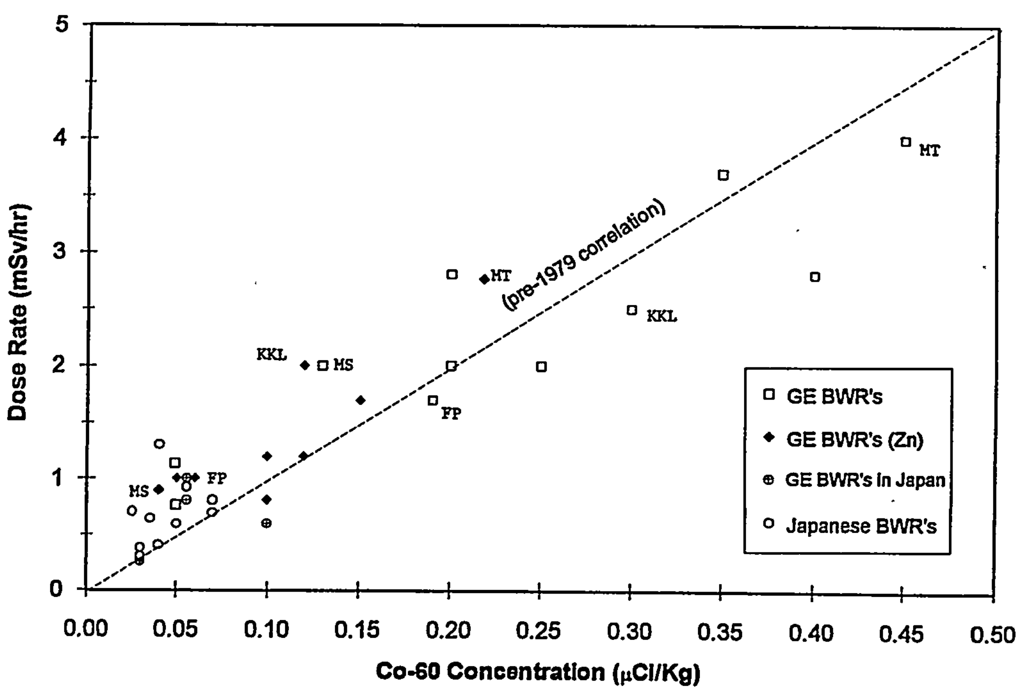

Figure 2. Correlation between soluble Co-60 concentration in reactor water and recirculation pipe dose rates

reduction of cobalt sources is the first priority, but other factors affecting the Co/Co-60 transport processes should be considered equally important. Some major factors are briefly described below:

Cobalt Source Removal

Figure 2 shows the relationship between the soluble Co-60 concentration in the reactor water and recirculation pipe dose rates. Note that the contact piping dose rate is linear with the soluble Co-60 concentration, equilibrium dose rates at approximately 100 mR/hr per 0.1 μCi/kg. Hence, one of the first priorities of reducing radiation field is to remove the sources of cobalt. There are several sources of cobalt including cobalt alloys and structure materials containing cobalt as impurities. All sources must be addressed, but the cobalt bearing materials in the core region should be given highest priority, because they are the most obvious contributors to the Co-60 in reactor water.

The benefit of replacing cobalt alloys or materials with non-cobalt or low cobalt materials may be very plant specific. A reliable cobalt transport model should be used to estimate the relative contribution of each component to the radiation field buildup. A cobalt replacement guideline has been published by Ocken.[3] One area easily overlooked are the feedwater heaters and steam dryer. If replaced, it is important that these large surface area components be made from material of controlled, low cobalt impurity concentration.

Control of Iron Input

In addition to producing Fe-55, Fe-59 and Mn-54 activities after neutron activation on fuel surfaces, iron plays an important role in Co/Co-60 transport and radiation field buildup in the primary system. A comprehensive review of the subject has been reported by Lin.[4] Iron acts as carrier of Co/Co-60 in reactor water; iron enhances Co and other transition metal ions (e.g. Zn, Ni, Cu) deposition on fuel surfaces, but it also enhances Co-60 or Zn-65 release from fuel surfaces when excessive iron is present on fuel surfaces. Excessive iron in water carrying the activities also create high radiation hot-spots in low flow regions in the primary system, including equipment drain lines, LPRM housing, vessel bottom, etc. Insufficient iron in fuel deposit will result in high soluble activities in reactor water and enhance the activity deposition on piping walls.

The optimum concentration of iron in feedwater should be controlled at ~0.5 ppb or lower, depending on the levels of transition metal ions in water. It has been hypothesized in GE cobalt transport model[5] that the iron crud (α-Fe_2O_3 is normally found in the fuel deposit) containing Co-60 may form a tight deposit by reacting with adequate transition metal ions which provide the "gluing" power for the bulky iron oxide deposit to form the stable mixed metal ferrites (spinel) in the deposit. In most U.S. domestic BWRs with relatively higher iron concentrations an increase of transition metal ions such as Zn^{+2} would certainly help reducing Co-60 release from the fuel deposit (see more below in Zn Addition). On the other hand, in most of the new Japanese plants the feedwater iron concentration is very low (<0.1 ppb). Under this condition NiO becomes the major component in the fuel deposit which is not stable and Co-58 and Co-60 are released easily from the fuel deposit. To minimize the cobalt activity release, Japanese have implemented a technique to inject the synthesized iron crud in the feedwater system to increase the Fe/Ni ratio up to approximately 5 (but the total Fe concentration is limited to 0.5 ppb) so that the Fe/Ni ratio in the fuel deposit is approaching 2, which is the stoichiometric ratio of Fe/Ni in the Fe and Ni mixed oxide in a spinel form, $NiFe_2O_4$.

Iron constitutes approximately 80% of the corrosion product oxides in the reactor coolant and fuel deposit. The majority of iron originates from corrosion of balance of plant carbon steel components in the steam/condensate and feedwater systems and is delivered to the reactor by the feedwater. To control the feedwater iron input, the first priority is to identify and eliminate the sources of iron. If the sources cannot be all eliminated, at least it is practical to identify key source terms and mitigate them by replacing the key components with corrosion resistant materials and/or coating the surfaces with corrosion resistant materials. The iron in the condensate upstream of the condensate treatment system should be effectively removed by improving the crud removal capabilities. For plants having deepbed demineralizers, addition of pre-filter is possible. Backwashable filters would generate minimum radwaste, while a greater majority of iron originates from corrosion of balance of plant carbon steel components in the steam/condensate and feedwater systems and is delivered to the reactor by the feedwater. To control the feedwater iron input, the first priority is to identify and eliminate the sources of iron. IF the sources can not be all eliminated, at least it is practical to identify key source terms and mitigate them by replacing the key components with corrosion resistant materials and/or coating the surfaces with corrosion resistant materials. The iron in the condensate upstream of the condensate treatment system should be effectively removed by improving the crud removal capabilities. For plants having deepbed demineralizers, addition of pre-filter is possible. Backwashable filters would generate minimum radwaste, while greatly extending the run length of the downstream deepbed demineralizer. Another alternative is to improve the crud removal efficiencies by using new types of resins (either in the deepbed demineralizer or the powdex filter system). Smaller bead sizes of resins and low cross-linked resins have been tested successfully in some plants.

Zinc Addition

Control of radiation field buildup in BWRs by zinc addition in the feedwater was first introduced by Marble[6] in 1986. It was hypothesized that soluble zinc inhibits the corrosion of stainless steel and thereby reduces the buildup of Co-60 on the piping surfaces. Currently, there are 13 plants operating worldwide utilizing zinc addition for shutdown radiation control.

Laboratory test results[7] confirmed that the Co-60 deposition rate is significantly lower in water containing 5-15 ppb of soluble zinc. The deposition rate was found even lower under HWC conditions with same levels of soluble zinc in water. Zinc ions appear to not only reduce the corrosion rate but also provide competition with Co-60 ions for the reaction sites on the corroding surface. With the overwhelming concentration ratios (Zn/Co $\geq$100), Co-60 is easily prevented from depositing on the stainless steel surfaces. In reactor experience, maintaining a constant level of zinc in reactor water has been proven to be an effective means to control the piping radiation field buildup on out-of-core piping. One important effect of zinc addition, which was not considered initially was the reduction of Co-60 concentrations in reactor water. A factor of 2-3 reduction in Co-60 has been observed in several reactors implementing GEZIP (GE Zinc Injection Process) (Figure 3). Some reactor data are also indicated in Figure 2.The reason for this effect has been discussed previously in the last section. This significant benefit of zinc addition is probably equally as important as the reduction of piping contamination in reactor operation.

After several years of GEZIP experience in operating BWRs, it has been observed that the Zn-65 activity, produced by the $^{64}Zn\ (n,\gamma)^{65}Zn$ reaction in natural zinc, can not be ignored. The benefits of zinc addition on Co-60 radiation buildup control are diminished by the presence of Zn-65 in some plants: much higher Zn-65 activity contribution to piping dose rates than expected (20-80%), particularly under HWC conditions; tramp Zn-65 found around the site in unwanted places; shutdown releases have increased the Zn-65 concentration in reactor water during shutdown cause higher than desired refueling floor dose rates; and the radwaste Curie content can be significantly increased. To eliminate these unwanted problems, it is recommended that Zn-64 depleted zinc oxide (DZO) replace the natural zinc in reactor applications. The Zn-64 content in depleted zinc is reduced from ~48.6% in natural zinc to ~1%. The quantity of DZO requirement in a reactor is very plant specific, mostly dependong on the iron concentration in the feedwater and reactor water. Therefore, reduction of iron input in a high crud plant should significantly reduce the cost of uzing DZO.

Reduction of Ionic Impurities

As described earlier, the main mechanism of Co-60 deposition is incorporation of soluble Co^{60} into the corroding stainless steel surfaces and the Co-60 deposition rate is known to be related to the stainless steel corrosion rate. Certain ionic impurities are known to enhance the corrosion rate of stainless steel and, therefore, increase the activity buildup rate. Laboratory experiments[7] have clearly demonstrated that when common laboratory chemicals like Na_2SO_4, H_2SO_4, NaOH were used as additives in water significantly higher Co^{60} buildup rates were observed (see Figure 4). Thus, minimizing the impurity input to maintain the reactor water conductivity at $\leq$0.08 µS/cm is essential to reduction of radiation field buildup. An exception would be the higher conductivity condition that accompanies zinc addition.

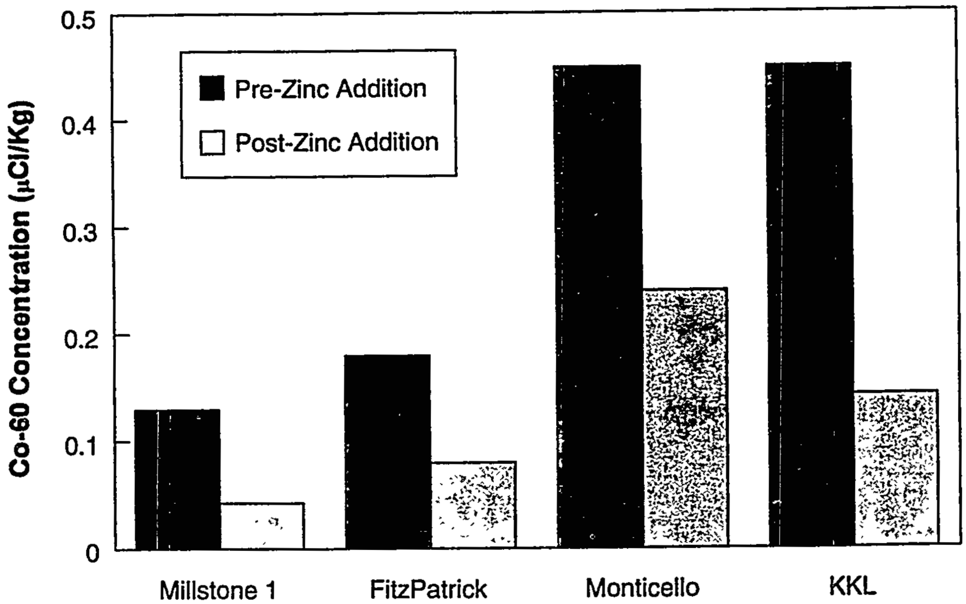

Figure 3. Fuel cycle average total Co-60 concentration in reactor water before and after implementation of zinc addition at four operating BWRs

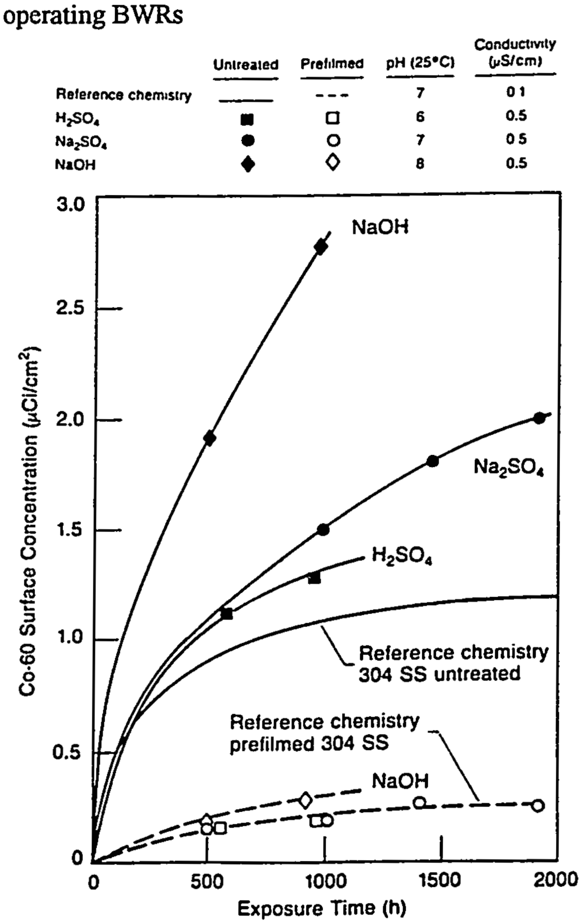

Figure 4. Effects of ionic impurities on Co-60 activity deposition on stainless steel test samples under NWC conditions

Role of Decontamination

The decontamination, if performed safely and efficiently, is probably the quickest way to reduce the radiation fields inside the drywell working area. However, the cost, the schedule, the radwaste produced in decontamination process, and the exposures associated with the performance of decontamination are significant enough to prohibit some operators from accepting it as a routine procedure. Furthermore, the surfaces after decontamination are generally corroding much faster without an established oxide film to protect from recontamination of Co-60 from reactor water. Consequently, the dose rates on decontaminated piping surfaces generally increase quickly within a cycle back to a pre-decontamination level. The decontamination operation may save some exposures immediately following the decontamination, but the plant may require repeated decontamination operations in every outage maintenance schedule. The decontamination vendors and plant operators should carefully consider some ways to minimize the recontamination problem. A strategy involving decontamination coupled with DZO injection is probably the cost effective way to radically limit subsequent contamination.

There are only a few chemical procedures which have been qualified for decontamination in BWR piping systems. The major concern is the attack of chemicals on the base metal of system materials. Some discussion on the issues of corrosion and the role of chemical decontamination in radiation control can be found elsewhere.[8] More recently, a feasibility study on full system decontamination has been performed, and the results of this study are reported in Ref. (9) and summarized in Ref. (10).

Effect of HWC

Laboratory test results[7] have shown that Co-60 deposition on stainless steel will probably be slightly enhanced by switching from NWC to HWC. The activity buildup rate is more profound under cyclic HWC/NWC conditions (see Figure 5).[11] In some U.S. reactors after switching from NWC to HWC, an increase in piping dose rate has been reported, but some plants including a few foreign plants have shown very minimal or no effect.[12] In some plants enhanced release of Co-60 activity has been observed, probably due to frequently changing HWC/NWC conditions. In the one plant which is adding higher levels of hydrogen for protection of internals, and which is also using GEZIP, an enhanced Zn-65 activity deposition on piping surfaces has also been obsrved. All these phenomena may be related to the result of oxidation/reduction processes, occurring in the oxide film as water chemistry environment is changed. While these effects are believed to be transient, they may persist for several cycles till corrosion films and solubilities stabilize under reducing conditions. A study is in progress to evaluate the HWC effect on radiation buildup through laboratory experiments and assessment of plant data.[11]

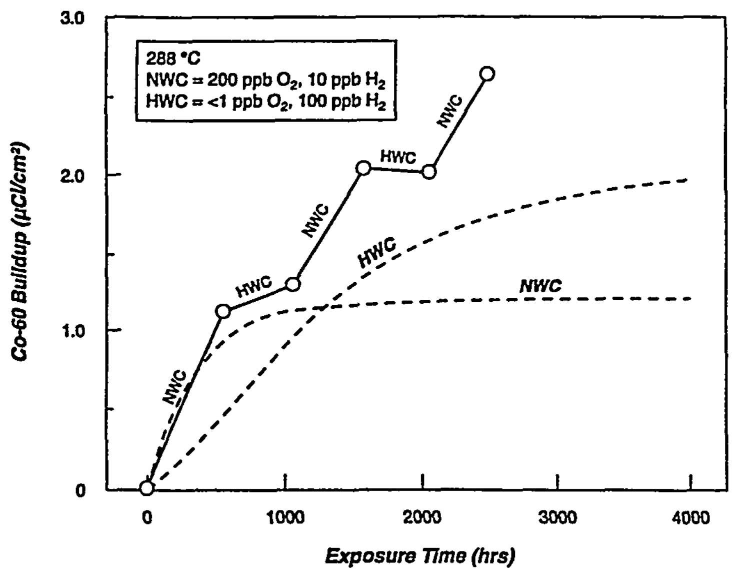

Figure 5. A comparison of Co-60 activity deposition on stainless steel samples under NWC, HWC and cycling between NWC/HWC

SUMMARY AND CONCLUSION

The concept of optimum water chemistry can be realized in radiation field reduction. Among the key chemistry parameters, cobalt and other metallic and ionic impurities should be minimized. The feedwater iron input should be controlled at 0.1 to 0.5 ppb to ensure a lower activity release rate from the fuel surface deposit. Addition of zinc in reactor water would also decrease the Co-60 concentration in reactor and activity buildup on out-of-core surfaces.

Cobalt/Co-60 activity buildup model calculations are essential to define effective approaches to control and reduce radiation field buildup.

Effects of HWC on radiation field buildup have been observed, but the magnitude may be minimized with source term reduction and proper operation procedures.

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Author Biography

Chien C. Lin is a Technical Leader and Radiation Reduction Project Manager in GE Nuclear Energy, where he has been primarily responsible for research and development in the areas of reactor coolant chemistry and radiological technology since 1971. Dr. Lin's current research interests includes radiation and radiochemistry in water and the radiation exposure control technology in BWRs. He has a B.S. in Chemical Engineering from Tunghai University in Taiwan and a Ph.D. in Chemistry from the University of New Mexico. He did his post-doctoral research in nuclear chemistry at Washington University in St. Louis before joining GE at Vallecitos Nuclear Center in 1971.

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Background

- Controlling radiation field buildup to achieve the ALARA exposure goals is an important task for nuclear power plant operators
- Plant specific strategies aimed at controlling radiation field buildup and minimizing personnel exposure have to be developed and implemented
- Reduction in radiation field by water chemistry control is a formidable challenge

Optimum BWR Chemistry Goals

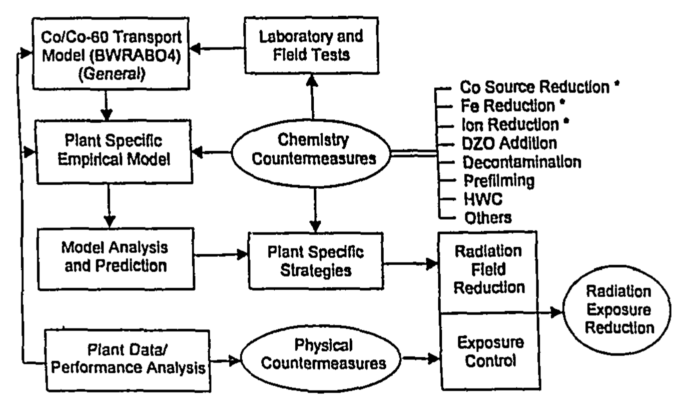
- No new IGSCC initiation or growth
- Yearly exposure <100 manRem
  Reduction in general area dose rates to:
  - Drywell <10 mR/hr
  - Undervessel <10 mR/hr
  - Refueling Floor < 2 mR/hr
- Reduction in annual radwate volume <100 m^3
- Elimination of fuel clad corrosion

Proposed Key Chemistry Parameters in BWR Coolant

Parameter	Feedwater	Reactor Water
Iron	0.1 to 0.5 ppb	*
Cobalt	<2.0 ppt	*
Copper	<50 ppt	<0.5 ppb
Nickel	<30 ppt	*
Sulfate	<50 ppt	<5 ppb
Chloride	<50 ppt	<5 ppb
Co-60	**	<2 Bq/g (or ~0.05 μCi/Kg)
Conductivity	**	<0.08 μS/cm (excluding additives)
Electrochemical corrosion potential	**	Value that achieves goals of - No new IGSCC crack initiation - Minimum crack growth rate (<0.01 in/yr)

*Unspecified, controlled by feedwater limits; **Unspecified, controlled by reactor water limits

BWR Radiation Field and Exposure Reduction Project



Countermeasures for Radiation Field Buildup

- Co source reduction* - reducing Co-60 source
- Feedwater Fe reduction/control* - controlling Co-60 transport
- Ionic impurity reduction* - reducing material corrosion
- DZO addition* - reducing Co-60 in water and on piping
- Decontamination - removing radioactivities on piping
- Prefilming - reducing initial activity buildup
- HWC* - reducing material corrosion
- Others

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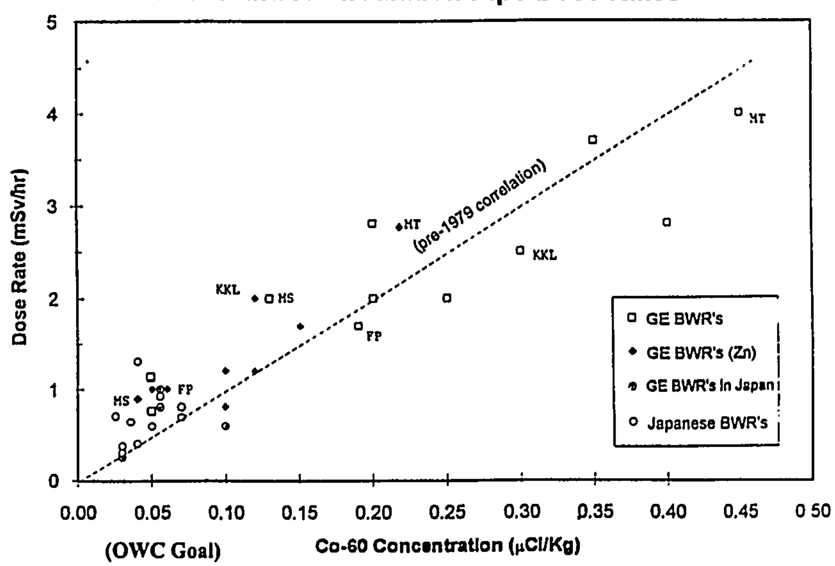
*Optimum Water Chemistry Parameters

Relationship between Recirc. Piping Radiation Field and Soluble Co-60 Concentration in Reactor Water

- Laboratory tests results clearly demonstrate that Co or Co-60 deposition on stainless steel surfaces is proportional to Co or Co-60 concentratin in water
- Reactor data show the equilibrium contact piping dose rate is proprotional to soluble Co-60 concentration in reactor water in mature plants at approximately 100 mR/hr to 0.1 µCi/Kg

Correlation Between Soluble Co-60 Concentration in Reactor Water and Recirculation Pipe Dose Rates



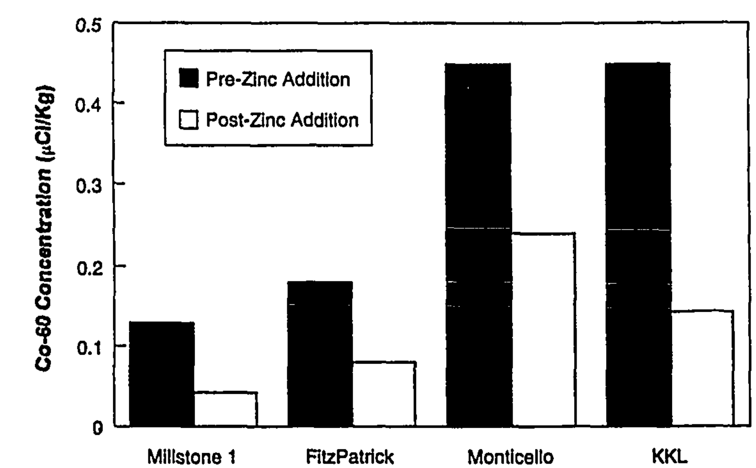
Cobalt-60 Source Term Reduction

- Remove in-core Co-60 sources
  - Control blade pins and rollers
  - Cobalt control of in-core materials
- Systematic identification and removal of other plant cobalt sources
- See EPRI Cobalt Reduction Guidelines Rev. 1 EPRI TR-103296 (Dec. 1993)
- Control of Co-60 release from fuel deposit
  - Feedwater iron control
  - DZO injection

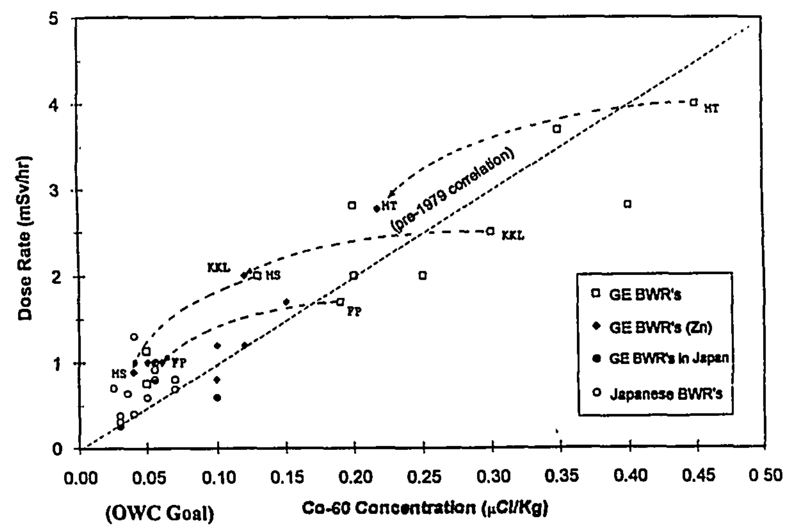
Effects of Zn Addition in Radiation Buildup Control

- Zn (depleted in Zn-64) is recommended to avoid Zn-65 production
- Laboratory data confirm that Zn at 5-10 ppb in water slows down the corrosion rate and reduces Co-60 deposition on stainless steel surface
- Reactor data show Zn also effectively reduces the Co-60 release rate from fuel deposit resulting in lower Co-60 concentration in reactor water

Effect of Zn Addition on Co-60 Concentration in Reactor Water



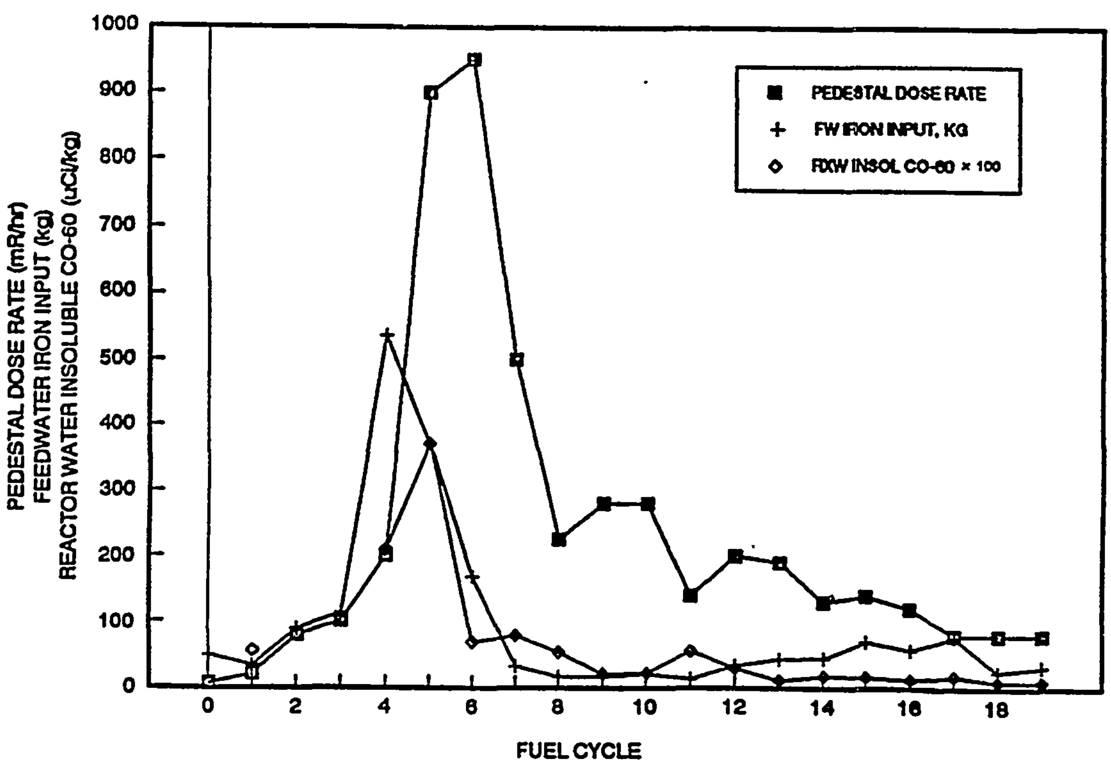
Correlation Between Soluble Co-60 Concentration in Reactor Water and Recirculation Pipe Dose Rates



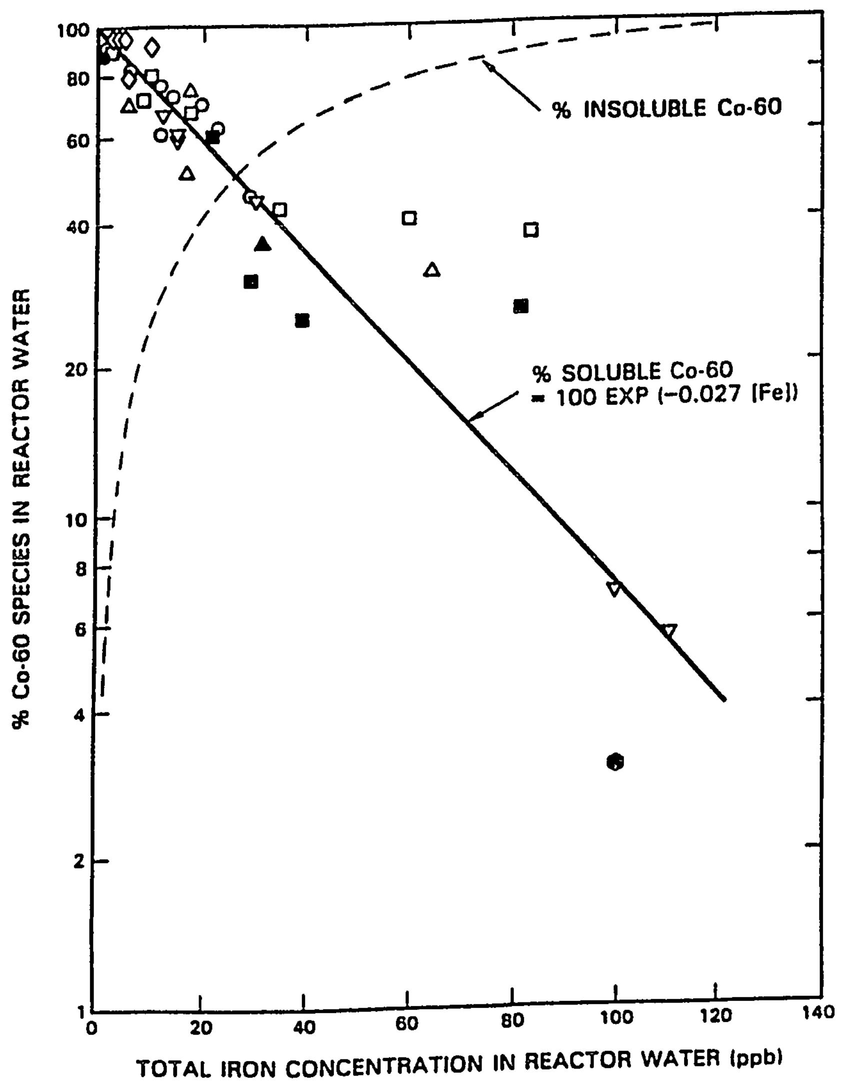
Iron Crud in BWR Coolant

- Plays an important role in Cobalt Transport and Radiation Field Buildup
  - As carrier of Co and Co-60 in water
  - Enhances Co (or Zn) deposition on fuel surfaces
  - Enhances Co-60 release from fuel surfaces when excessive Fe is present on fuel surfaces
  - Creates high radiation hot-spots in low flow regions in the primary systems
- Produces Fe-55, Fe-59, and Mn-54 activities after neutron activation on fuel surfaces
- Increases Radwaste Production
- See EPRI NP-6942 "Foreign Approaches to Controlling Radiation Field Buildup in BWRs"

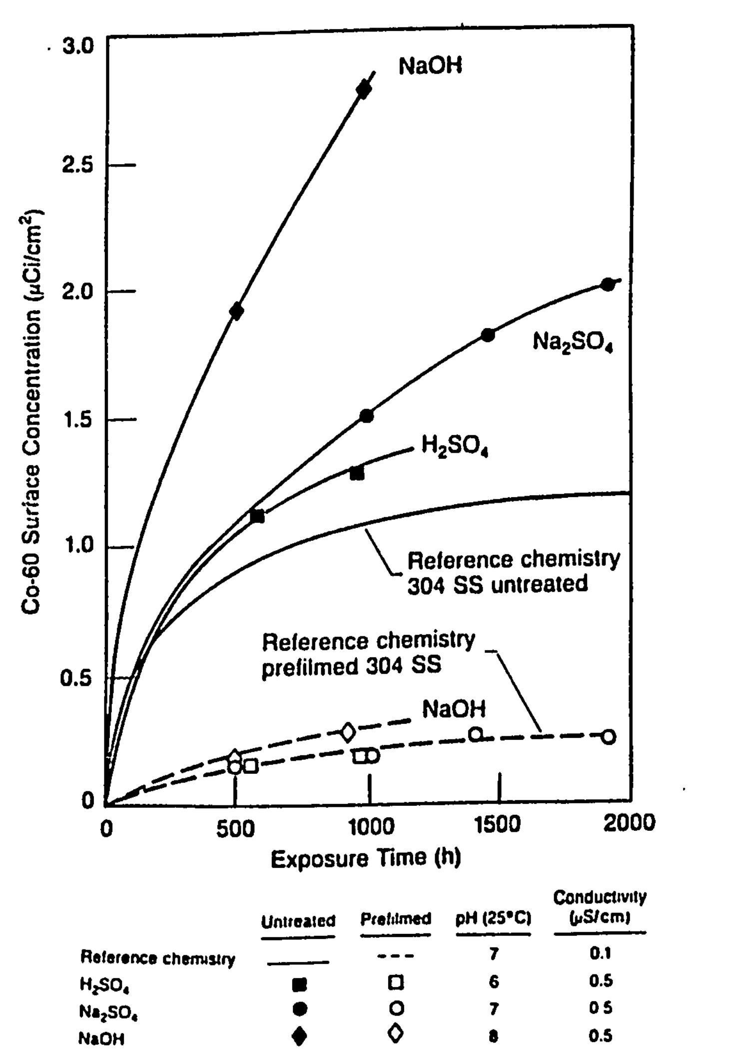
COMPARISON OF PEDESTAL RADIATION FIELDS, FEEDWATER IRON INPUT AND INSOLUBLE Co-60 CONCENTRATION IN REACTOR WATER IN A JAPANESE BWR



Effect of Iron Concentration on Co-60 Solubility in Reactor Water

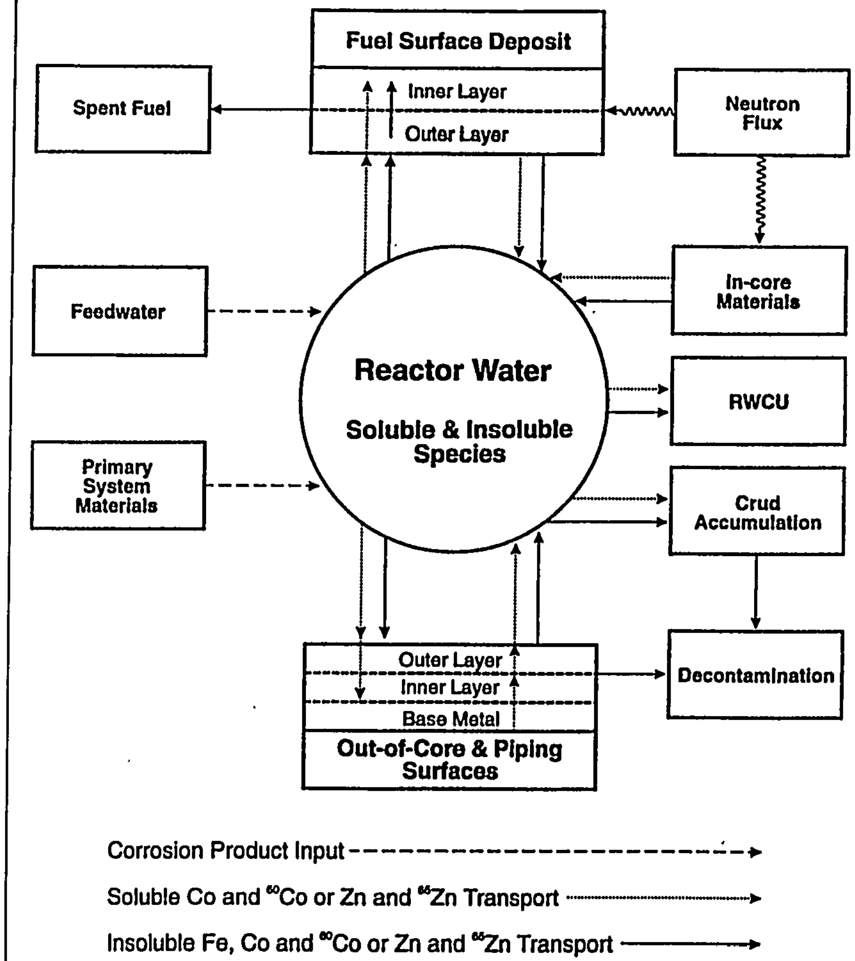


Effect of Water Chemistry on Co-60 Deposition on Stainless Steel Surfaces



Objectives of Cobalt/Co-60 Transport Model Calculation

- To define effective approaches to control and reduce radiation field buildup
- To predict the consequences of water chemistry changes and material replacement in the primary system
- To assist in cost/benefit evaluation for countermeasures under plant specific conditions



Block Diagram of Corrosion Product Transport Model

Co/Co-60 Transport Model

Input Data

Feedwater Fe, Co concentrations
Reactor water conductivity
Cobalt release rate from in-core materials
Cobalt release rate from primary system surfaces

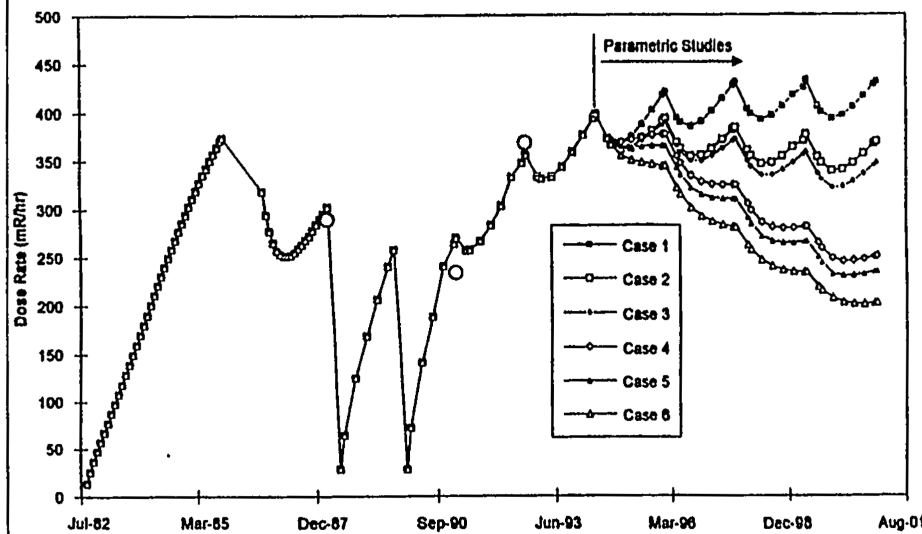
Output Data

Fuel deposit: Fe, Co, Co-60 (inner/outer layer)
Reactor water: Fe, Co, Co-60 (soluble/insoluble)
Out-of-core surface: Fe, Co, Co-60 (inner/outer layer)
Recirc. piping surface: Fe, Co, Co-60 (inner/outer layer), dose rate
RWCU: total Fe, Co, Co-60, insolbule Co-60

Example:

Actual plant data for 5 cycles and parametric studies after 6th cycle

BWRAB02 MODEL CALCULATIONS
Piping Dose Rates



BWRAB02 MODEL CALCULATIONS --- INPUT DATA FOR PARAMETRIC STUDIES IN CYCLES 7-10 *

	FW Fe, ppb	FW Co, ppt	IN-CORE Co gm/day	Rx SYSTEM Co gm/day	Rx WATER COND., uS/cm
CASE 1	3.6	8	0.05	0.17	0.15
CASE 2	3.6	8	0	0.17	0.15
CASE 3	3.6	8	0	0	0.15
CASE 4	3.6	3	0	0	0.15
CASE 5	3.6	3	0	0	0.08
CASE 6	0.5	3	0	0	0.08

* Actual data are used in cycles 1-6 to reproduce the reactor data

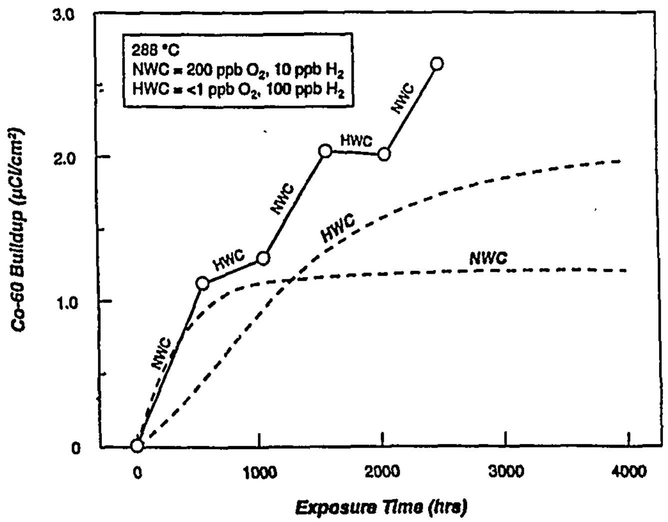
Effects of HWC on Radiation Field Buildup - Summary of Observations

- Shutdown dose rate increase due to increased Co-60 deposition in some reactors, but overall annual personnel exposures in most HWC plants are not significantly affected (one exception)
- Dose rate buildup varies among plants
  - No effect for some plants
  - Substantial effect for other plants
- Soluble and filterable Co-60 concentrations in reactor water vary among plants
- At some plants with both HWC and GEZIP, shutdown dose rate increase was dominated by Zn-65
  - Some hot-spots observed
  - Decontamination was not uniformly successful

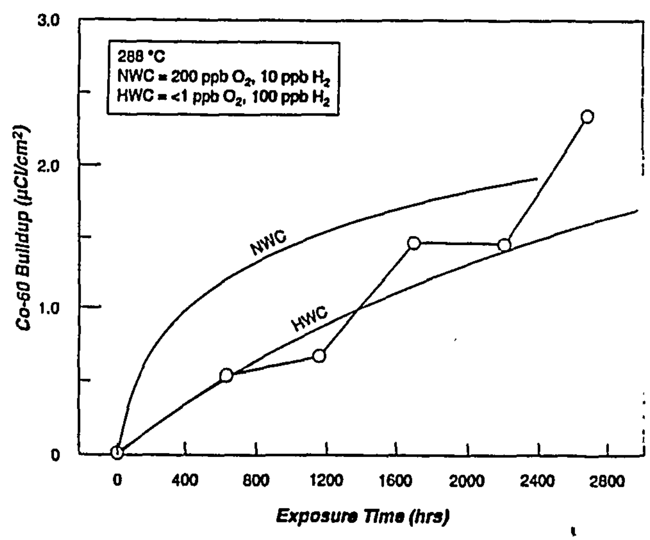
Recommendations for Minimizing the Impact of Switching from NWC to HWC on Radiation Field Buildup

- Maintain steady power operation
- Avoid NWC/HWC alternation. If reduction of H_2 injection rate is necessary, return to normal H_2 injection rate as soon as possible.
- In a GEZIP plant, develop a strategy to quickly change out natural Zn with DZO to eliminate any problems created by Zn-65
- To achieve two fundamental goals of optimum water chemistry is essential:
  - To implement and accelerate a Co source reduction program so that Co-60 is quickly reduced to <0.05 µCi/Kg in reactor water.
  - To reduce feedwater Fe concentration $\leq$0.5 ppb

Comparison of Co-60 Deposition on 304SS Samples under NWC, HWC, and NWC/HWC Cycling Conditions



Co-60 Deposition on 316 SS



Summary and Conclusion

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

Radiation Exposure Reduction

ALARA:	**As Low As Reasonably Achievable**
ALAFA:	**As Low As Financially Affordable**
ALATA: (OWC Goal)	**As Low As Technically Achievable**

1-3

EFFORTS TO REDUCE EXPOSURE AT JAPANESE PWRs: CVCS IMPROVEMENT

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ABSTRACT

Many reports have been focused on the reduction of radiation sources and related occupational exposures. The radiation sources mainly consist of corrosion products. Radiation dose rate is determined by the amount of the activated corrosion products on the surface of the primary loop components of Pressurized Water Reactor (PWR) plants. Therefore, reducing the amount of the corrosion product will contribute to the reduction of occupational exposures. In order to reduce the corrosion products, Chemical and Volume Control System (CVCS) has been improved in Japanese PWRs as follows :

a. Cation Bed Demineralizer Flowrate Control
b. Hydrogen Peroxide Injection System
c. Purification Flowrate During Plant Shutdown
d. Fine Mesh Filters Upstream of Mixed Bed Demineralizers

INTRODUCTION

In most nuclear power plants, annual inspection has been a major contributor to the occupational exposures. If operating plants have extra works such as maintenance without scheduled shutdown, their extra occupational exposures tend to increase. Mitsubishi and Japanese PWR utilities have been successful in reducing the occupational exposures, despite the numerous inspections of the PWR primary loop pipes and components. Figure 1 shows a trend of the collective exposures at the Japanese PWRs for the past two decades.

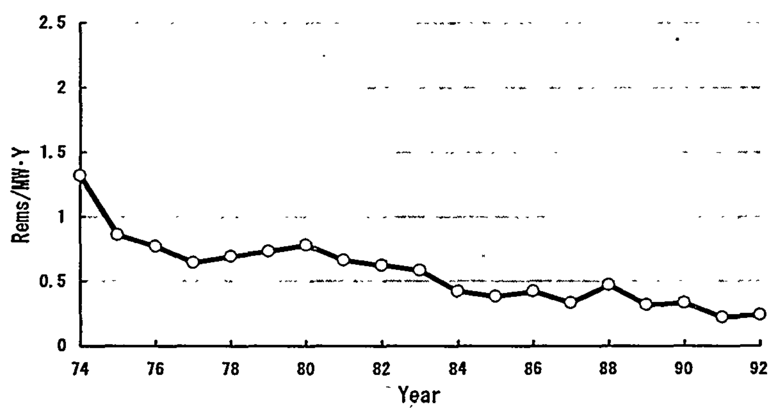


Figure 1. Radiation exposure : Japanese PWR plants

In 1974, 1.3 person–rem was incurred for each MW–year of electric power generated; in 1992, 0.24 person–rem was incurred. Then, there has been a fivefold reduction in person–rem per MW–year.
Because of the plan to replace steam generators (SG) in the near future, the need for extra works will increase. Therefore, the Japanese PWR utilities have intensive requirements to reduce both the collective and individual doses. In the early 1980s, Mitsubishi worked out a strategy for the exposure reduction and began to study the way how to reduce the exposures. The occupational exposures are composed of the time spent in the radiation field and the radiation level. The former depends on the amount, the difficulty, and the efficiency of work to be done. The latter is affected by the radiation sources, namely the amount of radioactive nuclides, especially cobalt, to exist on the surface of the primary loop components. The key elements associated with controlling the radiation source are summarized in figure 2. In the current paper, Mitsubishi introduces the outline of our works in this field.

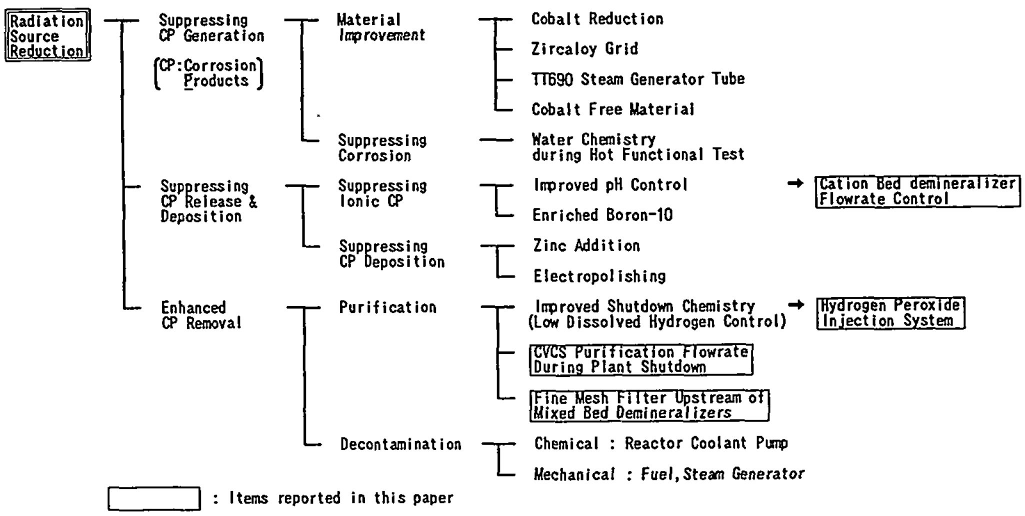


Figure 2. Key elements to control radiation sources

CVCS IMPROVEMENT

General System Feature

Figure 3 shows a simplified flow diagram of the CVCS. This shows a typical arrangement for a conventional 4–loop plant but all plants are basically similar. This system, which is usually referred to as the "CVCS" system, is one of the most important parts of PWR. It performs several functions when the plant is operating. Some of the functions are to purify the primary coolant water continuously and to adjust the primary water chemistry. Filters and demineralizers are provided for these purposes.

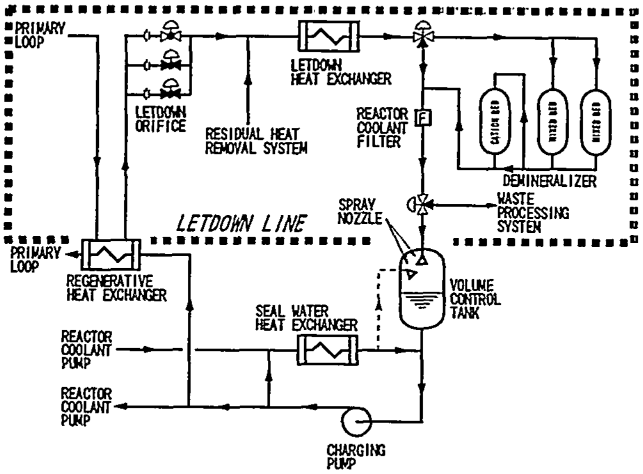


Figure 3. CVCS flow diagram

Figure 3 shows also what is called the letdown path. During a normal operation, water comes from one of the primary loops at a pressure of approximately 157 kg/cm^2 and is first cooled by the Regenerative Heat Exchanger. The letdown flow passes one of Letdown Orifices which reduces the pressure. The flow is then cooled further by the Letdown Heat Exchanger. The flow is cool enough to pass through the demineralizers for the purification downstream of the Letdown Heat Exchanger. The demineralizers remove ionic fission products and corrosion products. Filters are provided to ensure filtration of particles.
A further purification feature is provided for use during a shutdown operation. It is called the Low Pressure Letdown path. The Low Pressure Letdown flow bypasses the Regenerative Heat Exchanger and the Letdown Orifices, and passes into the letdown line upstream of the Letdown Heat Exchanger from Residual Heat Removal System.

System Improvements

Cation Bed Demineralizer Flowrate Control

The chemical control reagent employed for pH control is Lithium Hydroxide (LiOH) in PWRs. Li–7 is produced in the reactor core region due to irradiation of dissolved boron. The Li concentration is maintained within a certain control band. If the Li concentration exceed the control band, the Cation Bed Demineralizer is employed in the letdown in series operation with the Mixed Bed Demineralizer. Optimum control of the water chemistry is recognized to suppress the transport and activation of corrosion products. Mitsubishi has thought that the pH of 7.3 (@ 285° C) is appropriate to reduce the radiation source, which is based on our experiments and foreign information. Maximum Li is still 2.2 ppm, because we do not have enough data of the effect on Inconel yet. Mitsubishi has proposed to the Japanese PWR utilities that the Li is held at 1.8 to 2.2 ppm until a pH of 7.3 ± 0.1 (@ 285° C) is reached and then the Li is controlled to maintain the pH of 7.3 ± 0.1 (@ 285° C) until the end of the core cycle. This Li band is shown in figure 4.

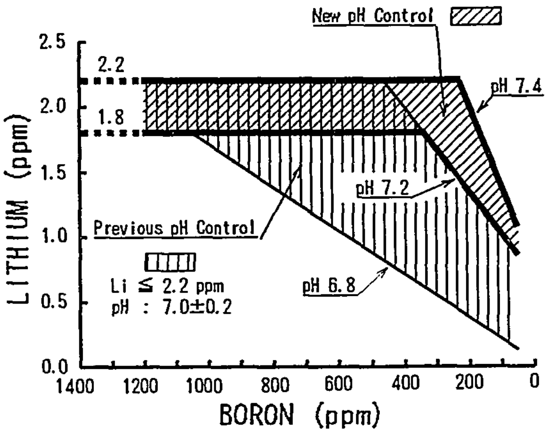


Figure 4. Lithium control band proposed by Mitsubishi

The Cation Bed Demineralizer flowrate is usually equivalent to the normal letdown flowrate. When it is necessary to remove some excess Li, the letdown flow passes through the demineralizer. In most operating plants, the operator numerously has to divert the flow path around the demineralizer, especially in the beginning of the core cycle (Figure 5).

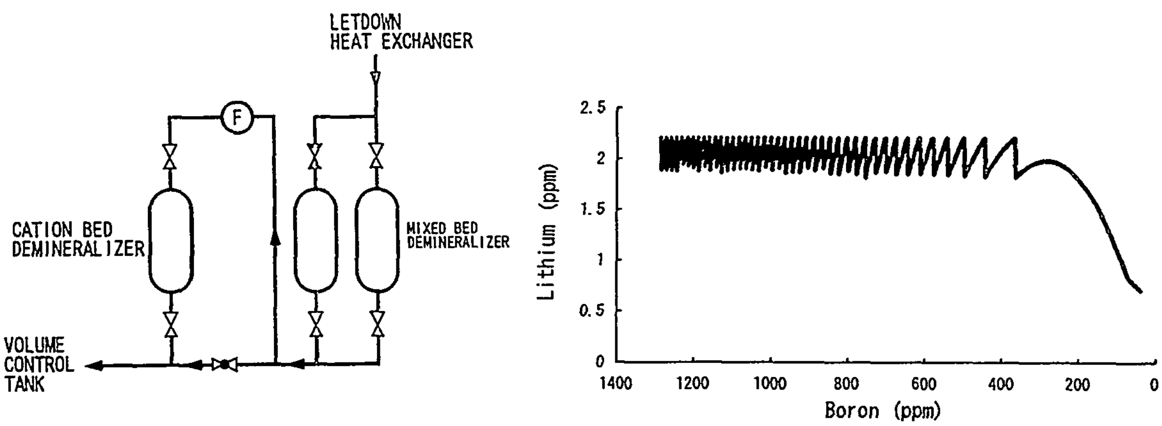


Figure 5. Manual control system and Lithium variation

Such numerous diversions would load on the operator. Therefore, the Japanese PWR utilities have preferred an automatic Cation Bed Demineralizer flowrate control system to lighten the operator's work. Mitsubishi has improved the cation bed demineralizer flowrate control system to control a low flowrate about 0.5 m^3/hr (nearly 2 gpm) continuously and remotely. Figure 6 shows the above improved system that will decrease significantly the frequency of the flow path diversion.

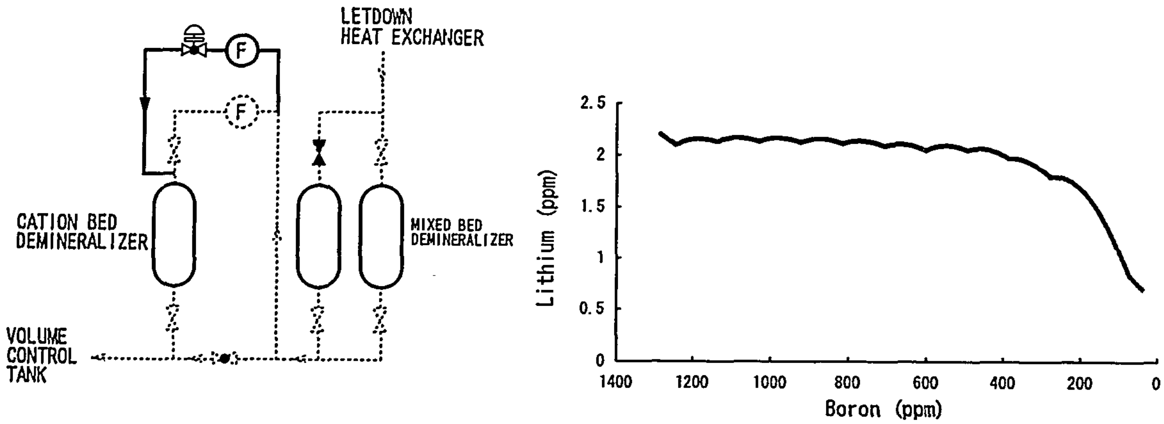


Figure 6. Continuous & remote control system and Lithium variation

Figure 7 presents our future plan of a full-automatic control system with a Boron meter and a Li meter, which would be controlled by a computer.

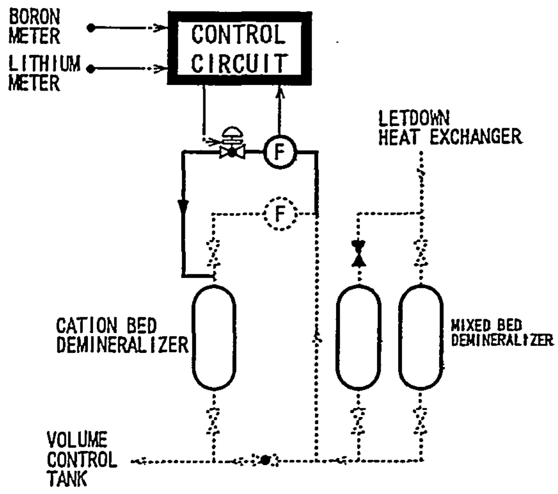
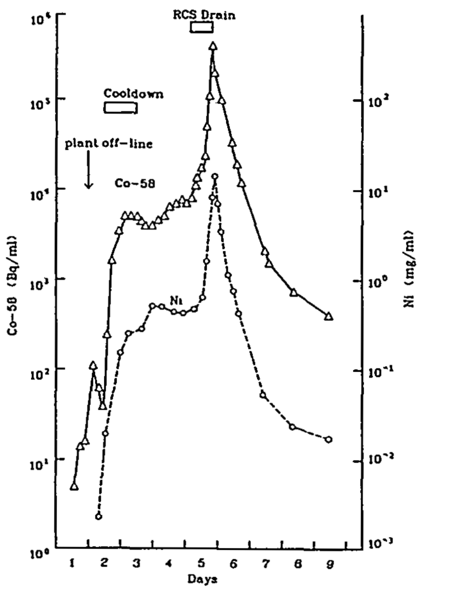


Figure 7. Full-automatic control system

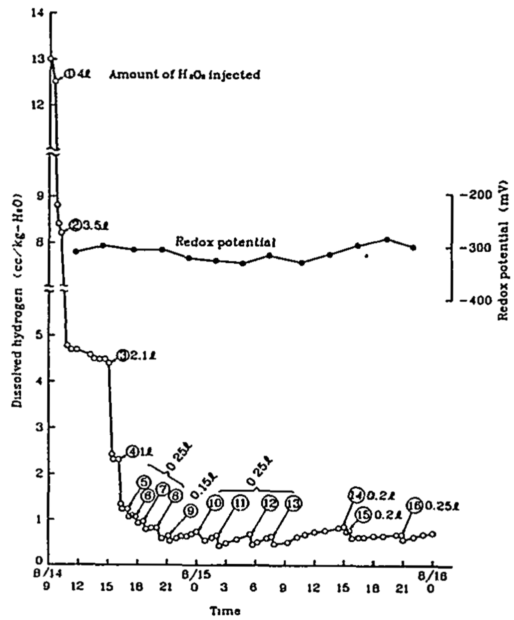
Hydrogen Peroxide Injection System

It is well-known that the soluble corrosion products reach an extremely high level after a plant hot shutdown, because the temperature and the chemical condition of the primary coolant vary to a great extent. Figure 8 shows an example of cobalt and nickel concentration during the shutdown operation. The concentration was observed to be 1000 or 10000 times higher than that during the power operation.



Co-58 and Ni concentration changes during the plant shutdown process (Genkai nuclear plant unit 2 6th annual inspection)

Figure 8. Co and Ni concentration during plant shutdown[1]



Changes of dissolved hydrogen concentration and redox potential by hydrogen peroxide injections (30% H_2O_2)

Figure 9. Dissolved hydrogen concentration with Hydrogen Peroxide addition[2]

Thus, the removal of the corrosion products during the shutdown operation is effective in reducing the radiation sources. In order to promote the corrosion product removal, Mitsubishi studied "Low Dissolved Hydrogen Control" with the Japanese PWR utilities. A low dissolved hydrogen concentration of approximately 0.5 cc/kg-H_2O was obtained to be optimum during the shutdown operation, promoting the corrosion product removal. Hydrogen Peroxide (H_2O_2) is injected to the primary coolant circuit after the cooldown operation to accomplish the chemistry condition.

The chemical reagent can be added by pouring them into the Chemical Mixing Tank which is connected with the Charging Pump suction line. Figure 9 shows a result of the dissolved hydrogen concentration with the Hydrogen Peroxide addition in an operating plant. The dissolved hydrogen was revealed to decrease rapidly. However, a lot of Hydrogen Peroxide addition was necessary to reach and maintain the low dissolved hydrogen.

Since the study, Mitsubishi developed the "Hydrogen Peroxide Injection System" to add the hydrogen peroxide semi-automatically (Figure 10). This system is composed of one tank, one positive displacement pump and several valves. The Hydrogen Peroxide is poured to the tank in advance. As a operator starts the system, an adequate quantity of the Hydrogen Peroxide is injected to the Charging Pump suction line. Then, an equipped timer terminates the injection. The flowrate is controlled by the pump stroke and/or the pump rotation.

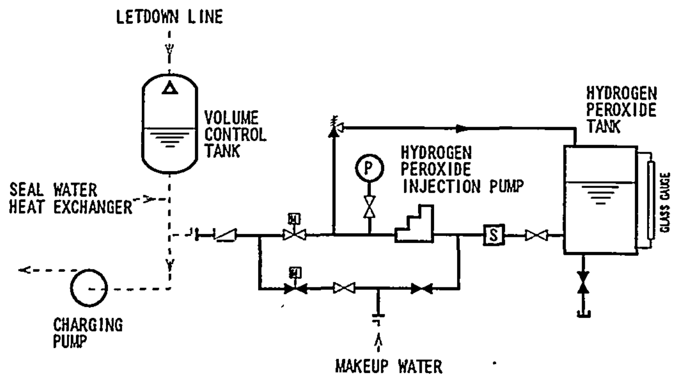


Figure 10. Hydrogen Peroxide injection system

Purification Flowrate During Plant Shutdown

The CVCS is used to purify the primary coolant water continuously and demineralizers are provided for this purpose. During the shutdown, concentration of cobalt and nickel is observed to be 1000 or 10000 times higher than that during the power operation (Figure 8). Therefore, the corrosion products can be removed effectively during the plant shutdown. With increasing the purification flowrate, the corrosion products would be removed more effectively. The CVCS purification flowrate is applied to two periods, such as the power operation and the shutdown operation. Increasing the flowrate during the power operation would increase the plant construction cost and on the other hand, would increase the plant heat loss. However, increasing the flowrate during the shutdown does not cause much increase in the plant construction cost nor the increase in the plant heat loss. Mitsubishi assessed the purification flowrate during the shutdown at approximately 60 m^3/hr (260 gpm) for conventional 4-loop PWR plants. If the purification flowrate during the shutdown would be increased for aging plants, the following improvements are required (Figure 11).

- Shift of the piping diameter to larger size, in the Low Pressure Letdown line.
- Increasing the capacity of filters, both upstream and downstream of the Mixed Bed Demineralizers.
- Additional installation of piping and valve, in order to use the Auxiliary Spray Nozzle of the Volume Control Tank together with the normal nozzle.

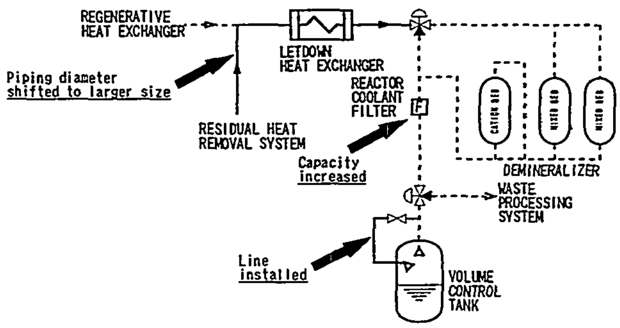


Figure 11. Improvement for increasing purification flowrate during shutdown

Fine Mesh Filters Upstream of Mixed Bed Demineralizers

In the letdown line of the conventional plant, only the Reactor Coolant Filter is provided downstream of the Mixed Bed Demineralizers, in order to collect the resin fines and particulate matter larger than 25 micron. The particles in the fluid would accumulate in the Mixed Bed Demineralizers. The accumulated particles would return to the primary circuit as released from the demineralizers. Diverting the demineralizer's effluent to the waste processing system would prevent the particles from returning to the primary circuit. This operation would load on the operator. By installing a fine mesh filter upstream of the Mixed Bed Demineralizers, the particulate corrosion products could be removed and the surplus operation should not be necessary. The diameter of the filter porosity is lower than one micron. The fine mesh filter would result in the radiation source reduction.

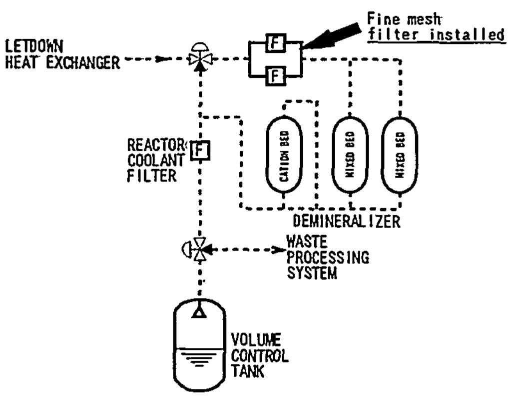


Figure 12. Installation of fine mesh filters

CONCLUSIONS

This paper introduces four CVCS improvements in Japanese PWRs. Mitsubishi has endeavored to reduce the occupational exposure by improving the system design and water chemistry. However, the efforts for further reduction of the exposure are still continuing.

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Author Biography

Ryosuke Terada is a Senior Engineer with the Water Reactor Division of Mitsubishi Atomic Power Industries, Inc. in Japan. Mr. Terada has worked on designing the Nuclear Steam Supply System of commercial Japanese PWRs, that involved "Chemical and Volume Control System," "Emergency Core Cooling System," "Residual Heat Removal System," and other safety-related systems. Recently, he became a member of the Dose Reduction Program in Mitsubishi Group. Previously, he was engaged in the start-up test and operation at Takahama unit 4 of Kansai Electric Power Company, Inc., Japan. He holds an M.A. Sci. in Nuclear Engineering from Osaka University in Japan.

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PAPER 1-3
DISCUSSION

Khan: I have two questions. The first question is about your automatic pH control system. Is it exclusively designed for Mitsubishi-designed reactors, or can it also be installed in other types of PWRs?

Terada: No, it isn't. We can apply it to any plants. This automatic system is for operating plants have decided to install and we have another option. It is manual needle valve system to control the low flow rate. So in this case about ten or thirteen plants have decided to install.

Khan: My other question relates to when you increase the purification flow during shutdown. Typically, I think, in PWRs you take a few percent. By how much do you increase the purification flow?

Terada: We increase the purification flow during shutdown up to 60 m^3/h for the conventional plants.

Helman: Did you backfit these or are these all a part of new construction? Did you add these systems to existing plants, or are these new construction ideas?

Terada: Yes, we did. The system has been added to existing plants.

Helman: I know that you added a couple of new fine filters, and usually when you add filters to systems you also increase your rad waste and increase your maintenance exposure to change those out. Did you see any of that?

Terada: Yes. Waste disposal will increase, but the fine mesh filter is mainly for preventing accumulated matter from returning to the primary circuits. Additionally the filter is effective to radiation exposure reduction.

Khan: I just want to add the point about the filters. Several years ago John Baum visited the Obrigheim Nuclear Power Plant in Germany and they strongly recommended the use of these ultra-fine pore filters. They suggested that you gradually transition to them, using intermediate pore sizes. In that case, your rad waste problems and exposures are much reduced. You lower the particulate activity gradually, and eventually you end up with the very fine mesh filters that you have been talking about.

EFFORTS TO CONTROL RADIATION BUILD-UP IN RINGHALS

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ABSTRACT

It is well known that good control of the primary chemistry in a PWR is essential in order to minimise material problems and fuel damages. It has also been well established that the water chemistry has a great influence on accumulation of corrosion products on the fuel and the radiation build-up on primary system surfaces. Ringhals was one of the pioneers to increase operating pH in order to reduce radiation build-up and has now been operating for ten years with pH at 7.4 or (in later years) 7.2. Our experience is favourable and includes low radiation levels in the new (1989) steam generators of Ringhals 2. Ringhals 4 has operated almost its whole life at pH 7.2 or higher and it remains one of the cleanest PWRs of its vintage.

In addition to strict adherence to a stable operating chemistry, Ringhals is now working on a program with the aim to find optimum shut-down and start-up chemistry to reduce activity levels in the primary systems. A particular goal is to use the shut-down and start-up chemistry at the 1994 outage in Ringhals 3 in order to reduce doserates in preparation for the planned steam generator replacement in 1995.

The paper summarises the experience to date of the established operating chemistry, on-going tests with modified shut-down and start-up chemistry and other measures to limit or reduce the activity build-up.

INTRODUCTION

There are four nuclear sites in Sweden, and their 12 units provide about 50 % of the electricity in the country. The rest is mainly produced in hydroelectric plants. Of the nuclear plants, 9 are BWRs designed by ABB ATOM and 3 are PWRs designed by Westinghouse. The three PWRs are all located at the Ringhals site.

Ringhals is a four-unit site 60 km south of Gothenburg on the Swedish west coast. The site is owned and operated by Vattenfall.

Ringhals 1 is a 795 MWe BWR which has been in commercial operation since 1976. The plant was uprated from 750 to 795 MWe in 1989.

Ringhals 2 is an 875 MWe PWR which has been in commercial operation since 1975. The steam generators were replaced in 1989 and the plant was uprated from 800 to 875 MWe in 1990.

Ringhals 3 and 4 are identical PWRs with an installed capacity of 915 MWe each. Ringhals 3 started commercial operation in 1981 and Ringhals 4 in 1983. The steam generators of Ringhals 3 will be replaced in 1995 and at the same time modifications will be introduced to prepare for an increase in the capacity with 8 %.

ACTIVITY BUILD-UP AND MODIFICATIONS OF THE PWR PRIMARY CHEMISTRY

Chemistry During Operation

Activity build-up in Ringhals 2 was considered fairly normal during the first years after start up in 1975. The steam generator doserates, which are normally used to compare PWR activity levels, stabilised at 70 mSv/h (7 R/h) already after two years (1,2 EFPY) as can be seen in figure 1. Until 1979 the levels were about the same but we then saw a rapid increase in the channel head doserates. Another observation was very heavy crud layers on the fuel.

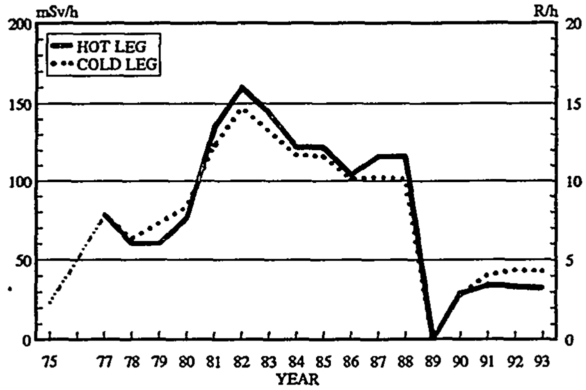

Figure 1. Average Doserates in the Steam Generator Channel Heads of Ringhals 2

Due to a considerable amount of work inside the steam generator channel heads we were quite concerned with the increasing doserates. Although we had been operating the plant within the Westinghouse chemistry specifications it was suggested that the causes of the problem were low Li and H_2 - concentrations. An increase in the minimum Li-concentration from 0.2 to 0.7 ppm was therefore introduced from the start-up after the 1979 outage and it was later verified that the rapid crud build-up on the fuel was halted.
Based on Westinghouse recommendations we made the next change, introduction of the original co-ordinated chemistry with a minimum Li-concentration of 0.7 ppm (the "dog leg" curve), after the 1980 outage. We now know that this chemistry will cause a transport of corrosion products from the system to the core, where it is activated, during the first part of the cycle. At low boron concentrations, and especially during coast down, the pH increases from 6.9 to as much as 7.4. This causes a change in the solubility of the corrosion products. This change in solubility has the effect that the activated corrosion products on the fuel will move to the coldest parts of the system, e.g. the steam generator cold leg side.

As seen in figure 1 our steam generator doserates continued to increase despite the chemistry modification. During one of our frequent mid-cycle shut downs late 1982 we found that the steam generator doserates had gone down from 135 mSv/h to 115 mSv/h during the few months of operation since the summer outage. Since we did not understand the reason for this nice break in the previously stable rise in steam generator doserates, we started to investigate possible causes.

During the spring of 1983 we came to the conclusion that the primary chemistry modifications at least had contributed to our high dose rates and this was confirmed by a paper[1] at the Bournemouth conference in the fall of 1983.

In order to improve the situation as fast as possible, we increased the pH from 6.9 to 7.1 after mid-cycle shut-downs in Ringhals 2 and 4 in late 1983. The pH was then kept constant for the rest of the cycle and ignoring the minimum specification for lithium. Ringhals 3 did not have a mid cycle shut-down and since we did not want to make the changes in Li (and pH) during operation we operated Ringhals 3 with constant pH of 6.9 through a continuous decrease in Li -content all the way to 0 ppm B and 0.35 ppm Li.

After the 1984 outages we wanted to operate all three units with a positive temperature coefficient throughout the cycle. This meant that we had to raise the maximum Li-content to 3.5 ppm for a limited time (25 to 106 days) in the beginning of the cycle. This level was above the fuel vendor specifications and also above the requirement in the technical specifications. We got approval for a one year test at a maximum level of 3.5 ppm Li in Ringhals 3 and 4 by our plant Safety Review Board, the authorities and the fuel vendors. Ringhals 2 was not operated with 3.5 ppm Li until 1985 because of fear for accelerated corrosion of the heavily crudded fuel. The permission from all parties did prescribe that a prolongation could only be accepted if extensive fuel examination could prove that this type of operation was not deleterious . The Li-levels were kept constant at 3.5 (2.2) ppm until we reached the pH of 7.4. From that point on the pH was kept constant until the end of the cycle. After this test, and after each of the following years with operation at elevated Li-levels, we performed fuel investigations. These included visual inspection and oxide thickness measurements every year and crud sampling some years. The results were encouraging, the fuel performed very well.

After the steam generator replacement in Ringhals 2 in 1989, it was uprated to 109% of the nominal capacity. After operation at about 80 % power for a couple of years, to save the steam generators, our fuel specialists and the fuel vendor feared accelerated fuel corrosion in one batch of fuel with cladding which was especially susceptible to corrosion. We therefore made an agreement with the fuel vendor to operate with maximum 2.5 ppm Li for a few weeks and then keep the pH constant at 7.25 for the remaining cycle. Our ambition was to keep pH as constant as possible in order to avoid solubility changes rather than aim for pH 7.4, which was thought to be the optimum pH.

The same year we also limited the maximum Li-concentration in Ringhals 3 and 4. The reason for this step was the preliminary result from on-going tests at Studsvik, which indicated shorter initiation time for stress corrosion cracking (PWSCC) in Inconel 600 steam generator tubing at elevated Li-concentrations. The Li-concentration was therefore limited to 2.2 ppm at these units.

In 1990 we discovered severe hydriding in the Ringhals 2 control rod guide tubes. It was later found that this hydriding was caused by the manufacturing process but initially the fuel vendor claimed that our chemistry could be the cause. Therefore the Li-concentration was limited to 2.2 ppm during the cycle 1990/91 . The changes in the chemistry of Ringhals 2 is summarised in figure 2 and 5. What is not seen in these diagrams is that from our first modifications in the chemistry we tried to keep the conditions as stable as possible since we believe that chemical transients can be almost as bad for the radiation levels as a low pH. The Li-values used, and the resulting pH-values, are illustrated in figures 3 and 4 for Ringhals 3 and 4 respectively.

Shut down chemistry

At Ringhals we will use a shutdown procedure closely following the intentions of the EPRI guidelines. However, we will use the hot acidic reducing chemistry only during the cool down stage. The residual hydrogen is removed by slow controlled injection of hydrogen peroxide. Then we will add the main amount of hydrogen peroxide and use the time available to remove the dissolved activity during RCP operation.

Start up chemistry

The traditional start-up chemistry involved among other things, chemical degassing by the abrupt addition of hydrazine. Today we use slow controlled injection, which seems to allow a better removal of nickel. Lithium is added when the Residual Heat Removal system (RHR) is disconnected and hydrogen is added just prior to dilution for criticality.

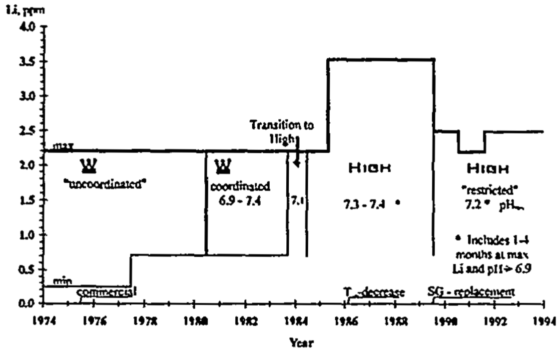

Figure 2. Ringhals 2 RCS lithium - pH history

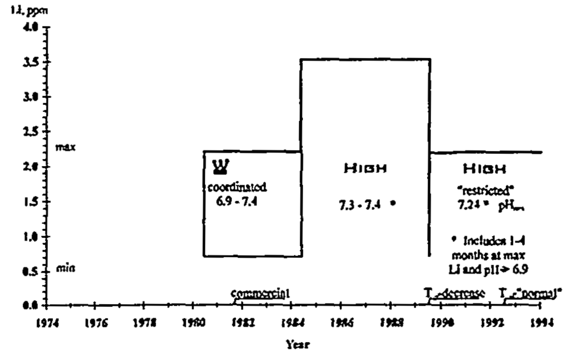

Figure 3. Ringhals 3 RCS lithium - pH history

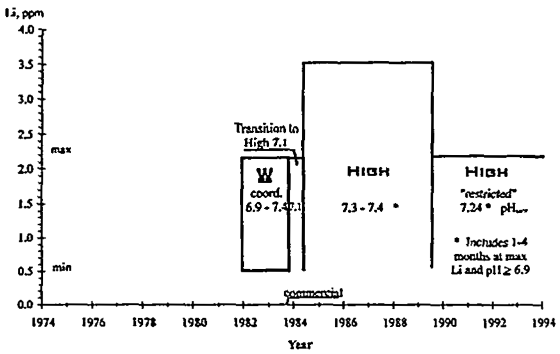

Figure 4. Ringhals 4 RCS lithium - pH history

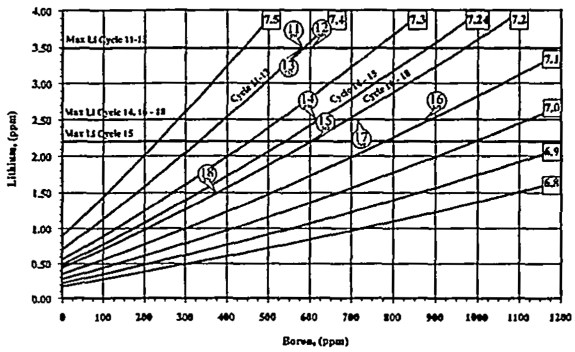

Figure 5. Ringhals 2. Diagram showing boron/ lithium correlation at different pH. Figures in circles indicate boron concentration at the beginning of each cycle.

Results of the modified chemistry

Doserate Measurements in the Steam Generators

From 1983 onwards we have seen an almost steady decrease in Ringhals 2 steam generator dose rates as seen in figure 1, and in Ringhals 3 and 4 the activity build-up was more favourable than in Ringhals 2 . As a matter of fact Ringhals 4 , where we introduced the high pH chemistry after only 0.5 EFPY of operation, is still the cleanest Westinghouse PWR. The early high build-up rate of Ringhals 2 and later improvements might also have been influenced by other effects. We know that some reactor vendors, including W who supplied our initial fuel, had problem with high Co-content in the Ni plating of the fuel spacers at the end of the seventies. Some European plants have experienced much higher dose rates than we did because of this. Later on all fuel vendors have got good control of the Ni-plating. They have also changed from Inconel to Zircalloy, which contains much less Co, in their fuel spacers and this change has reduced the potential for release of Co 60 into the Reactor Coolant System.

Dose Rate Measurement on the Primary System

The result of the measurements in the EPRI standard program indicates a continuous downward trend in dose rates on reactor coolant system. An example from Ringhals 2 is seen in figure 6.

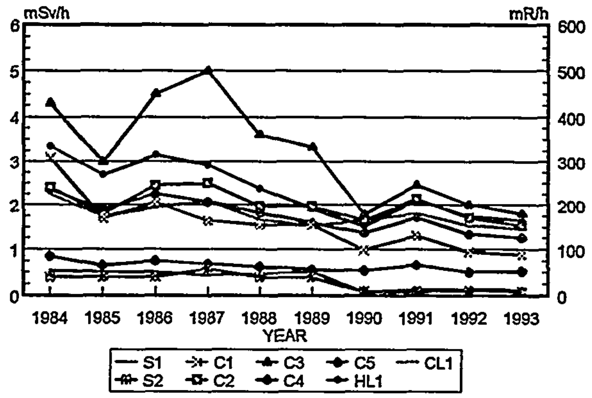

Figure 6. EPRI standard measurement program.
Dose rates on Ringhals 2 primary coolant system loop No 1.

Loose Contamination in Primary System Components

After the first cycle with high pH chemistry in Ringhals 2 we discovered that the amount of loose crud on the primary system surfaces was strongly reduced. This situation has been further improved after the changes in the shut-down chemistry (see below).

Activity Build-up in Regenerative Heat Exchanger after Decontamination

A positive long term effect of a decontamination is shown in figure 7. The regenerative heat exchanger in the Chemical Control (CS)-system was decontaminated in 1986 by the ODP-method. Although the dose rates have increased over the seven years since the decontamination was performed, they are still some three times lower than the original values. The question is, whether the ozone decontamination makes the oxide layer less prone to accumulate cobalt activity or if the new level just reflects a lower source term of Co-60.

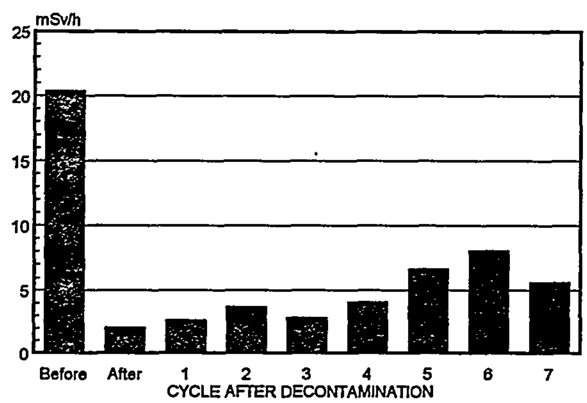

Figure 7. Dose rates on Ringhals 2 regenerative heat exchanger before and after decontamination. Average of 6 measurements.

Nuclide Specific Measurements on the Primary Circuit

One can note that the surface activity levels on the old system parts basically are of the same values as before the steam generator replacement.

The parts that were decontaminated with electropolishing during the steam generator replacement still have surface activity levels which are a factor of four, for Co 60, below the undecontaminated system surfaces.

The "elbows" that were replaced, were electropolished. The activity build-up to present level occurred during the first cycle but the surface activity levels of Co 60 still remain a factor of 14 below the undecontaminated system surfaces.

What happened when we reduced the pH from 7.4 to 7.2 or 7.24?

The change from pH 7.4 to 7.2 or 7.24 (from cycle 14 in Ringhals 2 and cycle 7 in Ringhals 3 and 4) does not seem to have a major influence on the radiation levels, but we are still convinced that pH 7.4 is closer to the optimum. What we have seen in all three units is a shift from having the highest doserates in the hot leg with pH 7.4 to somewhat higher doserates in the cold leg (figure 1). The high dose rate in the hot leg is probably due to residual activity from operation at "uncordinated" and coordinated chemistry at pH 6.9.

Release of Corrosion Products at Shut Down

In recent years much attention has been devoted to shut down chemistry. Traditionally, we have always been using hydrogen peroxide addition as a normal part in the shutdown procedure. It ensures a concentrated release of activated corrosion products, and the subsequent collection on demineraliser, before opening up the systems for refuelling. the activity released during this oxidising phase originates solely from the fuel crud. Thus, it yields no reduction in the doserates from the system surfaces. The long term advantage is that it reduces the amount of material on the fuel, which may be activated in the next fuel cycle.

To reduce the dose rates from the system surfaces, the concept "Hot, Acid, Reducing chemistry (HARC)" has been tested in many plants in recent years. The most striking results were obtained at Zion 2, where steam generator dose rates were reduced to about 40%. However, the total time used for this effort was some 70 days, which normally cannot be included in a shut down procedure.

We used HARC during the 1993 shut down of Ringhals 2 and our observations are in agreement with the Zion 2 data. However, the release rate during the reducing phase of the shutdown is much lower than during the oxidising phase after the addition of hydrogen peroxide. Although we did not observe any significant influence on the dose rates in the steam generators we found our channel heads extremely clean with regard to loose contamination.

We also noticed a small reduction in the Co-58 activity on the cross leg (some 20 %) by on-line gamma scan and doserate measurements. The Co-60 activity was not affected. During the subsequent oxidising phase, a recontamination of Co-58 occurred, so only about half of the initial reduction in Co-58 persisted. There were also some indications that the doserates were higher in the loop where the RCP was stopped first, at the higher concentration of Co-58. It seems as if this recontamination is reversible, provided that the surfaces are exposed to low concentration water at a high flowrate (enhances mass transfer from surfaces to solution). The conclusions from this is that we will use clean-up most of the available time during shutdown to remove activated corrosion products after the addition of hydrogen peroxide.

Table 1 shows a simplified balance of the inventory of gamma source strength in Ringhals 2. It can be seen that roughly 20 % of the total gamma source strength from corrosion products are removed in the shutdown transient. and that approximately the same amount remains on the fuel cladding.

It can also be seen that about 50 % of the gamma source strength resides on the stainless steel surfaces. The build-up on the new SG-tubing (1989) is some 10 %.

Table 1. Ringhals 2 shut down 1993. Gamma source strengths on various system parts

System part	Area,	Gamma source strength					
Co-58: 0,97 MeV/Bq	m2	Mev/s			Fraction of total		
Co-60: 2,5 MeV/Bq		Co-58	Co-60	Sum	Co-58	Co-60	Sum
SG-TUBING	15315	4,7E+12	5,8E+12	1,1E+13	4%	5%	10%
Stainless steel	2240	1,6E+13	3,6E+13	5,2E+13	15%	33%	48%
Fuel, Zircaloy cladding	4525	1,8E+13	4,9E+12	2,3E+13	17%	4%	21%
Shut down release		2,3E+13	6,8E+11	2,4E+13	21%	1%	22%
Sum		6,2E+13	4,7E+13	1,1E+14	57%	43%	100%
Reducing release/fract of SS		3,4E+12	3,7E+11	3,7E+12	6%	1%	7%
Based on	Gammascan of SG, MWI and loop pipes Radiochemical analysis of shut down transient Radiochemical analysis of crud samples from fuel cladding						

Table 2. Ringhals 2 shut down 1987. Gamma source strengths on various system parts

System part	Area,	Gamma source strength					
Co-58: 0,97 MeV/Bq	m2	Mev/s			Fraction of total		
Co-60: 2,5 MeV/Bq		Co-58	Co-60	Sum	Co-58	Co-60	Sum
SG-TUBING	14145	2,7E+12	4,8E+13	5,0E+13	2%	41%	43%
Stainless steel	2240	6,4E+12	4,8E+13	5,5E+13	6%	42%	47%
Fuel, Zircaloy cladding	4525	3,0E+12	6,8E+11	3,7E+12	3%	1%	3%
Shut down release		5,9E+12	1,8E+12	7,7E+12	5%	2%	7%
Sum		1,8E+13	9,8E+13	1,2E+14	16%	84%	100%
Based on	Gammascan of MWI, pulled tubes 1985 and replaced SG 1989 Radiochemical analysis of shut down transient Radiochemical analysis of crud samples from fuel cladding						

A comparison with 1987 data (before SG replacement) in table 2 indicates that the total gamma source strength is roughly the same as in 1993. However, the amount of Co-60 has been reduced by a factor of almost two. The SG replacement could be considered as a "50 % system decontamination with respect to Co-60" and probably, cobalt as a source for activation.

The 1987 data in table 2 indicates that operation at high pH throughout the cycle minimises the residual activity on the fuel cladding, compared with the modified operation mode employed in 1992 and 1993.

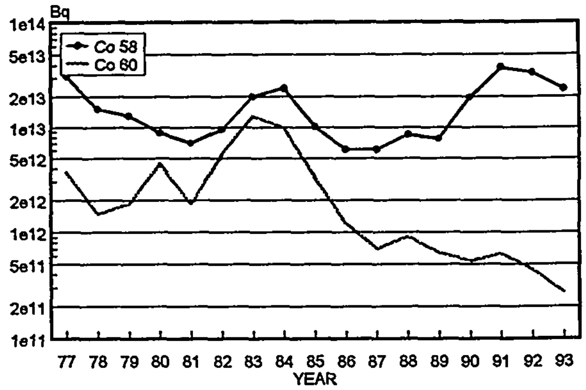

Figure 8. Ringhals 2. Reactor coolant total releases of Co 58 and Co 60 at shut down.

From figure 8 it can be seen that the release of Co-60 at shutdown has decreased over the years. It very probably reflects a decreasing inflow of cobalt to the reactor system. The low releases of Co-58 in 1987-1989 very probably reflects the effect of operation at pH = 7.4, which minimises the solubility and the transport to the core of the corrosion products.

The increasing amounts of Co-58 in the years 1990-1992 probably reflects the corrosion of the fresh SG-tubing material after the SG-replacement in 1989.

FEASIBILITY TEST TO REMOVE FUEL CRUD IN THE FUEL STORAGE POOL.

In the Ringhals PWRs the core is completely unloaded to the spent fuel pool during refuelling. We have observed that the high temperature in the spent fuel pool increases the dissolution rate of activated corrosion products. We have estimated that at 40 °C we may in 3 months remove the same amount of gamma source strength as during the shut down transient.

This method may be used in two different ways: either during prolonged shutdowns, e.g. in connection with SG replacements, or routinely during the cycle to clean the fuel that is to be reloaded into the reactor.
The potential of these methods are presently being investigated at Ringhals. Addition of hydrogen peroxide to the fuel pool water will also be tested in combination with the high temperature.

MATERIAL SPECIFICATION TO AVOID HIGH COBALT CONTENT

Ringhals 2 and 3 Replacement Steam Generators

The surface area of the steam generators is very large in relation to the rest of the primary system. We therefore felt that it was important to specify as low Co-content in the steam generator tubing of the replacement steam generators as was technically achievable and economically feasible. In our specification for the new steam generators for Ringhals 2 we required an average Co-content in the Inconel 690 tubing of 0.015 %. This specification was considerably lower than we had in our older generators but also lower than specified in earlier replacement generators. This low Co-content combined with the lower corrosion-rate in Inconel 690 (compared with Inconel 600) and a positive temperature coefficient means a reduction in corrosion products transported to the fuel surfaces.

The tubing for the steam generators for Ringhals 3, which will be replaced in 1995, are specified with a maximum Co-content of 0.02 %. The samples taken indicate an average value below 0.015 %.

Policy for replacement materials

In order to avoid uncontrolled introduction of new stellite components or other high Co-material the following policies are adopted:

- The project department has the responsibility to assure that new Co-sources are not introduced when old systems are modified or new systems installed. Project reports for modifications must include very good motivations if they want to use materials with Co content higher than specified in table 3.

- Stainless steel used in primary systems shall have as low Cobalt content as possible and always below 0.05 %.

- When stellite hard surfaces have to be replaced for functional reasons alternative low Cobalt materials have to be used.

- Strategic materials with low cobalt content shall be stored to avoid the necessity to use what happens to be available on the market.

Fuel Components

Table 3. Cobalt specification for core components.

Components (BWR and PWR)	Material	Maximum Co content
Top end piece	Stainless steel	0.04 % (440 ppm)
Bottom end piece	Stainless steel	0.04 % (440 ppm)
Springs (not spacers)[a]	Inconel	0.04 % (440 ppm)
Centering pins[a]	Stainless steel	0.04 % (440 ppm)
Bolts[a]	Incoloy	0.04 % (440 ppm)
Spacer grids	Inconel	0.02% (200 ppm)
Spacer grids	Zircaloy	0.001% (10 ppm)
Fuel cladding	Zircaloy	0.001% (10 ppm)
Guide thimble tubes (PWR)	Zircaloy	0.001% (10 ppm)
Fuel channels (BWR)	Zircaloy	0.001% (10 ppm)

[a] Minor deviations can be accepted, provided that total area Co-content as average is less than 0.04% (440 ppm).

New Low Pressure Turbines in Ringhals 1 (BWR)

A special case is the replacement low pressure turbines ordered for our BWR Ringhals 1. Some of the vendors offering the new turbines wanted to use their standard specification with Stellite surfaces to prevent erosion by the wet steam. Despite full flow cleaning of the condensate the input of cobalt from the turbine system is significant in relation to the contribution from the reactor systems. We therefore managed to include a requirement to have no Stellite blading, except in a very limited number of blades, although the vendor had no experience and therefore was forced to develop an alternative erosion protection.

SUMMARY

As seen in figure 9 collective exposures have been lowered considerably since 1983, when we started our first efforts to optimise our primary chemistry. If you disregard those years with prolonged outages (1989 SG replacement, 1992 and 1993 extensive work on vessel head penetrations) there is a steady reduction in doses. We believe that our high pH and modified chemistry has been of importance but of course we have also spent a lot of efforts to develop working methods, robotics, training etc. We have also been able to reduce the normal outage from 50 to 75 days ten years ago to 27 to 35 days today and avoid forced outages by improved maintenance programs. All these factors do of course contribute to the results we have acieved.

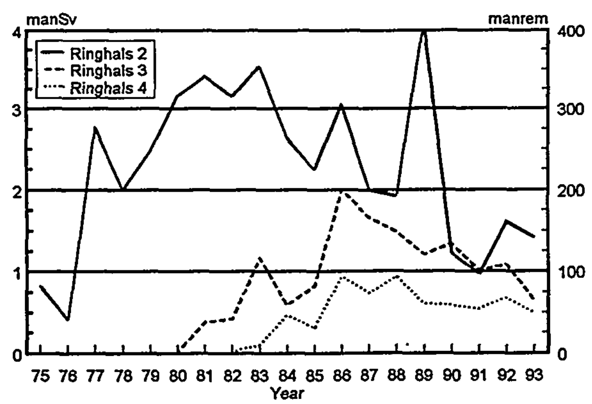

Figure 9. Ringhals 2, 3 and 4 annual collective exposures.

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Author Biography

Krister Egnér is Manager of the Environment and Safety Department at the Ringhals Nuclear Power Plant. For 15 years he has been the licensed Radiological Supervisor for the Ringhals site and a member of the Ringhals Executive Group. He has been the Swedish representative in the discussions leading to the formation of the ISOE system and, until recently, a member of the ISOE Steering Committee. He is the Chairman of the Reactor Safety Review Committee for the Ringhals site and a member of the Vattenfall Corporate Reactor Safety Review Committee. He has a M.Sc. in Mechanical Engineering from Chalmers University of Technology in Gothenburg, Sweden.

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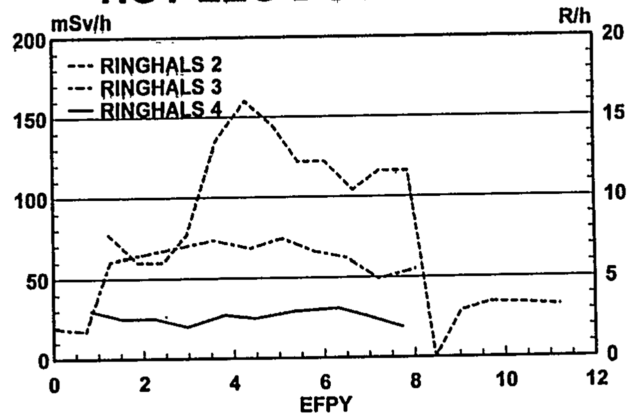
CONTENTS

- PRIMARY CHEMISTRY
  - BACKGROUND
  - OPERATING CHEMISTRY
  - SHUT-DOWN CHEMISTRY
  - START-UP CHEMISTRY

CONTENTS (continued)

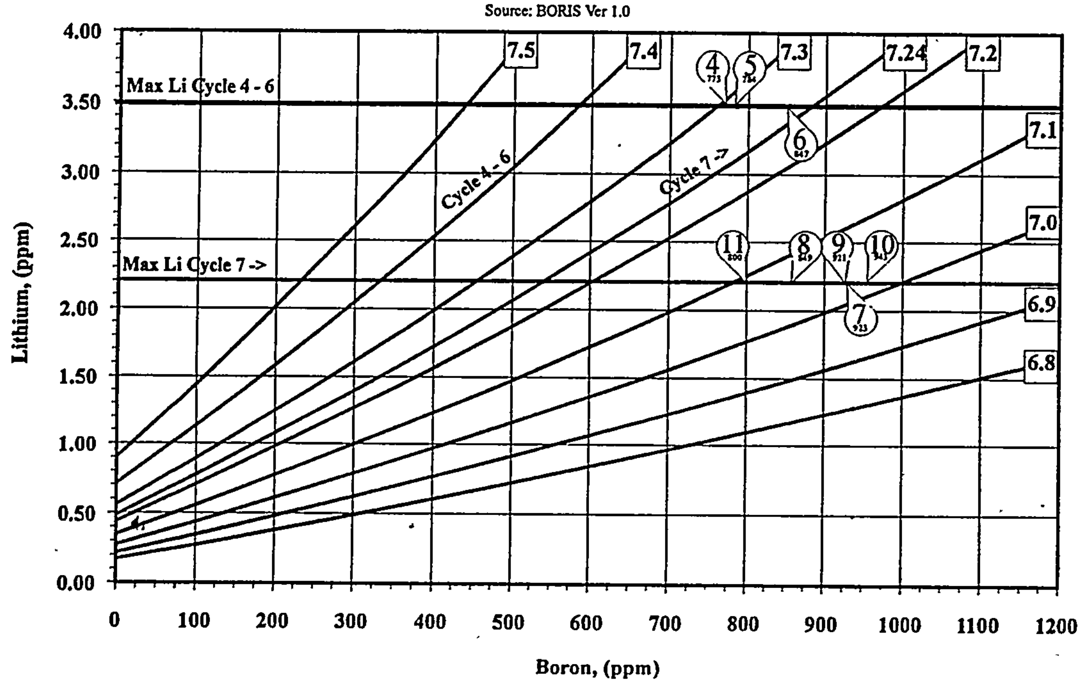
- TEST TO CLEAN FUEL
- REPLACEMENT MATERIALS
  - STEAM GENERATORS
  - R1 (BWR) LP TURBINES
  - OTHER MATERIAL
- SUMMARY

RINGHALS 2, 3 AND 4. AVERAGE HOT LEG DOSERATES



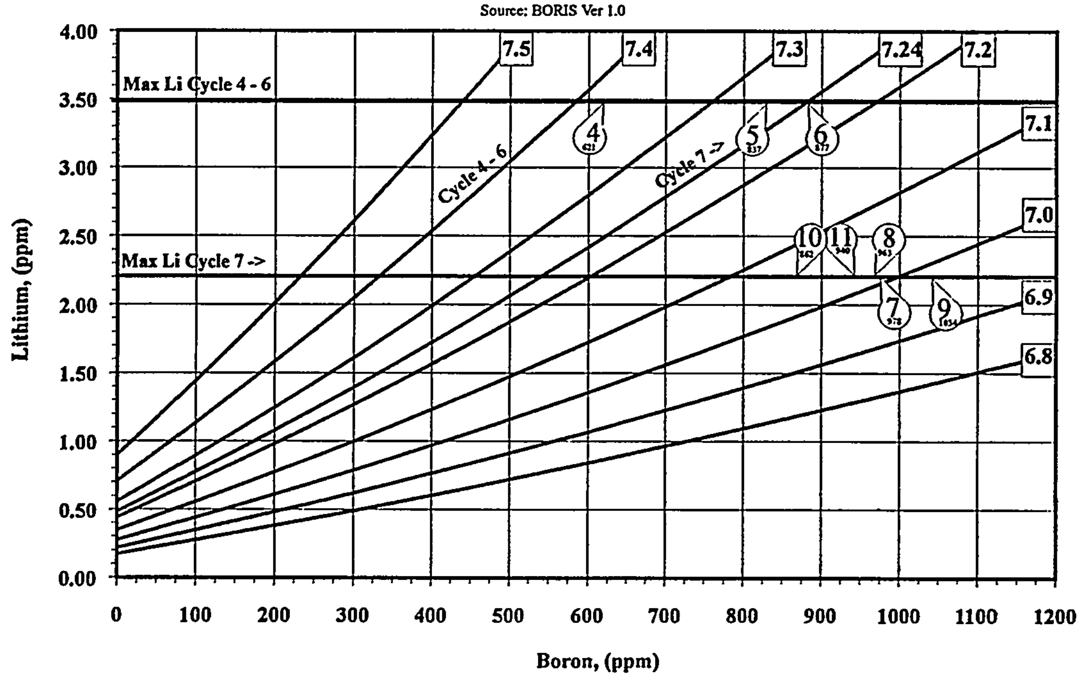
Ringhals Unit 3

B / Li Correlation @ different pH. Temperature = 300°C

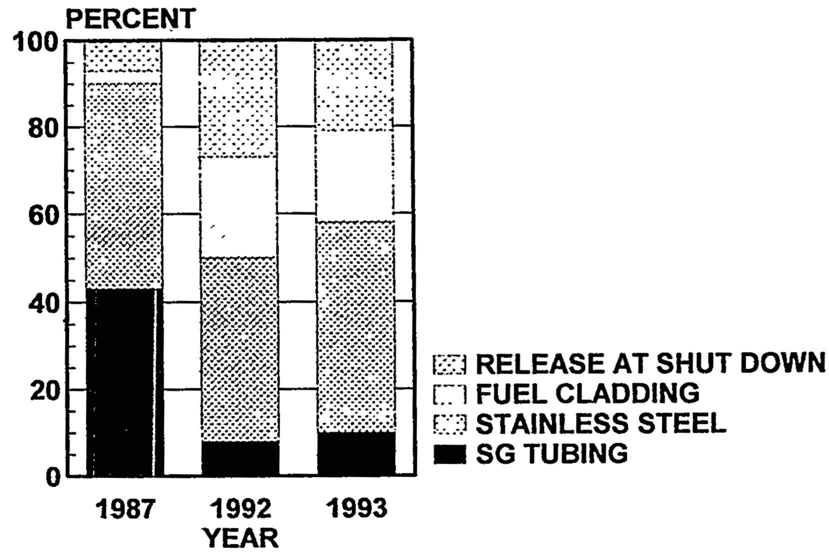


Ringhals Unit 4

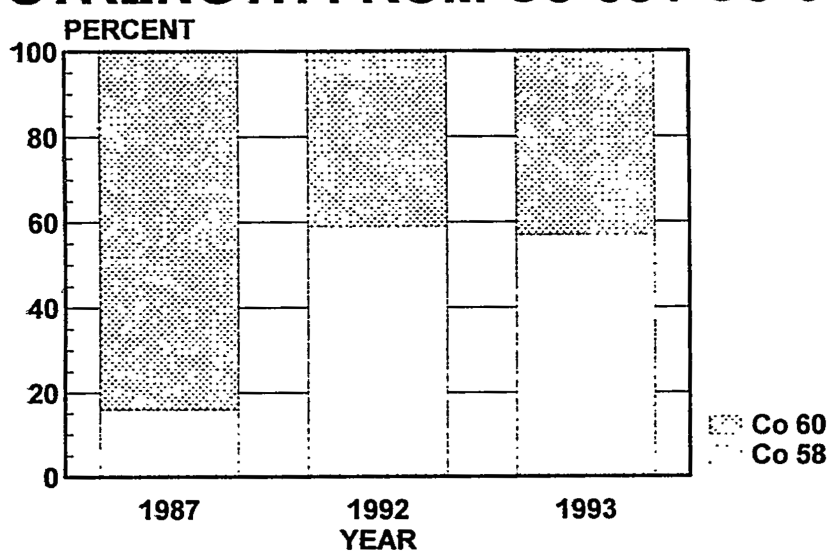
B / Li Correlation @ different pH. Temperature = 300°C



RINGHALS 2. GAMMA SOURCE STRENGTH ON VARIOUS SURFACES



RINGHALS 2. GAMMA SOURCE STRENGTH FROM Co-58 / Co-60



POLICY FOR REPLACEMENT MATERIALS

- STRICT LIMITATION OF COBALT CONTENT WHEN COMPONENTS ARE REPLACED
- STAINLESS STEEL IN PRIMARY SYSTEMS SHOULD CONTAIN LESS THAN 0,05 % COBALT
- BAN ON NEW STELLITE COMPONENTS
- STRATEGIC LOW COBALT MATERIAL STORED ON SITE

COBALT LIMITATION FOR CORE COMPONENTS

STAINLESS STEEL	0,04 %
ZIRCALOY	0,001 %
INCONEL	0,02 - 0,04 %
INCOLOY	0,04 %

REPLACEMENT STEAM GENERATORS IN RINGHALS 2 AND 3 HAVE A COBALT CONTENT OF LESS THAN 0,015 %

AN OVERVIEW OF ZINC ADDITION FOR BWR DOSE RATE CONTROL

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ABSTRACT

This paper presents an overview of the BWRs employing feedwater zinc addition to reduce primary system dose rates. It identifies which BWRs are using zinc addition and reviews the mechanical injection and passive addition hardware currently being employed. The impact that zinc has on plant chemistry, including the factor of two to four reduction in reactor water Co-60 concentrations, is discussed. Dose rate results, showing the benefits of implementing zinc on either fresh piping surfaces or on pipes with existing films are reviewed. The advantages of using zinc that is isotopically enhanced by the depletion of the Zn-64 precursor to Zn-65 are identified.

INTRODUCTION

Beginning in 1982, analysis of historical BWR radiation buildup data, sponsored jointly by the Electric Power Research Institute (EPRI) and GE Nuclear Energy, identified a correlation between low primary system dose rates and the presence of ionic zinc in the reactor water. Several BWRs were found to have zinc in the reactor water because they had a brass condenser and a powdered condensate treatment system. The brass provided a source of zinc which was not totally removed by the powdered resin condensate system. This resulted in reactor water soluble zinc concentrations of 5 to 15 ppb. These plants were dubbed 'natural zinc' plants and served as the foundation of the correlation. This correlation was hypothesized to be the result of a corrosion inhibition effect of zinc for stainless steel. Subsequent laboratory testing confirmed that ionic zinc is strongly incorporated into the protective oxide film which forms on stainless steel surfaces and that this film is more protective to the base metal than films formed without zinc present. As a result, a thinner layer of oxide is sufficient to curtail the corrosion process. Figure 1 shows the relationship between oxide film thickness and the concentration of ionic zinc in the water for laboratory tests conducted at BWR conditions of temperature and pressure.

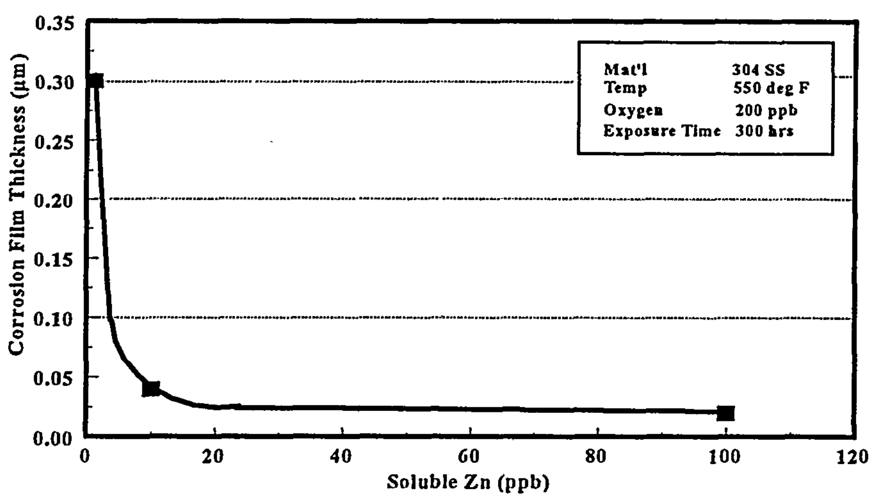

Figure 1. Effect Of Zinc On Stainless Steel Corrosion.

Additional testing was required to verify that the corrosion inhibition effect established by the data shown in Figure 1 would produce the reduced radiation buildup identified in the plant correlation. These tests showed that the corrosion inhibition effect of zinc did result in reduced Co-60 buildup on stainless steel under either normal BWR water chemistry (150-200 ppb oxygen) or hydrogen water chemistry (<15 ppb oxygen). Some of the data from these tests are shown in Figures 2 and 3.

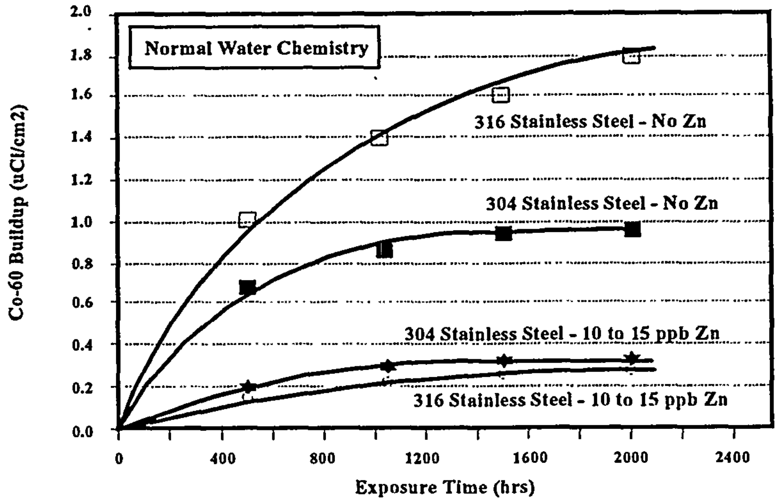

Figure 2. Effect Of Zinc On Radiation Buildup Under Normal Water Chemistry.

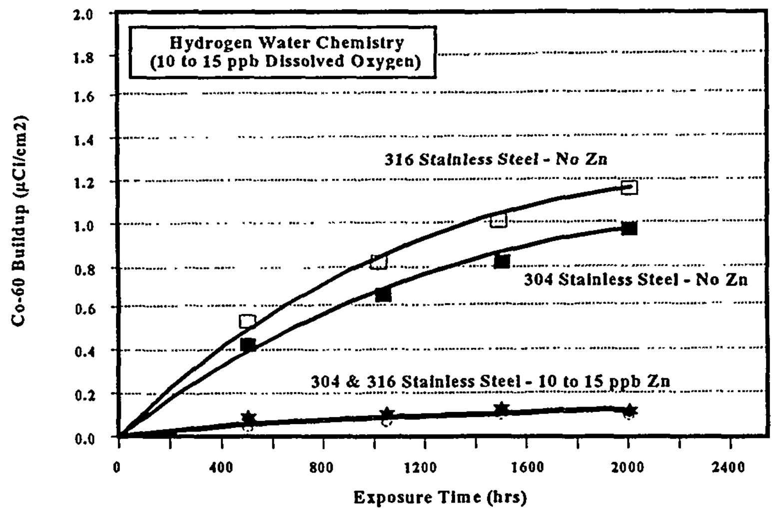

Figure 3. Effect Of Zinc On Radiation Buildup Under Hydrogen Water Chemistry.

The data analysis and laboratory testing described above have been documented in published reports[1,2,3] and resulted in the development by GE Nuclear Energy of systems for the addition of zinc to the BWR primary system under the registered trademark of **GEZIP**.

BWRs USING GEZIP

In the Fall of 1986, Hope Creek became the first BWR to intentionally add zinc to the primary system. At this time, there are a total of fourteen BWRs which have implemented GEZIP and several additional plants which are currently evaluating implementation. Figure 4 provides a list of the plants using GEZIP and a time line reflecting the implementation calendar.

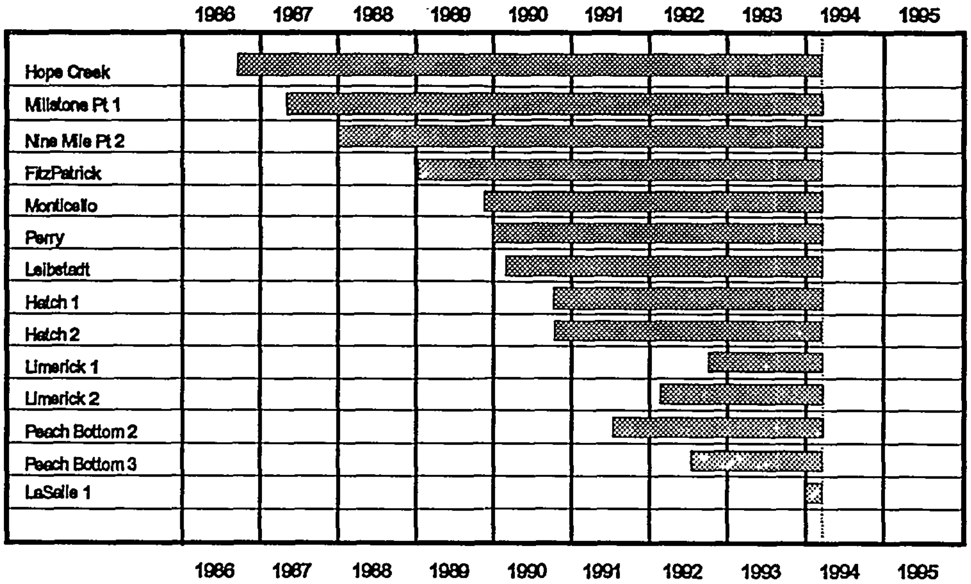

Figure 4. BWR Implementation Chart For GEZIP.

ZINC ADDITION SYSTEMS

As might be expected, the design of systems for adding zinc to the BWR have evolved since the first unit was installed at Hope Creek. This first unit used a low flow, positive displacement pump to inject a zinc oxide suspension into a recirculation loop around the final feedwater pump. Subsequent enhancements to this design yielded an improved, higher flow rate injection system which injected directly into the feedwater pipe and, thus, required no recirculation loop. Further innovation has yielded a passive design with no moving parts. The designs used in each of the GEZIP plants is provided below.

Table 1. GEZIP Equipment Application By Plant

Low Flow Pump	High Flow Pump	Passive Addition
Hope Creek	Monticello	Leibstadt
Millstone Pt 1	Limerick 1	Perry
Nine Mile Pt 2	Limerick 2	Hatch 1
FitzPatrick	Peach Bottom 2	Hatch 2
	Peach Bottom 3	LaSalle 1*

* - LaSalle 1 is temporarily using a High Flow Pump system until their Passive System is ready.

Simplified flow schematics and descriptions for these systems are included in the following sections.

Mechanical Injection

Figure 5 presents a schematic of the original, skid mounted, Low Flow Pump System for zinc injection. In this system, two, redundant, diaphragm pumps inject zinc oxide suspension from the continuously agitated supply tank into a recirculation pipe which takes suction on the downstream side of the final feedwater pump and returns it to the upstream side of the final feedwater pump. The injection pump flow rate is approximately 30 ml/min and the recirculation loop flow rate is approximately 50 gal/min.

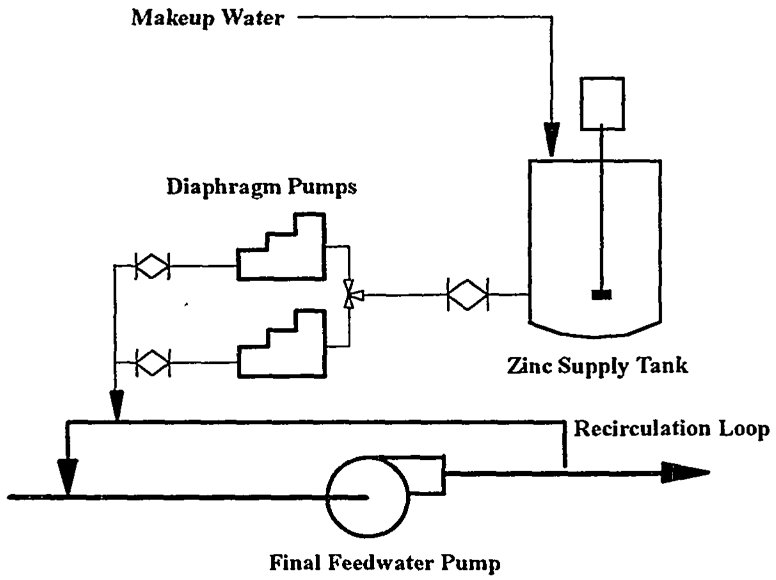

Figure 5. Low Flow Pump System for zinc injection.

Figure 6 presents a schematic of the High Flow Pump System for zinc injection. In this system dilution condensate is provided at the suction side of the injection pumps to improve flow characteristics and permit the use of larger pumps (approximately 300 ml/min) which have inherently larger components and provide improved performance. With this increased pump output, the recirculation loop around the feedwater pump is not needed. Thus the injection pump output is fed directly into the feedwater pipe on the suction side of the final feedwater pump.

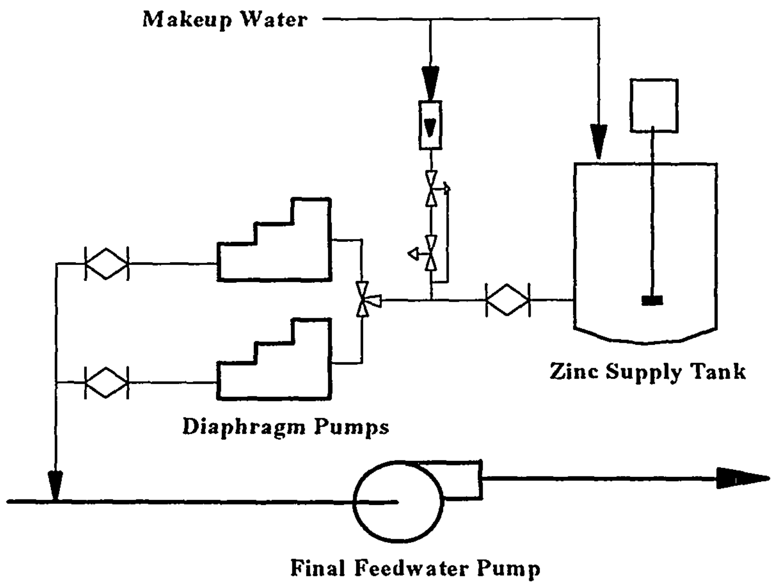

Figure 6. High Flow Pump System for zinc injection.

Passive Addition

The passive system was developed so that the operating and maintenance requirements of zinc addition would be minimized. The schematic for this system is shown in Figure 7.

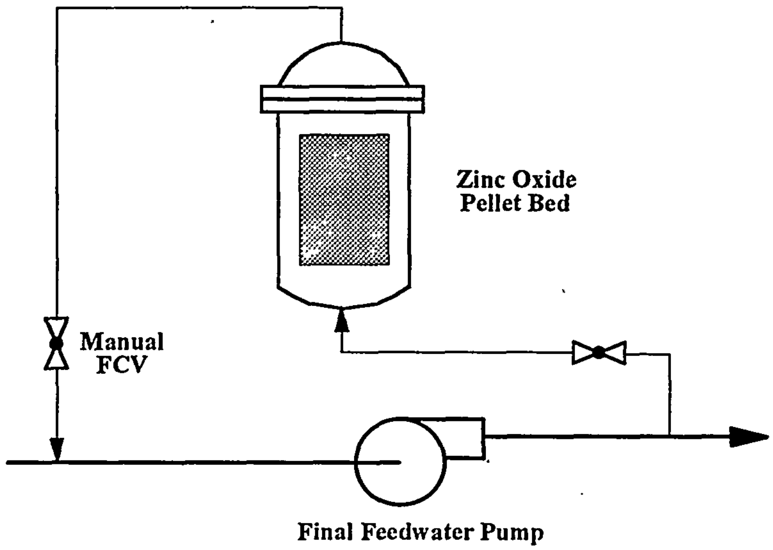

Figure 7. Passive zinc addition system.

In this system, a bed of sintered zinc oxide pellets is contained in a small pressure vessel. A bypass stream of less than 100 gpm is taken from the discharge side of the final feedwater pump, passed through the pellet bed and returned to the suction side of the feedwater pump. Sufficient zinc is dissolved from the pellets to maintain the desired concentration of zinc in the reactor water. The pellet bed is designed such that it will last at least one complete fuel cycle without requiring additional pellets. By virtue of its having no moving parts, this system provides a zinc addition option with essentially no maintenance and requiring minimal operator attention.

IMPACT OF ZINC ON REACTOR WATER Co-60

One of the impacts of zinc addition, which was impossible to anticipate prior to GEZIP, was the suppression of the reactor water Co-60 concentration. 'Natural Zinc' plants had zinc present from the first cycle of operation and, thus, there was no opportunity to know that this suppression was occurring (i.e. there was no 'before and after' available for comparison). Likewise, the first application of GEZIP was at Hope Creek, a new plant, and therefore there was no comparison basis there either. However, when zinc was introduced at Millstone Pt 1 for the first time, in April, 1987, the impact was immediately observed. It has subsequently been repeated at each BWR which has introduced zinc after operating for one or more cycles as a non-zinc plant. The marked impact observed at Millstone Pt 1, commencing with initial zinc injection, is shown in Figure 8. The comparison of total reactor water Co-60 before and after zinc addition is shown for Millstone Pt 1, FitzPatrick, Monticello, and Leibstadt in Figure 9.

Zinc acts to lower the Co-60 in two ways. First, it suppresses the corrosion release rate for in-core cobalt alloys, such as the stellite rollers and pins. Second, it is incorporated into the iron-based fuel deposits and results in an oxide which releases Co-60 at a lower rate.

This effect results in additional reduction of Co-60 buildup on primary system piping and components, as well as decreasing the curies of Co-60, and Co-58 which enter the radwaste as a result of capture in the reactor water cleanup system. The curies of Co-60 and Co-58 released to the reactor water at shutdown are also reduced as a result of this suppression.

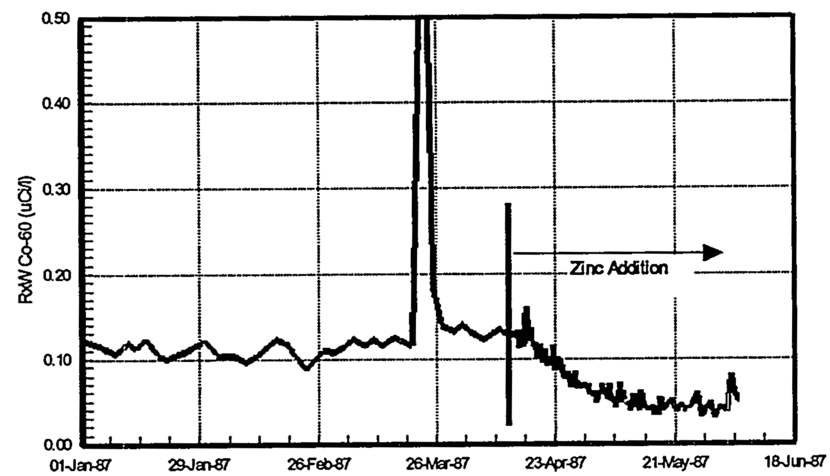

Figure 8. Suppression of reactor water soluble Co-60 by zinc addition at Millstone Pt 1.

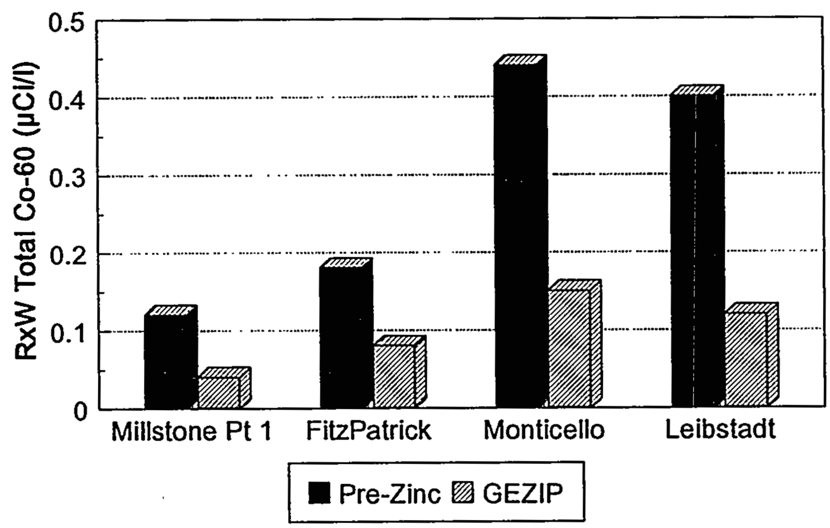

Figure 9. Reduction of reactor water Co-60 as a result of zinc addition.

IMPACT OF GEZIP ON PRIMARY SYSTEM DOSE RATES

The probable impact of GEZIP implementation at any BWR is dependent on several factors. Included in these factors are the following:

1. Was the plant a 'Natural Zinc' plant just prior to GEZIP implementation?
2. Is the plant a new plant?
3. If the plant is an operating plant, has a chemical decontamination been performed?
4. Is the plant using Normal Water Chemistry or Hydrogen Water Chemistry?

Several of the natural zinc plants identified in the original studies have since either replaced their brass condenser or added deep bed demineralizers to the condensate treatment system and thus lost their source of zinc to the reactor. Implementing zinc at these BWRs would be expected to maintain the historically low radiation buildup that they have experienced.

The expectation for pipes in a new plant at initial startup, and for pipes which have just experienced a successful chemical decontamination, are that subsequent radiation buildup should be analogous to the laboratory test data shown earlier in the report. Thus radiation buildup should be slower and equilibrium dose rates should be significantly below the average of non-zinc BWRs.

For piping systems in operating plants, which have an existing oxide film, formed during one or more cycles of operation and not subjected to a chemical decontamination, zinc can gradually alter the structure of that film so that dose rates will decrease gradually as exposure to zinc progresses.

For reference purposes, the BWRs listed above implemented GEZIP under the circumstances displayed in the table below.

Table 2. Status Of Plants Implementing GEZIP

New Plants	Non-Zinc Operating Plant w/ Chem Decon	Non-Zinc Operating Plant no Chem Decon	Nat'l Zinc Operating Plant
Hope Creek	Millstone Pt 1	Perry	Hatch 1*
Nine Mile Pt 2	FitzPatrick*	Leibstadt	Hatch-2*
	Monticello*		Limerick 1
	LaSalle 1		Limerick 2
			Peach Bottom 2
			Peach Bottom 3

* - Reactors using Hydrogen Water Chemistry.

Normal Water Chemistry (NWC)

Hope Creek and Nine Mile Pt 2

Both Hope Creek and Nine Mile Pt 2 initiated GEZIP at the start of the first cycle of operation and have operated with NWC through the period covered in this report. Figure 10 shows the radiation buildup at these reactors over the first few cycles.

During the first cycles at each of these reactors, the zinc concentration in the reactor water was maintained in the range of 5 to 10 ppb. In subsequent cycles, concern about Zn-65 has resulted in operation at approximately 2 ppb. Even with the lower than recommended zinc concentration, both plants have experienced dose rates which are well below the non-zinc BWR average of approximately 300 mR/hr.

With dose rates <100 mR/hr, Hope Creek is in the lowest group of BWRs with respect to dose rates. This reactor has recently converted to both DZO (*D*epleted *Z*inc *O*xide, discussed later in the report) and HWC. It is expected that the impact of DZO will not be fully observed for several cycles because of the large natural zinc (i.e. Zn-64) inventory present in the reactor. HWC may result in an increase in dose rates over, at least, the next few cycles.

Nine Mile Pt 2 has higher dose rates than Hope Creek and is currently at 184 mR/hr. Niagara Mohawk is currently evaluating a switch to DZO as the zinc source material. This higher dose rate is the result of two factors. First, and most important, the average pipe wall thickness is only 1.0 in. compared to the more typical value of 1.25 in. for BWRs. This reduces the self-shielding of the pipe and results in higher dose rates for the same surface concentration of isotopes. Correcting for this thinner pipe and normalizing to the 1.25 in. wall thickness would result in an average dose rate of ~130 mR/hr.

Nine Mile Pt 2 also has higher than average insoluble activity in the reactor water and it is believed that deposition of this particulate matter is contributing more dose rate at the pipe surface than is typical.

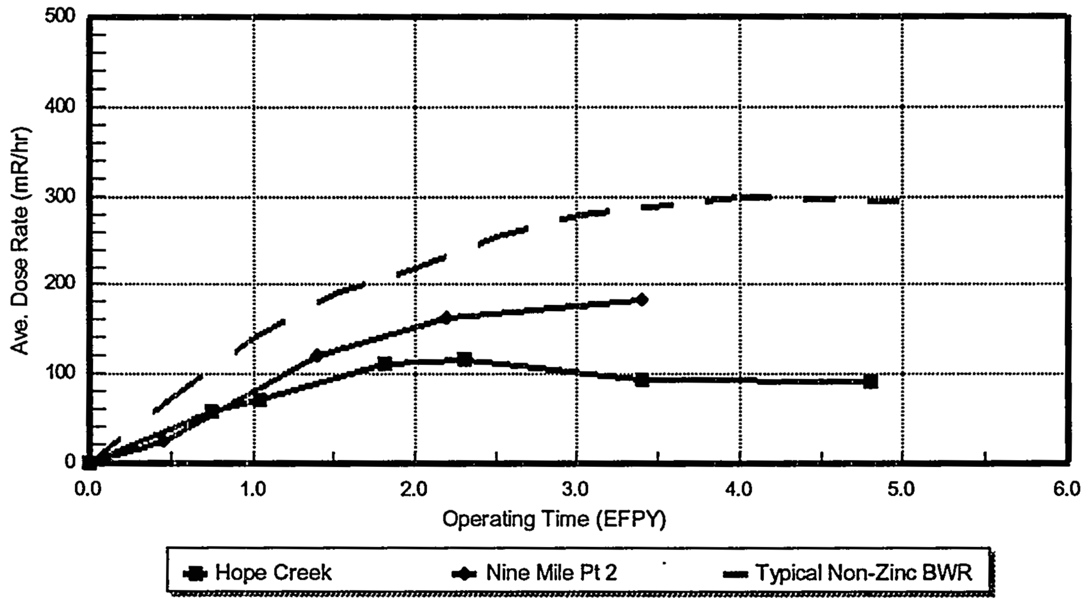

Figure 10. Radiation buildup with GEZIP at Hope Creek and Nine Mile Pt 2.

Millstone Pt 1

Figure 11 provides the history of radiation buildup at the Millstone Pt 1 reactor since they chemically decontaminated the primary system in 1984.

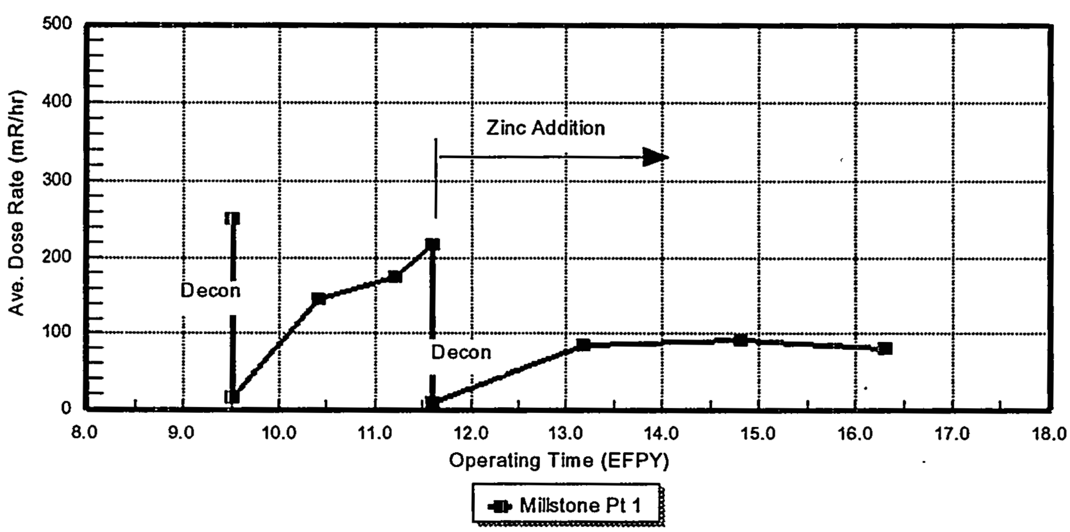

Figure 11. Radiation buildup with GEZIP at Millstone Pt 1.

Northeast Utilities implemented GEZIP on a test basis for two months prior to the refueling outage in 1987, and then continued on a permanent basis after another chemical decontamination in that outage. This gives an excellent opportunity to compare one reactor both with and without zinc addition.

In the period 1984 to 1987, doses increased from a post-decon value of ~10 mR/hr to a pre decon value of 217 mR/hr. Following initiation of GEZIP, dose rates have remained at <100 mR/hr. In the most recent cycle, Millstone has begun using DZO. Again, the inventory of natural zinc created over two cycles of operation will dictate that the impact of DZO will be obscured for a few cycles.

LaSalle 1

The LaSalle 1 reactor began GEZIP operation, using DZO, in January, 1994 at the end of their current cycle. A chemical decontamination will be performed during the Spring refueling outage. The impact of GEZIP at LaSalle 1 will not be known until, at least, 1995.

Perry

The Perry plant implemented GEZIP after one full cycle of operating as a non-zinc reactor. The radiation buildup for this plant is shown in Figure 12. The average dose rate reached 100 mR/hr in the cycle prior to GEZIP and has since increased to 173 mR/hr. Recent data received from plant personnel indicate that dose rates have leveled off at 200 mR/hr. While this average dose rate is still significantly below the non-zinc BWR average, it is somewhat higher than might be expected. A review of the reactor water chemistry data for this period indicates that the insoluble Co-60 concentration is varying between 0.2 and 2.0 µCi/l. This is one to two <u>orders of magnitude</u> higher than the typical BWR (normally 0.02 to 0.05 µCi/l) and suggests that insoluble deposition may be playing a greater role in the buildup at Perry. Perry continues to use natural zinc as the feedstock and maintained a reactor water zinc concentration of 4 to 6 ppb over this period.

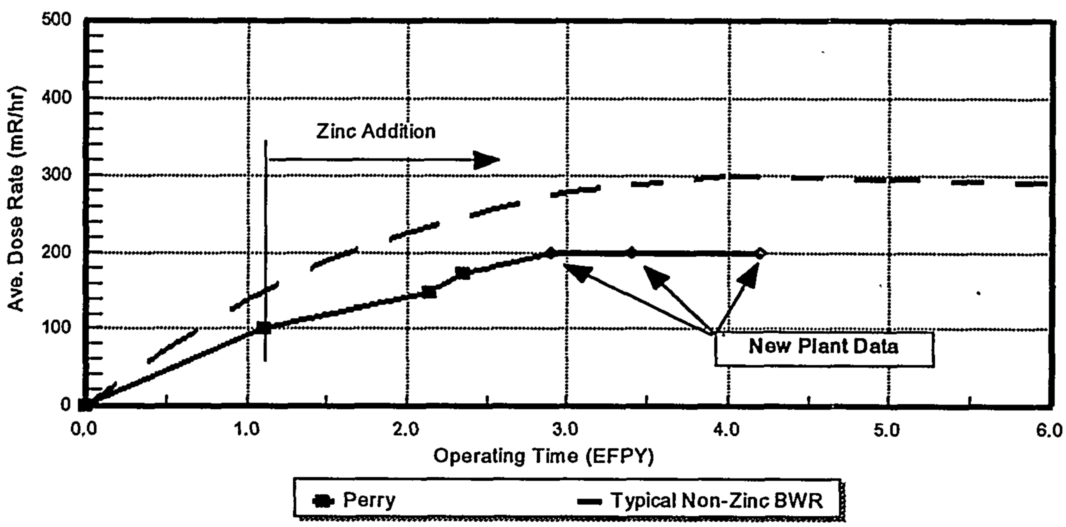

Figure 12. Radiation buildup with GEZIP at Perry.

Leibstadt

The Leibstadt plant installed GEZIP in the middle of fuel cycle 6 and, consequently, started zinc addition with no chemical decontamination. This was the first BWR to use the passive zinc addition system.

Wishing to minimize the impact of Zn-65 on the plant, Leibstadt has elected to maintain zinc concentrations in the reactor water in the 2 to 3 ppb range. This has undoubtedly impacted the rate of dose rate reduction observed. Figure 13 shows the steady decrease in the average dose rate obtained, even with the lower than recommended zinc concentration. At the most recent refueling outage in August, 1993, the measured dose rate appeared to show a slight increase from 196 to 203 mR/hr. The gamma scan data taken during this survey provided information that explained the dose rate data. The Co-60 loading on the pipe had continued to drop at a significant rate (~11%), as shown in Figure 14, but deposited fission products (Ru-103, Zr-95, and Nb-95) from a failed fuel bundle contributed 16% to the dose.

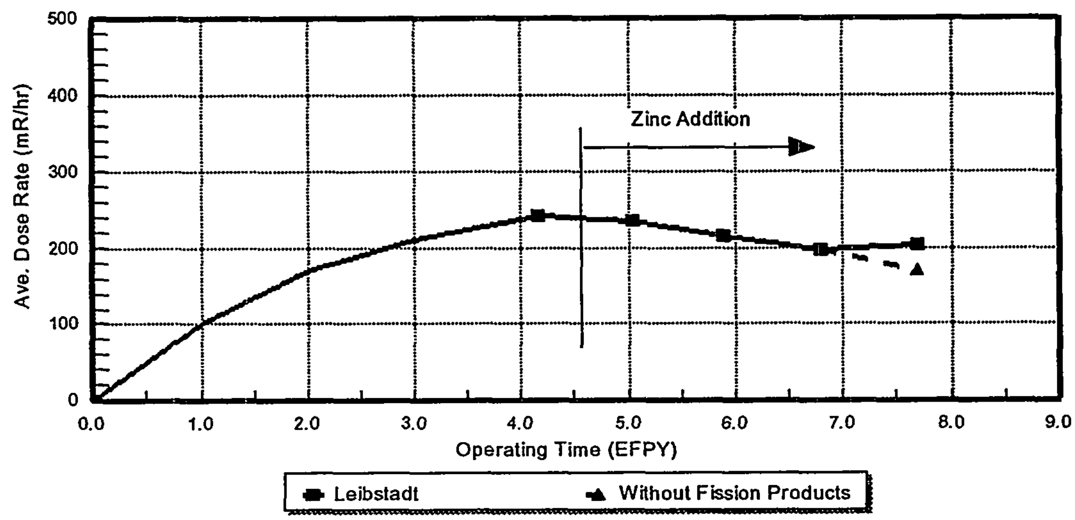

Figure 13. Radiation buildup with GEZIP at Leibstadt.

Figure 13 shows a calculated data point for the 1993 dose rate of 170 mR/hr which is based on subtracting the fission product contribution.

Leibstadt used DZO for this past fuel cycle and will continue to do so.

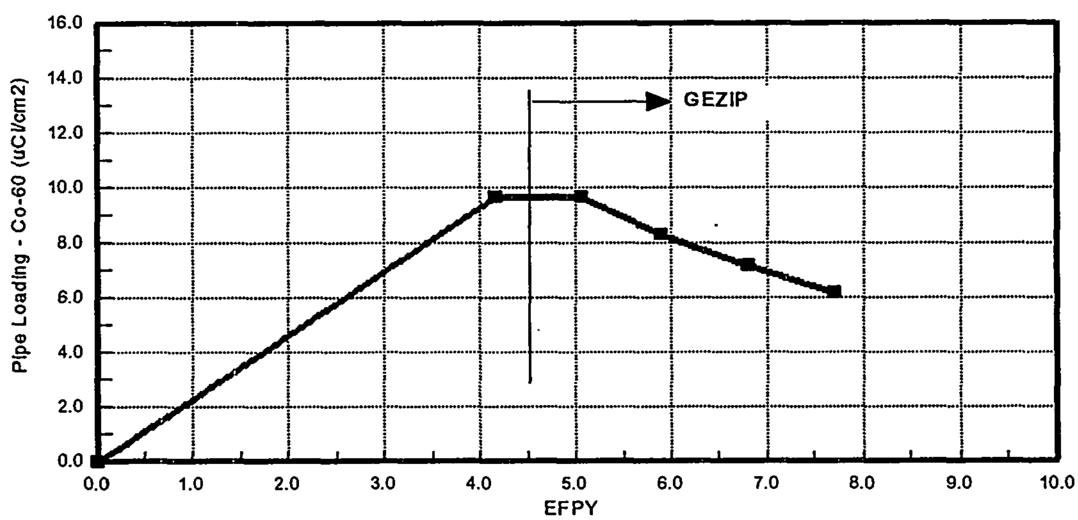

Figure 14. Decrease in Co-60 on Leibstadt piping with GEZIP.

Limerick 1 and Limerick 2

Limerick 1 and 2 were initially in the group of 'Natural Zinc' plants but they have added deep bed demineralizers to the condensate system and this has virtually eliminated the original source of zinc. They have implemented GEZIP to retain the dose reduction effect of zinc and appear to be continuing on the low dose rate track. Reactor water zinc concentrations have generally been maintained at less than 5 ppb.

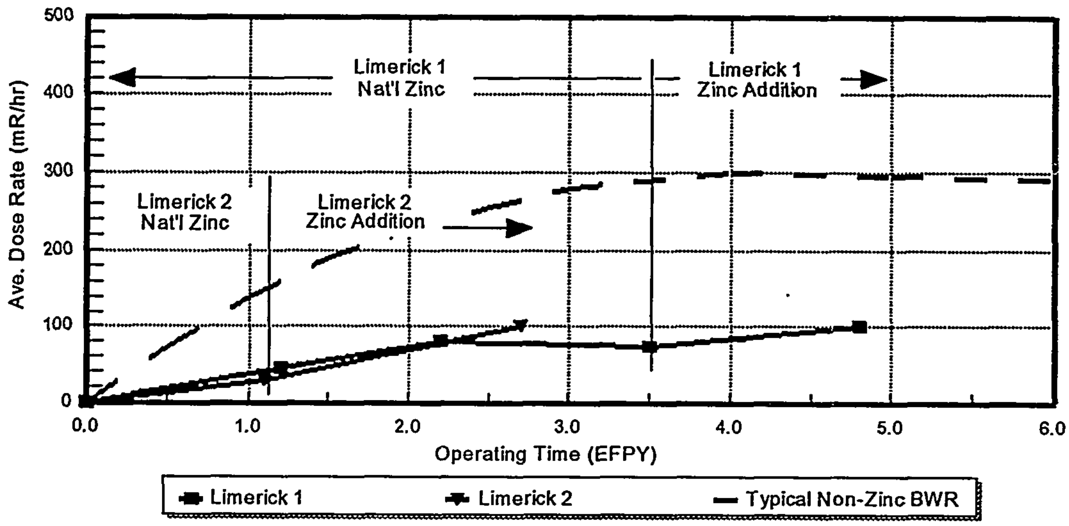

Figure 15. Radiation buildup at Limerick 1 and 2 with GEZIP.

Peach Bottom 2 and Peach Bottom 3

The Peach Bottom units were also 'Natural Zinc' plants initially but have replaced their brass condensers with titanium. To replace the lost source of zinc, Unit 2 implemented GEZIP in 1991, and Unit 3 in 1992. GE does not have subsequent dose rate measurements at this time.

Hydrogen Water Chemistry

FitzPatrick

At the end of 1988, and prior to the start of fuel cycle nine (8.5 EFPY), the FitzPatrick plant performed a chemical decontamination of the primary system as a prelude to beginning both HWC and GEZIP. They are currently in the middle of fuel cycle eleven. The radiation buildup experience for this period is shown in Figure 16.

Prior to switching to GEZIP and HWC, FitzPatrick was a typical non-zinc BWR with average dose rates at the standard locations which had peaked at approximately 300 mR/hr before stabilizing and drifting down by decay processes. The switch to HWC, for IGSCC mitigation of the primary system piping, produced insoluble transport of activated isotopes and resulted in the creation of localized hot spots of 1 to 2 R/hr. However, the general buildup in the primary system has been controlled by the zinc addition and the average at the standard locations peaked at ~120 mR/hr. The hot spot problem appears to have been a transition process and has diminished significantly in recent measurements.

Like the other GEZIP plants, FitzPatrick has elected to maintain reactor water zinc concentrations at 3 to 5 ppb, well below the recommended 10 ppb, so that the effects of Zn-65 are minimized. DZO has been used for most of the current cycle, but changeover problems have also necessitated the use of some natural zinc oxide.

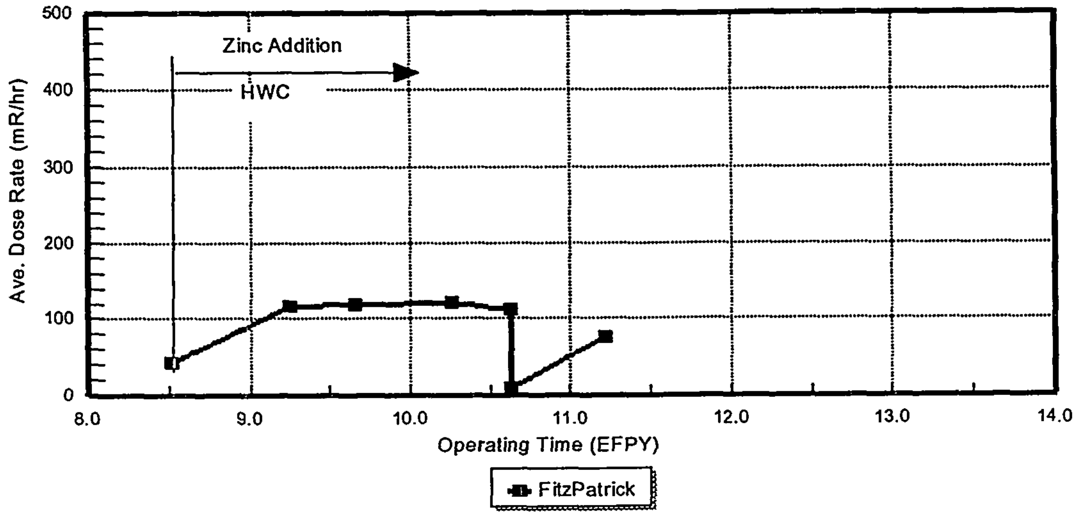

Figure 16. Radiation buildup at FitzPatrick with GEZIP and HWC.

Monticello

Monticello implemented HWC in the middle of fuel cycle thirteen, but did not start zinc addition until the beginning of fuel cycle 14. Even prior to HWC, the dose rates at Monticello had been above average. At the standard locations, the dose rates on NWC had climbed to ~400 mR/hr before the HWC switch (13.2 EFPY) and then jumped rapidly to 760 mR/hr during the last half of cycle thirteen (14.0 EFPY). During this initial HWC period, the hydrogen injection rate was limited to 15 scfm.

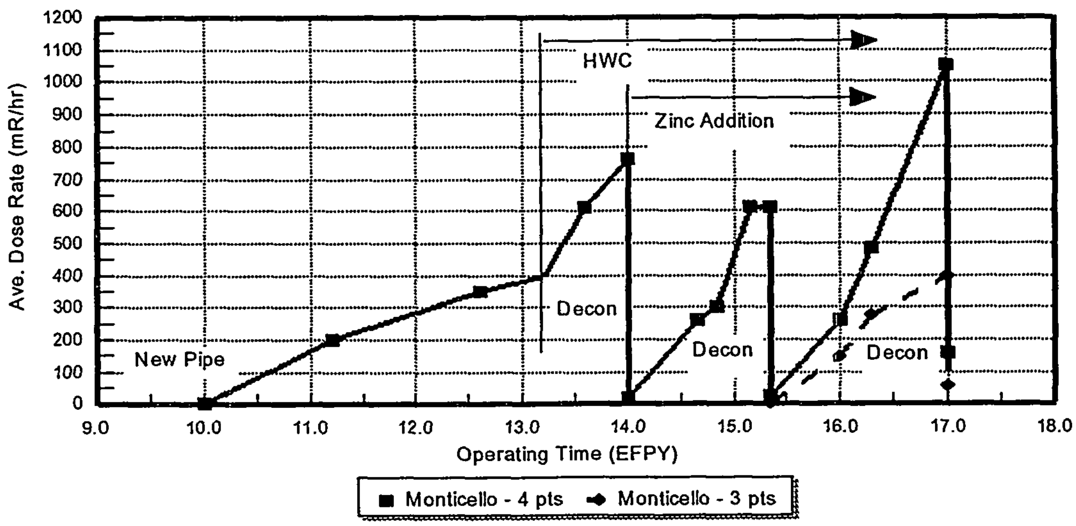

Figure 17. Radiation buildup with GEZIP and HWC at Monticello.

At the end of cycle, a successful chemical decontamination was performed to reduce dose rates prior to fuel cycle fourteen, when HWC was going to be full time at 28 scfm and GEZIP would be initiated. Over cycle fourteen, the dose rates rose to 600 mR/hr and another successful decontamination was performed at the end of the cycle (15.3 EFPY). In cycle fifteen, the hydrogen injection rate was raised to 40 scfm to protect the core internal materials. Over this cycle, dose rates for the four standard locations rose to 1050 mR/hr, driven primarily by one of the locations, which rose to 3000 mR/hr. Excluding this exceptionally high location, the three point average was 400 mR/hr. This large differential existed throughout other measurement locations in the primary system and is a strong indication of particulate deposition, rather than the typical uniform corrosion film buildup experienced in most BWRs operating under NWC. It is thought that the oxidative-reductive milling caused by cycling between NWC and HWC results in colloidal sized particulate which is significantly more transportable than normal BWR crud.

The other unusual aspect of this high buildup was that Zn-65 was the dominant isotope, contributing approximately 65% of the dose. Experience at other GEZIP plants has been that Zn-65 contributes only 10% to 40%.

The chemical decontamination employed at the end of cycle fifteen (17.0 EFPY) was not uniformly successful like the earlier ones at Monticello and the HWC/GEZIP decons at FitzPatrick and Hatch 1. While dose rates were reduced to ~55 mR/hr at three of the standard locations, the dose rate at the fourth was 155 mR/hr. Some other hot spots in the risers experienced dose reduction factors of only ~2. The reason for the non-uniform decon results, and the very poor results at some locations, is still under review. It is believed to be the result of decon process application problems, but the potential influence of high hydrogen addition rates and/or zinc addition are being examined.

Monticello is continuing to add 40 scfm of hydrogen in cycle sixteen and is now using DZO as the source of zinc. The inventory of natural zinc will cause Zn-65 to continue as a significant isotope for several cycles before dissipating.

Before GEZIP was implemented at Monticello, the reactor water Co-60 concentration was in the high range for BWRs at ~0.45 µCi/l and was the dominant isotope causing the dose rates. The presence of zinc has caused the reactor water Co-60 to decrease to <0.15 µCi/l. This is one of the significant benefits of zinc and has decreased the dose rate caused by Co-60. The unanswered question is why the Zn-65 on the pipes has been abnormally high at this site compared to other GEZIP reactors. Candidate explanations include both the high hydrogen addition rate and the extremely high hydrogen cycling frequency associated with plant maintenance work.

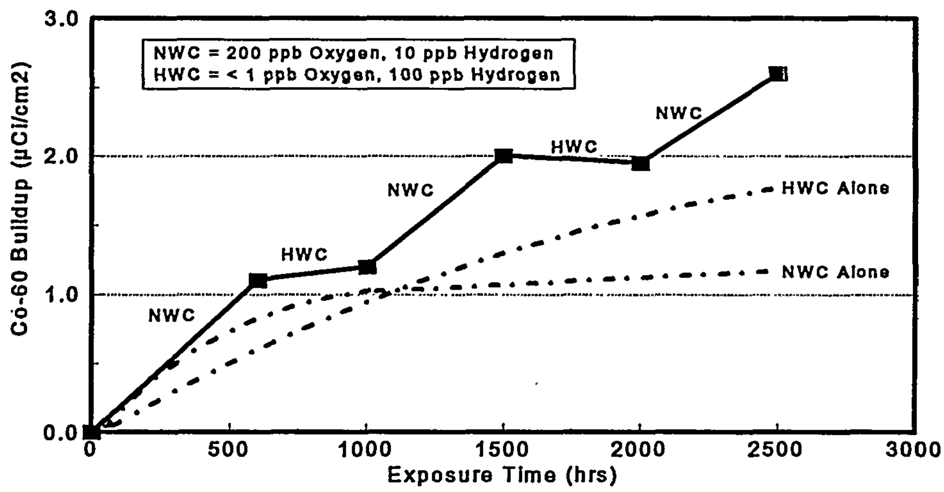

Figure 18. Laboratory data showing the effect of cycling between HWC and NWC for 304 SS.

Laboratory testing by GE has demonstrated that the cycling between HWC and NWC causes accelerated buildup rates which are significantly worse than steady operation under either chemistry[4]. HWC/NWC cycling laboratory data for 304 SS is shown in Figure 18.

Monticello is the lone GEZIP plant which has experienced high radiation buildup. The reasons and mechanisms causing this result are still ill-defined and will require continued monitoring and examination before remedies can be known with confidence.

Hatch 1 and Hatch 2

The Hatch reactors were 'Natural Zinc' initially but replaced the brass condensers and thereby lost their source of zinc. GEZIP was implemented at both units in 1990. Hatch 1 began using HWC to protect primary system piping in 1987, while Hatch 2 did not start adding hydrogen until 1991. Both units have been gradually increasing the hydrogen addition rate over 1993 to reach the level which will protect the vessel internals. At Hatch, this rate is 35 scfm.

Hatch 1 reached the 35 scfm addition rate in January, 1994 and maintains a reactor water zinc concentration of approximately 5 ppb using natural zinc oxide as a feedstock. A switch to DZO is planned for later in 1994. Figure 19 shows the radiation buildup experience at Hatch 1 from its initial operation as a 'Natural Zinc' plant, through its transitions to HWC and GEZIP. The average dose rate increased ~100 mR/hr following the onset of HWC but has returned to pre-HWC dose rates of 100 mR/hr in the most recent measurements at the beginning of 1994.

This experience is in direct contrast to the high buildup at Monticello. Even though the hydrogen addition rate has been steadily increased and is now protecting core internals, dose rate buildup is very low. Georgia power has made a concerted effort to minimize the cycling of hydrogen and maintains an availability for HWC of greater than 90%.

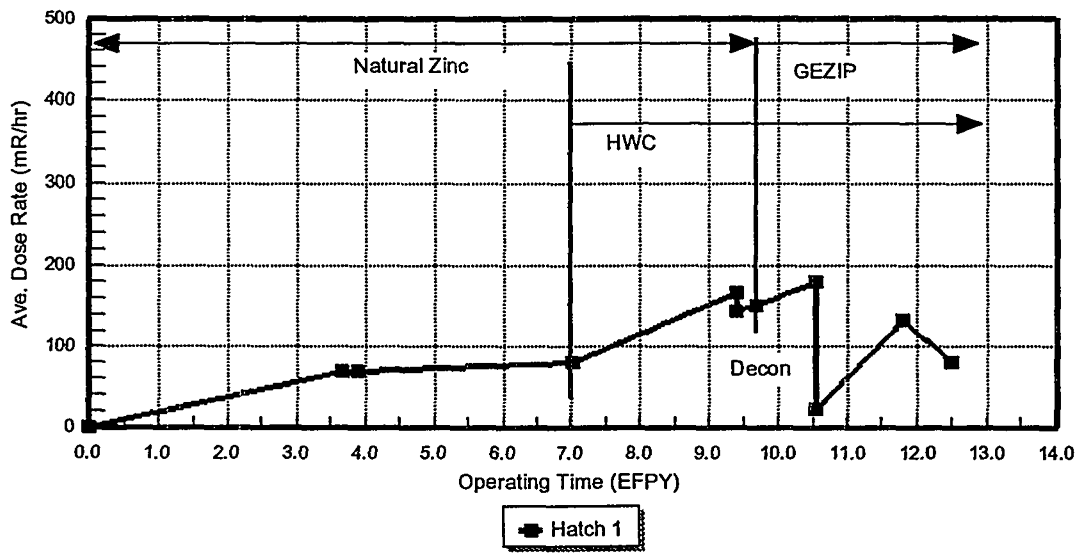

Figure 19. Radiation buildup with GEZIP and HWC at Hatch 1.

Hatch 2 is currently operating at 20 to 25 scfm of hydrogen addition while system and instrumentation adjustments are made which will permit them to operate at the 35 scfm needed for full protection. Reactor water zinc is maintained at 5 ppb using DZO as the feed material. Figure 20 shows the radiation buildup history for Unit 2.

Again, a minimal increase from an NWC dose rate of 150 mR/hr to an HWC transitional level of 210 mR/hr was observed. The most recent measurements suggest that dose rates are decreasing to the range experienced under NWC.

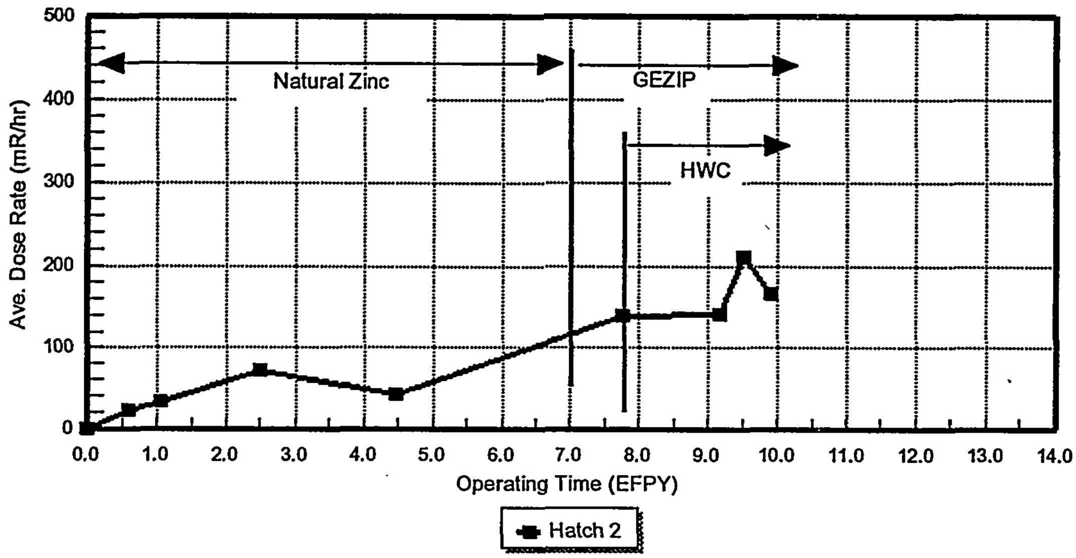

Figure 20. Radiation buildup with GEZIP and HWC at Hatch 2.

RELATED ISSUES

DZO (Depleted Zinc Oxide) and Zn-65

While it was known that the presence of zinc in the BWR primary system results in activation on the fuel surface and distribution of Zn-65 through the plant, experience in 'Natural Zinc' plants had not indicated that this was a significant problem. However, at the first refueling outage for Hope Creek it became clear that some fundamental differences existed between the 'Natural Zinc' plants and the first plant to implement GEZIP. At that refueling, Zn-65 was released from the fuel deposits to the reactor water at a high rate and resulted in a peak concentration of approximately 200 µCi/l in the reactor water and a total release of approximately 3000 Curies of Zn-65 to the radwaste system. These unexpectedly large quantities of Zn-65 caused difficulties in handling and disposal for plant personnel and made it clear that actions were needed to understand and deal with the problem that Zn-65 could cause in GEZIP plants.

After review, it was determined that the important difference between the 'Natural Zinc' plants and Hope Creek was the amount of iron entering with the feedwater. The powdered resin condensate systems in the 'Natural Zinc' plants are excellent particulate filters and typically result in less than 2 ppb iron in the final feedwater. Conversely, the deep bed demineralizers in plants such as Hope Creek are less efficient particulate filters and result in higher iron inputs. In the case of Hope Creek, the iron input averaged approximately 12 ppb over the first cycle. This iron incorporates 5% to 15%, by weight, zinc and carries this zinc to the fuel surface, where over 80% of the iron is deposited. Thus, the inventory of zinc and Zn-65 on the fuel was much higher at Hope Creek and would be expected to be at any plant with higher iron input than the typical 'Natural Zinc' plant.

Interim approaches to minimizing the problems associated with Zn-65 have been identified and communicated to those BWRs using GEZIP. These recommendations include the following:

1. Reduce feedwater iron input with a goal of 0.1 to 0.5 ppb. Optimize condensate treatment system performance with improved resins and/or addition of new filters.

2. Use 'soft' shutdown procedures contained in GE Nuclear Energy's Service Information Letter (SIL) #541. These procedures attempt to minimize the hydrodynamic turbulence, associated with shutdown, that is believed to promote isotopic release.

3. If using HWC, minimize the cycling on and off of the hydrogen.

The best resolution to the concerns about Zn-65 is to use DZO (*D*epleted *Z*inc *O*xide) as the feedstock for GEZIP. DZO has been isotopically engineered to reduce the concentration of the Zn-64 precursor of Zn-65 from the naturally occurring 48% to less than 1%. Figure 21 shows the isotopic split for both natural zinc and DZO.

The isotopic enhancement of DZO is accomplished by separation in gas centrifuges and is then converted to either zinc oxide powder or pellets as needed for the specific plant. Currently, Leibstadt, Hope Creek, Millstone, FitzPatrick, Monticello, and both Hatch units are either using or planning to use DZO. By virtue of the isotopic enrichment processing, the cost of DZO is high but it is anticipated that as experience is gained and market demand rises, the price will decrease.

For the plants mentioned, the fact that they have added natural zinc for one or more cycles means that the switch to DZO will not mean an immediate elimination of Zn-65 in the plant environment. The inventory of Zn-64/Zn-65 is expected to gradually diminish in importance over several fuel cycles. Plants using DZO from initial GEZIP operation, such as LaSalle 1, should experience no significant Zn-65 impact in the plant environment. It is strongly recommended that any BWR which implements GEZIP should plan on using DZO as the source of zinc so that the benefits are maximized.

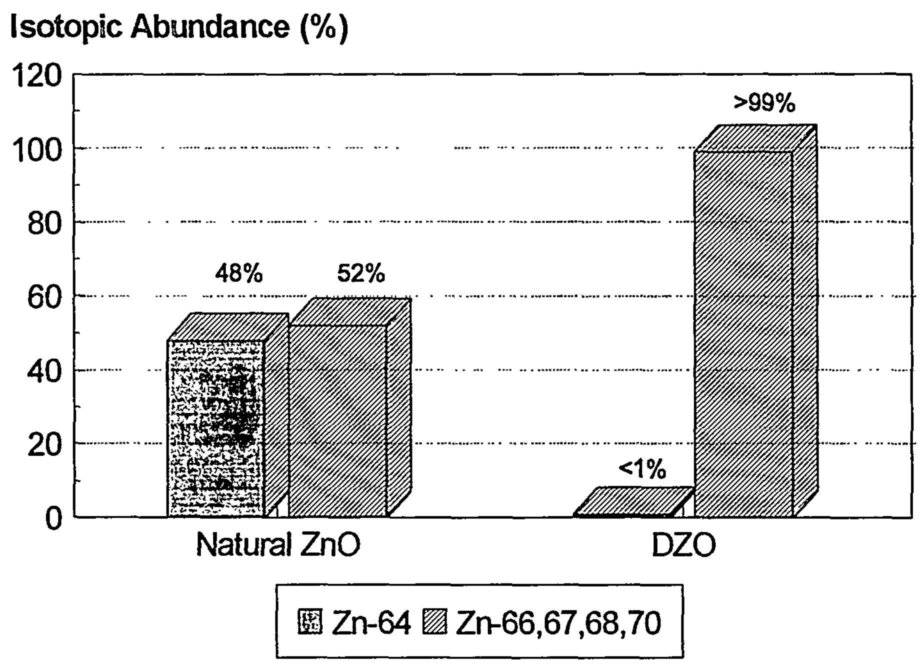

Figure 21. Isotopic concentrations for natural zinc and DZO.

IGSCC (Intergranular Stress Corrosion Cracking)

When it was verified that zinc suppresses the corrosion of stainless steel by forming an oxide film which was more protective to the base metal, it opened the possibility that zinc addition might also contribute to the suppression of IGSCC. Reviews of early BWR experience, as well as scoping tests, soon began to reinforce these additional hypothesized benefits. Two early BWRs had operated with zinc concentrations of approximately 100 ppb and reported very little IGSCC, even though some of the piping components had been fabricated from highly susceptible material. Soon controlled testing from several organizations were demonstrating the beneficial effects of zinc on IGSCC for both BWR and PWR, materials[5,6,7].

Extensive tests perfromed by General Electric, under BWR conditions, have confirmed that the presence of the zinc ion can reduce the crack growth rate of IGSCC in stainless steel and high nickel alloys, especially at the low electrochemical potentials associated with HWC. However, the beneficial effect has been found to vary from heat to heat of material. This has made it very difficult to quantify the benefit and define the role of zinc in IGSCC mitigation for the BWR.

Chemical Decontamination At GEZIP Plants

Early in the development of GEZIP, concern was expressed in the industry as to whether oxide films formed in the presence of zinc would be able to be decontaminated using the current chemical processes. In the past several years, chemical decontaminations have been conducted at Millstone Pt 1 (after two months of zinc operation), FitzPatrick, Hatch 1, Leibstadt (local), and Monticello (twice).

All but one of these decontaminations were highly successful and resulted in post-decon dose rates of approximately 10 to 20 mR/hr. The one exception is the most recent experience at Monticello. As indicated earlier, the results at this decon were non-uniform, with various locations experiencing DFs as high as 10 or as low as 2. Pending the outcome of related evaluations, it is believed that the difficulties were related to chemical process application and control.

SUMMARY

Over the last ten years, it has become possible to control dose rates in the BWR using the addition of trace quantities of ionic zinc. This technology has proceeded from the stage of hypothesis, through controlled laboratory testing, to application at fourteen BWRs. It has been applied, and been successful, in new and old plants, with and without chemical decontamination, in both non-zinc and previously 'Natural Zinc' plants,and under both NWC and HWC. This success has been attained in spite of the fact that zinc concentrations have been maintained at 2 to 5 ppb instead of the desired concentration of 10 ppb. The refinement of creating isotopically engineered zinc, DZO, offers the opportunity to eliminate Zn-65 as a concern in the application of zinc addition. Several successful decontaminations at GEZIP reactors seem to verify that the films created with zinc addition can be handled with current technology. Monticello, the lone anomaly in the fourteen plant applications for zinc addition (data reported for eleven), requires additional investigation before the mechanisms at work will be understood. However, as DZO continues to be used there, even the high dose rates at this site are expected to be greatly reduced.

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Author Biography

William J. Marble is a Principal Engineer for GE Nuclear Energy in Gilroy, California. For the past 11 years, Mr. Marble has been working on the development and implementation of zinc addition for use in controlling radiation buildup in the primary system of BWRs. Prior to his work on zinc addition, he worked in various assignments related to the development, design, and application of equipment and systems for use in the chemistry and water treatment area of the BWR. He earned a B.S. in Chemical Engineering from Cornell University in 1965.

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PAPER 1-5
DISCUSSION

Helman: What is the zinc contribution of that 80 to 100 mrem?

Marble: It varies from plant to plant how much they're putting in and how much iron they've got, but the range would be something like 15-30% or 40%, and in one case it was as high as 60%.

Helman: What do you expect the estimated annual cost of using zinc will be in the future, assuming that you get the price down as you are anticipating?

Marble: The current range would be something between $200,000 per year and $1 million per year, depending on the size of plant that you have. We think that we can get the price down to hopefully about half of where we are now. Depending on the size of your plant, that would be between $100,000-$500,000. It depends very significantly on the amount of iron you have in your feedwater. If you can also get the feedwater iron down, then you can significantly and dramatically drop your zinc costs.

REDUCTION OF RADIATION EXPOSURE IN JAPANESE BWR NUCLEAR POWER PLANTS

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INTRODUCTION

The reduction of occupational exposure to radiation during the annual inspection and maintenance outages of Japanese boiling water reactors (BWR) is one of the most important objectives for stable and reliable operation.

It was shown that this radiation exposure is caused by radionuclides, such as Co-60, Co-58 and Mn-54 which are produced from the metal elements Co, Ni, and Fe present in the corrosion products of structural materials that had been irradiated by neutrons. Therefore, to reduce radiation sources and exposures in Japanese BWRs, attempts have been reinforced to remove corrosion products and activated corrosion products from the primary coolant system. This paper describes the progress of the application of these measures to Japanese BWRs.

Most Japanese BWR-4 and BWR-5 type nuclear power plants started their commercial operations during the 1970s. With the elapse of time during operations, a problem came to the forefront, namely that occupational radiation exposure during plant outages gradually increased, which obstructed the smooth running of inspections and maintenance work. To overcome this problem, extensive studies to derive effective countermeasures for radiation exposure reduction were undertaken, based on the evaluation of the plants' operation data.

In particular, the Improvement and Standardization Program to establish Japanese Light Water Reactors, which aimed at improving plant reliability and availability and at reducing occupational radiation exposure, was established after 1975 under cooperative efforts by the Japanese electric power companies and plant manufacturers.

Following this program, a series of countermeasures also were applied to the older BWRs which had started their commercial operations before the Improvement and Standardization Program was adopted.

Historically speaking, all the Japanese BWRs of the initial generation, the so-called older BWRs, gave unexpectedly high occupational radiation exposures that increased annually during their refueling and maintenance outages. After several years of effort to clarify this problem, it became clear that the occupational exposure was determined by radioactive corrosion products, particularly Co-60 and Mn-54 deposited on major components and pipings of the primary coolant system, and that the following factors have a big influence on the buildup of radionuclide and, consequently, on occupational radiation exposure:[1]

1. input of iron crud into the reactor water from the feedwater (concentrations of Fe crud in the feedwater),
2. cobalt contents in structural materials, especially in-core materials,
3. capacity of the reactor's water cleanup system,
4. quality of the reactor water, and
5. condition of the inner surface of pipings and components.

Based on these findings, a wide variety of countermeasures were developed and proposed to reduce and control the radiation dose level around the primary coolant circuits (Figure 1). A detailed breakdown of these measures is shown in Table 1; most of them already had been adopted by both older and newer plants, achieving excellent BWRs with very low occupational radiation exposures.

Now, during this decade the water quality and radionuclide concentrations in the reactor water of all the Japanese BWRs changed greatly, and there are new problems which have to be solved to further reduce radiation exposure.

In this paper, I discuss some of the typical measures which were proven to be effective for reducing radiation dose rate in older and new Japanese BWRs from the standpoint of the improvements in materials, systems, and operations, and the results from adopting these measures. Some of the new items being studied in Japan also will be presented.

MATERIAL IMPROVEMENT

Sampling Line Material

In advancing the studies on the behavior of corrosion products, it is very important and fundamental to obtain accurate data on water chemistry for the operating plants. It was found that the conventional stainless-steel sampling line for sampling water at high temperatures gave incorrect values for the concentration of corrosion products, especially for Co and Ni due to the contamination of corrosion products released from materials of the sampling line.

Titanium (Ti) was found to be the most appropriate material for the high-temperature sampling; consequently, conventional sampling line tubings and valves for the final feedwater sample made of stainless steel and Stellite were replaced by titanium ones in most Japanese BWRs.

Reduction of Iron Crud

Most of the iron (Fe) crud fed into the reactor comes from feedwater. Its origin in the final feedwater is the crud which was generated upstream of the main condenser and leaked through the condensate treatment system. Iron crud also comes from corrosion products generated from the components of the feedwater system.

Low alloy steels, STPA-23 and A387Gr.11 (1.25 Cr-0.5 Mo steel), were used to reduce corrosion-erosion in extraction steam pipings and their drain pipings. These materials were selected from laboratory loop tests.[2] Newer BWR plants also chose these alloys for the moisture separator and its drain pipings. Even in older plants, the same materials were adopted, in part, for the equipment and pipings of the condensate and heater drain systems.

For the material of main condenser, a special carbon steel which contains small amounts of chromium and copper, SMA-41 (0.3 Cr-0.3 Cu steel), was used in place of plain carbon steel after considering the results of inplant corrosion tests. Figure 2 clearly shows the effect of material replacement on reducing the concentration of Fe crud in condensate water is clearly seen.

Table 1. Improvements in Radiation Dose Reduction in Japanese BWRs

MATERIAL IMPROVEMENT	IRON CRUD REDUCTION	Low Alloy steel Corrosion Resistant Material
	COBALT REDUCTION	Co Free Alloy -Pin/Roller -Hard-facing Low Co Stainless Steel -Feedwater Heater Tubing -Reactor Internals Low Co Inconel -Fuel Springs
	TITANIUM	Sampling system Condenser Tubing
	PRETREATMENT OF MATERIAL	Electropolishing High Temperature Air Oxidation
SYSTEM IMPROVEMENT	WATER TREATMENT SYSTEM	Condensate Demineralizer -New Resin -Non-regenaration Operation Prefilter -Powdered Resin Type -Hollow Fiber Filter Capacity of RWCU (2%)
	SHIELDING IN DRYWELL	PLR Pipings RWCU Pipings
	MINIMIZATION OF RWCU RADIATION SOURCES	RWCU Pump Relocation Shortening of RWCU Pipes
OPERATIONAL IMPROVEMENT	OXYGEN CONTROL Ni/Fe RATIO CONTROL OPERATION IMPROVED SHUTDOWN OPERATION	
	INTEGRATED LAYUP PRACTICLES	Condenser/Hotwell Cleaning Before Restartup Flushing

Table 2. Countermeasures to Improve CF Performance

Countermeasures \ Plant	2F-1	2F-2	2F-3	2F-4	K1	K2	K5
(1) Cation Exchange Fiber	●	●	●	●	●	●	●
(2) Swelling Operation	●	●	×	●	×	×	●
(3) Flow Pattern Improvement (Draft Tube)	●	×	×	×	×	×	●
(4) Body-Feed Operation	×	×	×	×	×	×	×
(5) Low Cross Linkage Cation Resin	×	×	×	×	×	×	×

● Applied
O Under Consideration
× Not Applied

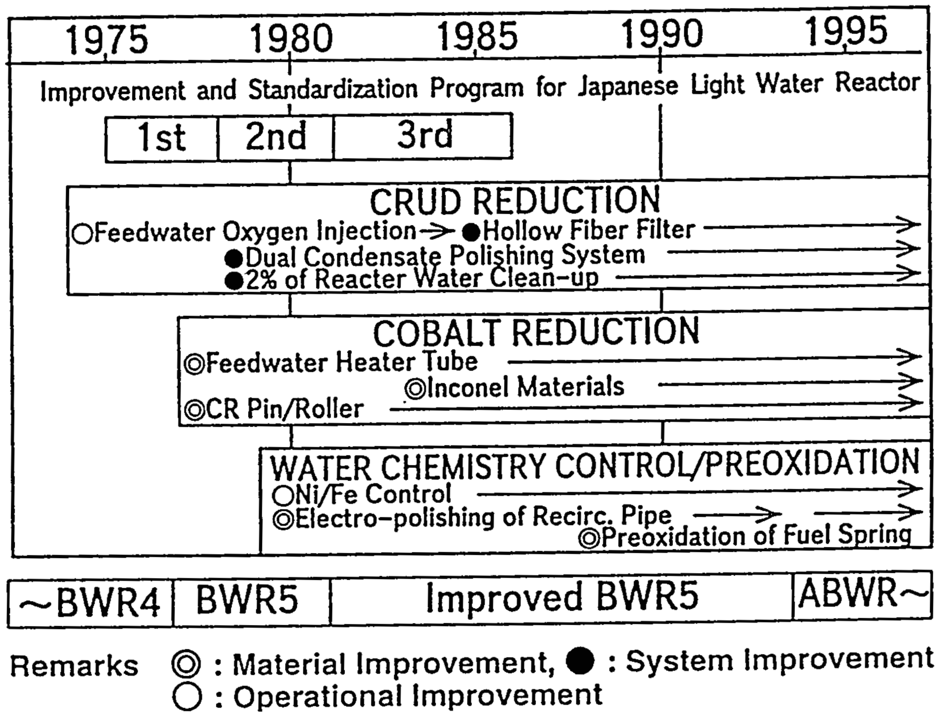

Figure 1. Reduction of radiation exposures in Japanese BWRs

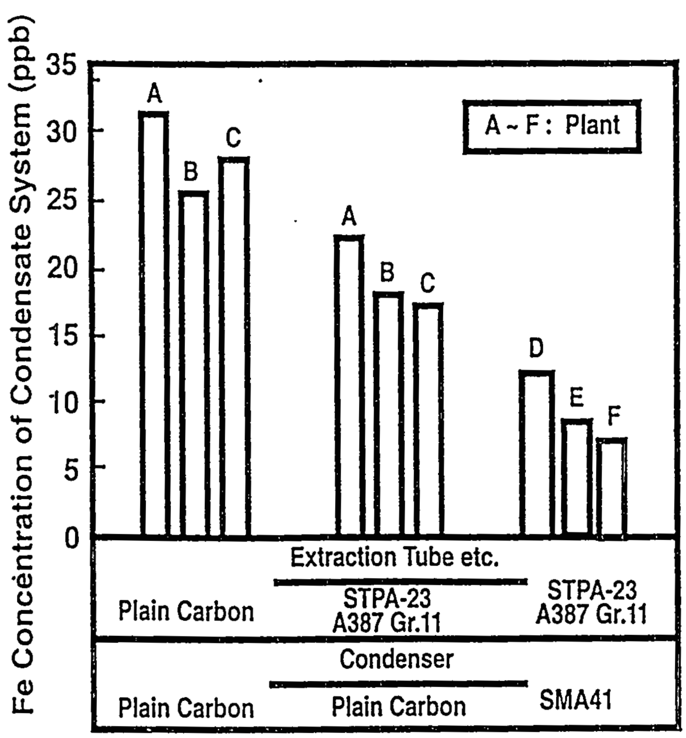

Figure 2. Change of Fe concentration upstream of condensate system by material replacement

Reduction of Cobalt

Cobalt is released into the primary cooling water from various parts of the primary system. Generally, cobalt is contained in stainless steel and in nickel-based alloys as an impurity of the nickel component, and also in cobalt base alloys. Therefore, cobalt can be reduced by using low cobalt materials in place of conventional structural materials. The materials to be replaced with low cobalt materials should be selected after an evaluation of the contribution of each material to Co-60 concentration in reactor water.

From this point of view, the first materials to be replaced by low cobalt materials should be 1) materials with a large surface area in contact with the high temperature primary water, such as stainless steel tubings of the feedwater heater system, 2) materials with a high cobalt content, such as Stellite, and 3) in-core materials which receive high neutron irradiation, such as springs made of Inconel in fuel bundles.

For example, the material used for pins and rollers on the control rods of conventional plants were made of the cobalt-base alloy, Stellite. Although the surface area of these parts is small, it meets the two of these conditions of high cobalt content and in-core materials, and so should be replaced with low cobalt alloys. All recent Japanese control rod pins and rollers are made of a newly developed cobalt-free alloy[3] instead of the cobalt base alloy, Stellite.

Inconel springs in the spacers of fuel bundles were evaluated as the biggest sources of Co-60 and Co-58 in reactor water because ordinary Inconel contains approximately 0.2% of element cobalt and they always receive high levels of irradiation from thermal and fast neutrons. Consequently, all Japanese plants recently changed to Inconel springs of low cobalt content (<0.05%).

Low cobalt stainless steel in which the cobalt content is less than 0.05% was recently used for feedwater heater tubings in newer plants instead of unspecified ordinary stainless steel. In the newer plants, some of the reactor's internal equipment also was replaced by this low cobalt stainless steel.

Figure 3 shows how the extent of the influence of each component as a source of Co-60 has changed, from conventional plants with low cobalt materials to newer plants that have incorporated them to a great extent.[4] Assessments showed that Co-60 sources were reduced to about one fourth by the adoption of low cobalt and cobalt free materials.

From this evaluation, it was found that the main sources of Co-60 consist of those generated from Stellite in valves. Therefore, it is most important that new, cobalt-free alloys are developed for the hard-facing materials of valves, especially for large bore valves.

Titanium Condenser

As shown in Table 1, titanium was adopted for the tubing material in main condenser of newer plants to improve the efficiency of the turbine heat exchanger and the quality of condensate water.

The feedwater of the plants which adopted a Ti condenser, in addition to using a system called a dual condensate treatment system, showed quite excellent water quality, approximately 0.06 μS/cm. Consequently, non-chemical regeneration operation of the condensate demineralizer was unnecessary for more than five years.

One of the desirable influences of this improvement was that the generation of secondary radwaste was reduced to a large extent. Another one, the most important one from the viewpoint of control of radiation buildup, was that the impurities in reactor water were steadily maintained at very low level, around 0.1 μS/cm. This suggests that the buildup rate of radioactivities on out-of-core pipings and equipment is controlled at desirable levels.

Pretreatment of Material Surface

Pretreating the material surface to reduce buildup of activity by controlling the characteristics of its oxide film is considered as one effective measure among other remedies. Factors which contribute to the rate of deposition of activity by pretreating materials are the smoothness of the material's surface, protectiveness of the oxide film, thickness of the oxide film, and so forth.

One pretreatment method, prefilming by dissolved oxygen, was employed in several plants. The first experience of prefilming for the recirculation pipings was carried out at Kashiwazaki-Kariwa No. 1 unit (K1). In K1, the prefilming was carried out before the commissioning test. The concentration of oxygen in the reactor water was kept about 300 ppb by dosing oxygen gas into the water of the control rod drive system. The heat source was Joule heat from running the recirculation pump. Radiation suppression rate was about 15% determined by the coupon test during the first cycle operation of K1.

Many different kinds of pretreatment methods have been developed worldwide. Figure 4 shows an example of the results of those studies. Two sorts of coupons, one group of which consisted of as-received coupons relative to ordinary recirculation piping, and other coupons which were mechanically and electrolytically polished, were exposed in the reactor water of four operating plants. The figure shows that electropolishing the recirculation pipes could control the radioactivity buildup to less than one half in comparison with unpolished ones.

Our recent efforts focused on developing techniques to reduce the dissolution of ionic radioactivities from in-core materials, particularly Inconel springs in the fuel spacers. After many kinds of pretreatment studies, including oxidation by chemical treatment, high-temperature oxidation in air was chosen as the most realistic technique from laboratory tests. Figure 5 shows the trends in metal release from a coupon oxidized by conventional treatment and a coupon in high-temperature air when tested in BWR water. The rate of release of the metal was very small for the latter. After this test, the distributions of metal oxide in the oxide layers were determined, as shown in Figure 6. It seems that the excellent protective properties of the air-oxidized coupon against corrosion came from the high contents of nickel ferrite and chromium oxide in the oxide film.[5]

SYSTEM IMPROVEMENT

Various kinds of improvements to systems to reduce radiation exposure during each maintenance and inspection outage were proposed from the viewpoints of both reducing radiation dose-rate and saving working time around the radiation fields. Table 1 shows some improvements that have been applied to the older plants, newer plants, will be used in future plants.

Improvements in the Water Treatment System

Generally, the efficiency of crud removal by a mixed-bed type, cation/anion ion-exchange beads resin in a condensate polishing system were thought to be insufficient to reduce Fe crud concentration in feedwater below 1 ppb. Through observations of water chemistry during condensate demineralizer operations, it was found that cation resin, especially aged cation resin, removed not only crystallized iron oxide but even less well crystallized iron compounds generated from the material surfaces upstream of the condensate system, as shown in Figure 7.[6,7]

Based on studies of the properties of aged cation resins, several new cation resins were developed and tested in actual condensate water, one of which showed an excellent ability for crud removal (Figure 8). At present, the mechanical properties of other improved cation resins are being tested, using condensate water from an operating plant.

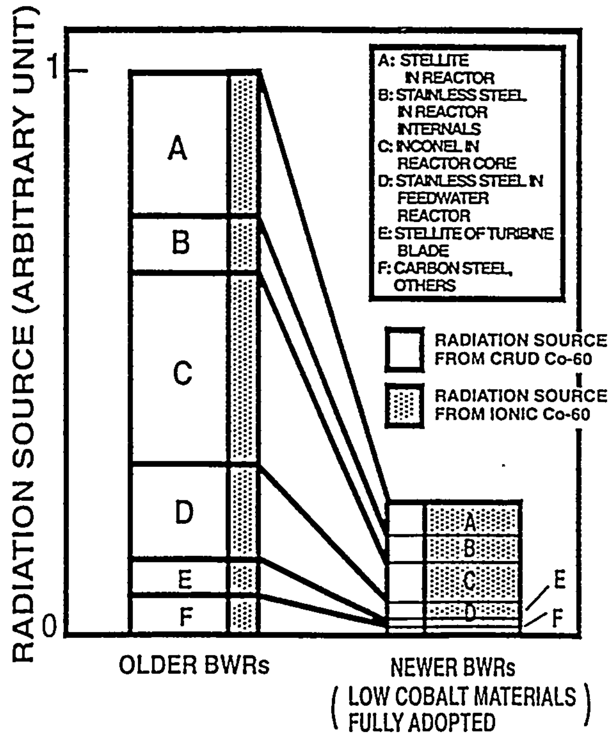

Figure 3. Evaluation of effect of using low cobalt material on the generation of Co-60 radiation

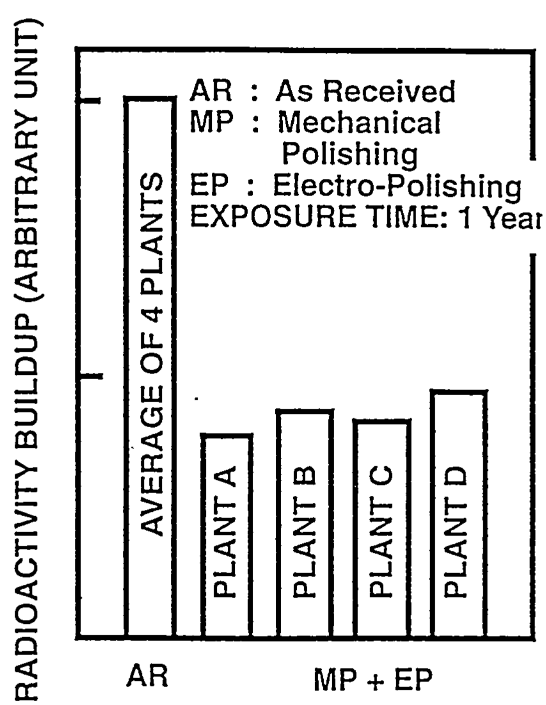

Figure 4. Effect of mechanical polishing and electropolishing of stainless steel on radioactivity buildup for one year exposure in reactor water of operating BWRs

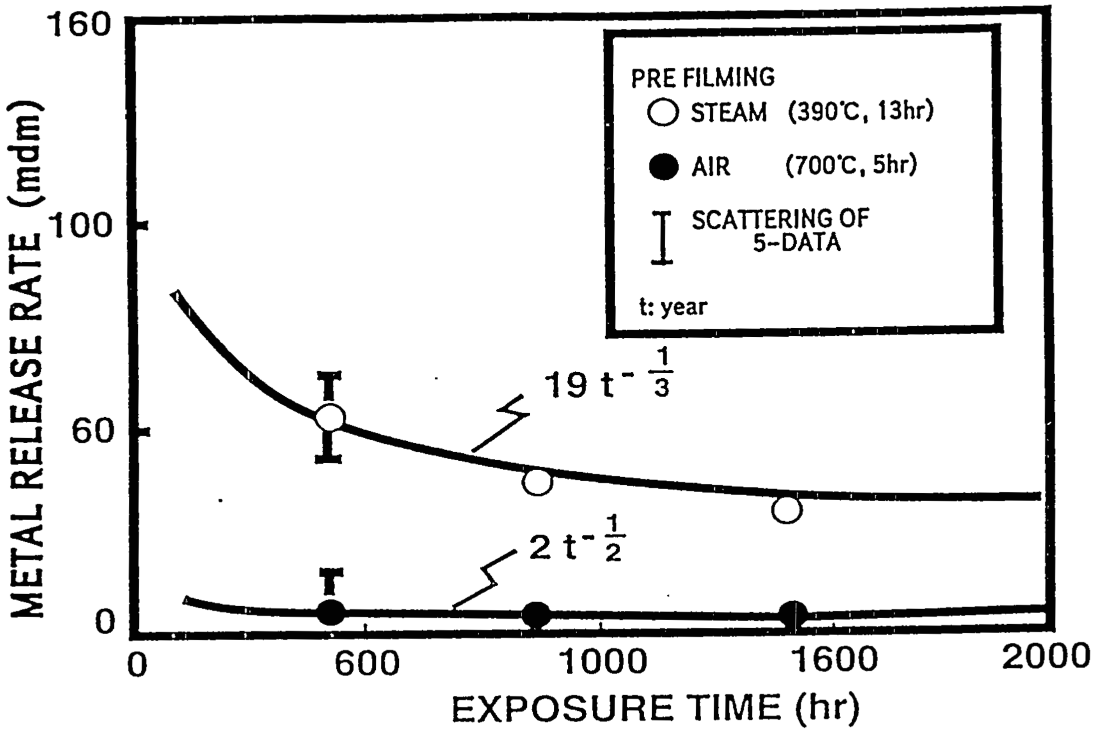

Figure 5. Metal release rate from pre-filmed Inconel X750 in high temperature water loop

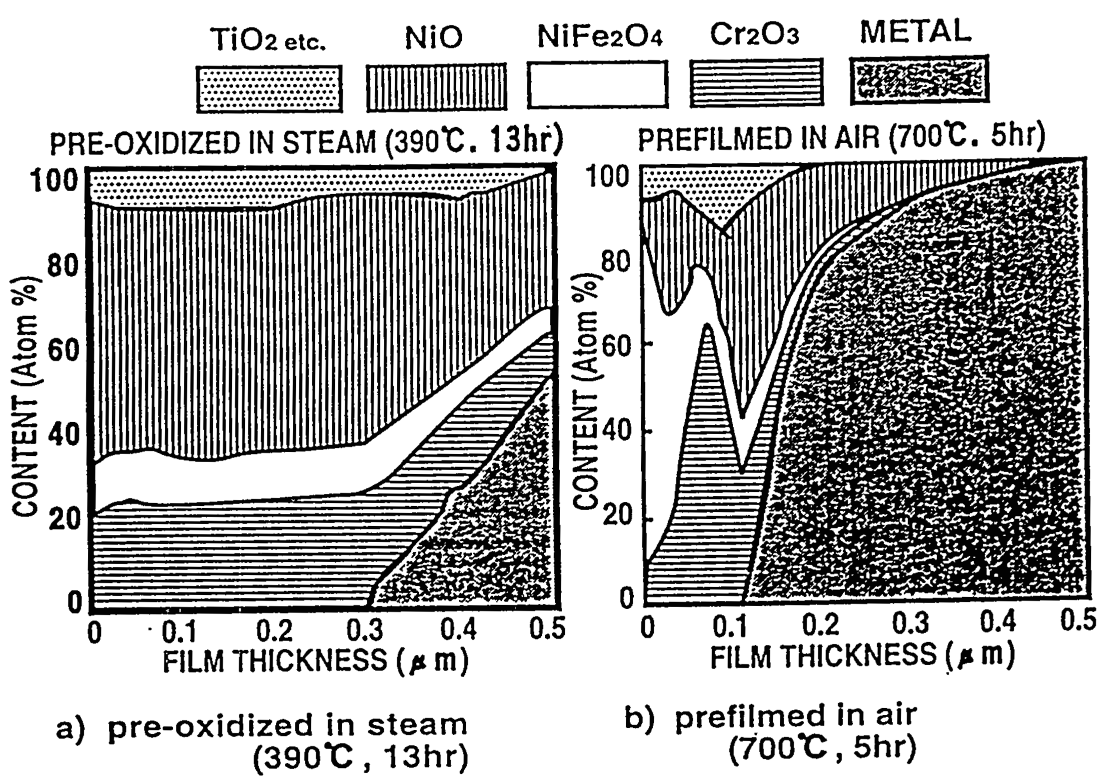

Figure 6. Distribution of corrosion oxides formed on Inconel X750 by two different oxidation methods

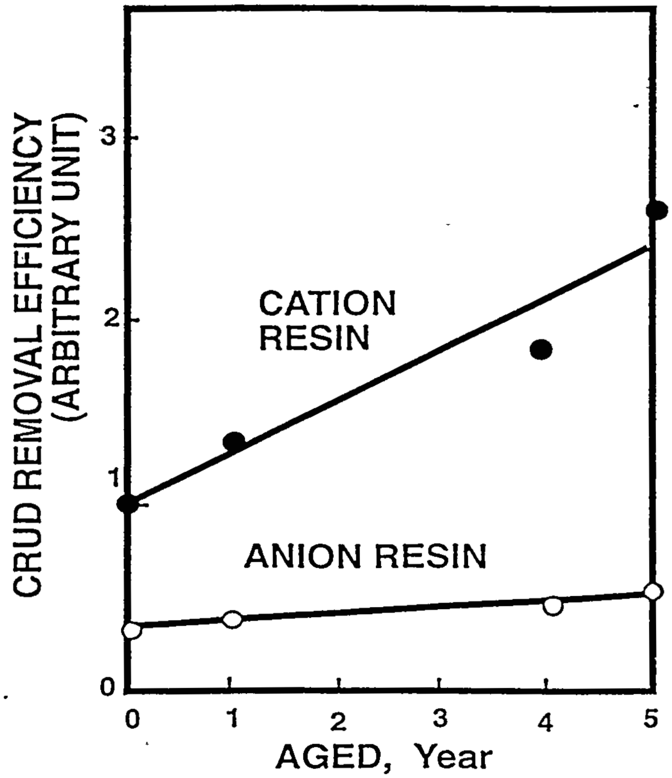

Figure 7. Improvement of efficiency of crud removal by aging of condensate demineralizer resins

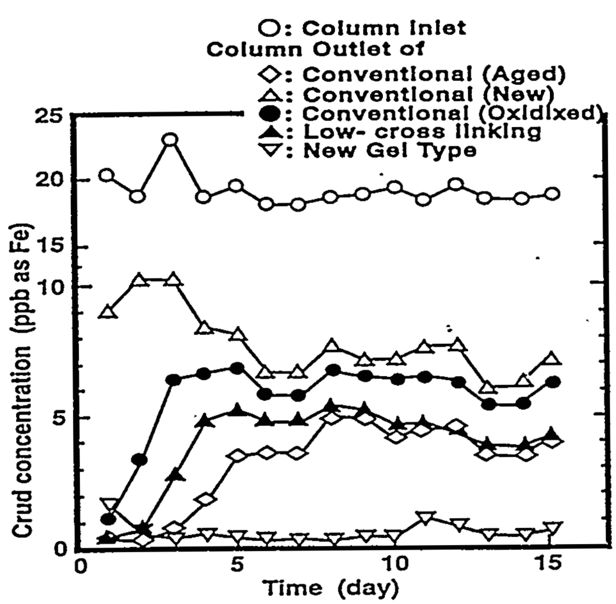

Figure 8. Breakthrough of crud in condensate water f rom various demineralizer resins (in-plant column test)

Most Japanese 1100MWe-class BWRs employ a dual condensate treatment system to reduce Fe crud input into the reactor. The concept of dual condensate treatment system is a purification system consisting of a prefilter to remove Fe crud in condensate water and a demineralizer to eliminate impurities during occasional sea-water intrusion. The prefilters used before the condensate demineralizer are classified into two types, powdered resin-type water polishers, and hollow-fiber filter systems (HFF).

However, some technical problems with powdered resin-type filters were revealed, including the efficiency of crud removal and shortening of run length. Typical decontamination factors and run lengths of powdered resin type filter ranged between 3 to 7 ppb, and 10 to 25 days, respectively. To solve these problems, several countermeasures were proposed and tested. Recently, some countermeasures, shown in Table 2, were proved effective by in-plant tests at operating plants; in the best case a very long run length of about 200 days was achieved (Figure 9).

The other improvement was the adoption of newly developed Hollow Fiber Filter (HFF) system upstream of the deep bed demineralizer. This system was first applied to a radwaste treatment system, and then to the conventional plants to save outage time by shortening the re-startup cleaning time of condensate and feedwater. For the latter purpose, a HFF having the capacity of 30% feedwater flow was first applied to Fukushima Daiichi Nuclear Power Station No. 3 Unit (1F-3); after operating successfully during restartup of the plant, it was put in-service to purify the condensate water during steady state power operation.

The efficiency of crud removal was measured by the iron species contained in the condensate water. A result, shown in Figure 10, gives measurements for a precoat-type filter. HFF had an excellent performance, even for amorphous iron species; Figure 11 shows the efficiency of Fe crud removal measured for condensate water of 1F-3 plant. Hollow fibers used in the 1F-3 plant initially were hydrophobic; later ones were improved and had hydrophobic properties, tending to have a longer module life and being easier to handle. It was proved by in-plant tests that the hydrophilic hollow fiber had smaller increase in differential pressure compared to the hydrophobic one. This hydrophilic filter module has been used in 1F-3 in place of a hydrophobic module after four successful years in operation.

Now, the HFF system has been adopted by many Japanese BWRs, and replacement of powdered-resin type condensate polishers by HFF system is also discussed because of its better crud removal performance and its substantial reduction of secondary radwaste generation.

Shielding Radiation Sources in the Drywell

To attain a low dose-rate for the working areas in the drywell during maintenance and inspection outages, shieldings were applied to the main radiation sources from the recirculation and reactor water cleanup (RWCU) pipings of conventional plants and new plants. Figure 12 shows an example of the savings in radiation exposure by using shieldings for the radiation-contaminated pipings in drywell. This has become one of routine remedies for reducing radiation dose-rate in Japanese BWRs.

Minimization of Radiation Dose Rate in the RWCU System

The reactor water cleanup (RWCU) system has been manufactured with thin- walled pipes, adequate to withstand the inner pressure, but not thick enough to function as a shield against radiation deposited on its inner surface area.

Some improvements for reducing radiation dose rate reduction in this system were carried out in the newer plants. The RWCU pumps were relocated from a high-temperature area to a low-temperature area downstream of the nonregenerative heat exchangers; the temperature dependency of activity deposition on carbon steel (used in Japanese BWRs).

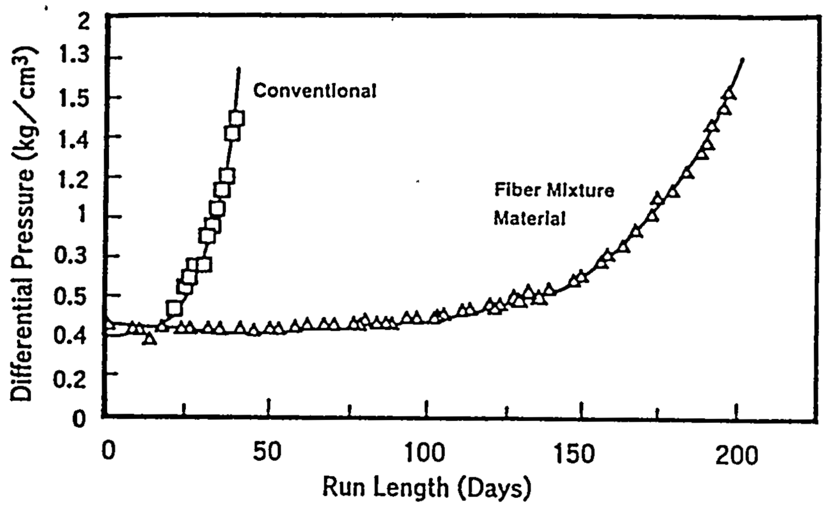

Figure 9. An example of improvement in condensate filter performance (K2 plant)

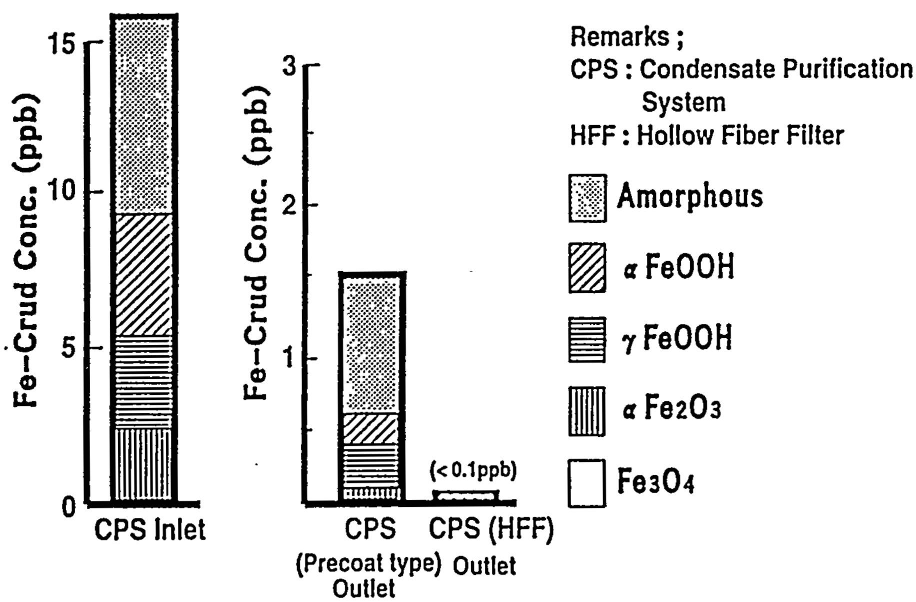

Figure 10. Removal of iron crud in condensate water by condensate filters

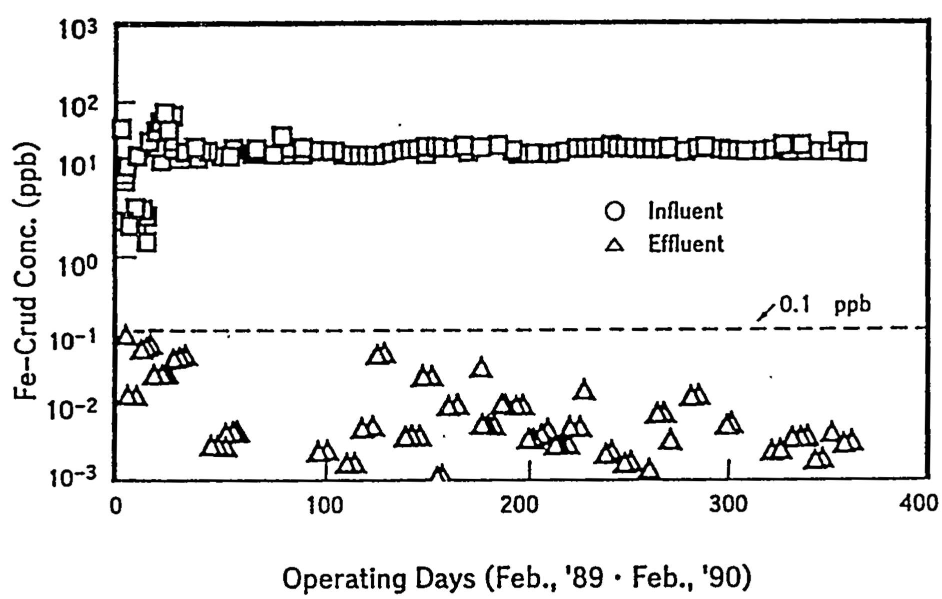

Figure 11. Removal of iron crud by HFF

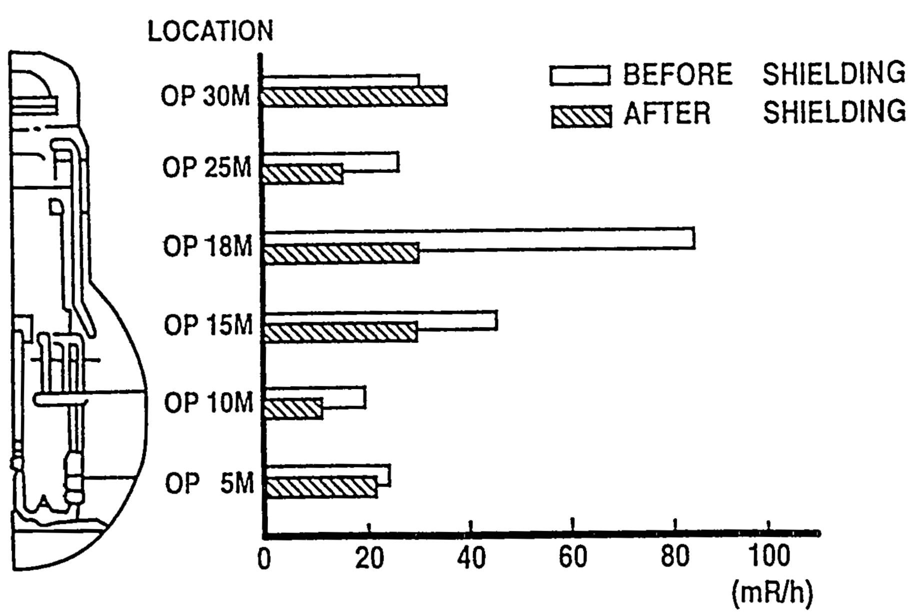

Figure 12. Reduction of radiation dose rate by shielding

The second improvement of this system was shortening the pipe length by rearranging the piping routes in the drywell. The third improvement was shielding part of the RWCU pipings in the drywell.

OPERATIONAL IMPROVEMENT

Ni/Fe Ratio Control Operation

In newer Japanese BWR plants, it has become easy to decrease the input of Fe crud into the reactor water by using the dual condensate polisher. Therefore, at first it was thought that it would be easy to keep crud radioactivities such as Mn-54, Fe-59, Co-58, and Co-60 at low levels.

However, in one plant, an adverse phenomenon on the radioactivities in reactor water was observed; iconic Co-60 and Co-58 in the reactor water in the presence of a small amount of Fe crud was strongly effected by the ratio of Ni/Fe input. That is, to suppress ionic Co-60 and Co-58 concentrations in reactor water, a little more than two times the amount of Fe crud compared to nickel should be fed into the reactor water. The results of several experiences in controlling the Ni/Fe concentration ratio are shown in Figure 13.

A key process may be the reaction between Ni and Fe on the fuels' surface:

$$Ni^{2} + Fe_2O_3 + H_2O = NiFe_2O_4 + 2H^+$$

Iconic cobalt also reacts with Fe, forming cobalt ferrite. Precipitation of many kinds of iron oxides with ionic Ni, Co-58, and Co-60 were carried out in laboratory experiments in the presence of trace amounts of Ni and Co to simulate the reactor water environment. Iron oxides which were not highly crystallized reacted faster with ionic species to form Ni ferrite and Co ferrite compared with crystallized iron oxides, such as hematite and geothite.[8]

Taking these findings into consideration, measures to control and lower iron concentrations in the feedwater were applied to crud chemistry in newer BWR plants. These measures included iron crud dosing with a partially bypassed flow-condensate prefilter, and an Fe crud dosing system installed at the feedwater system. The optimum Fe concentration in feedwater appear to range from 0.2 to 0.5 ppb. Crud control in newer BWR plants is operated carefully, with this target of Fe crud concentration in the feedwater, so keeping radiation levels low.

Suppression of Radionuclides during Shutdown Operation

It is well known that radioactive crud concentration in the reactor water becomes higher during shutdown operations than during normal power operations, sometimes increasing by more than tenfold. This phenomenon is thought to be caused by the release of part of the fuel crud into the reactor water; this crud has a higher specific radioactivity than that of crud in the reactor water during normal power operation.

Some shutdown procedures to minimize the above increase were studied in several Japanese BWRs, and the following procedures were effective and realistic; the cooling rate of the reactor water was reduced to less than 15°C/h, and the reactor pressure was held constant for 3 to 4 hours at 50kg/cm^2. The results showed that the modified shutdown procedures suppressed the maximum radioactivity concentration in the reactor water by one to two orders of magnitude compared to that from conventional practices.

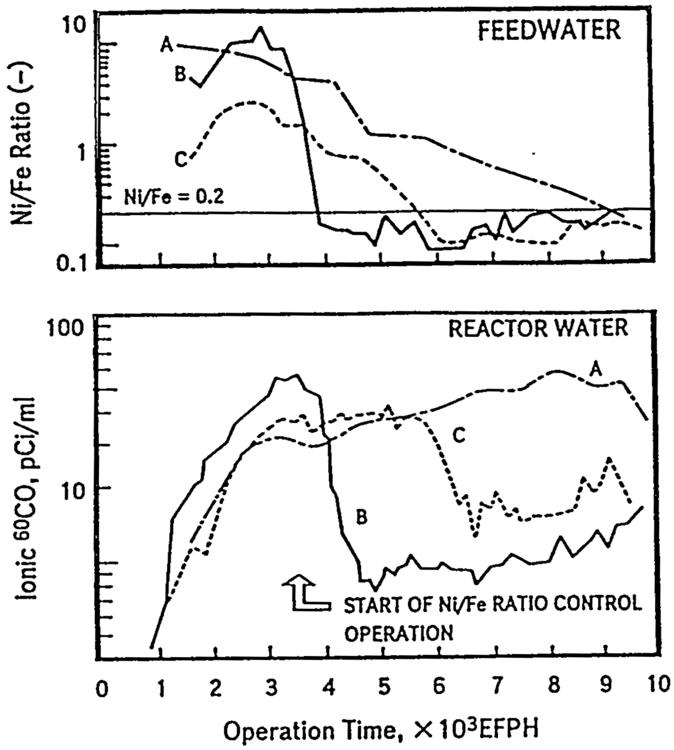

Figure 13. Control of Ni/Fe concentration ratio in feedwater

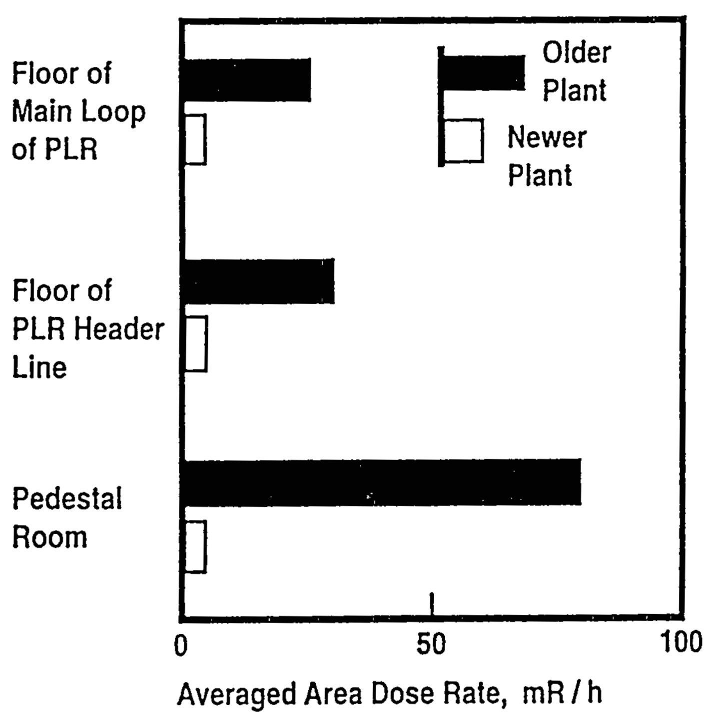

Figure 14. Comparison of averaged area dose rate around primary circuit during maintenance and inspection works between older and newer plants

Integrated Layup Practices

The adoption of integrated layup practices is thought to significantly reduce the amount of corrosion products in feedwater. Without layup practices, high crud loadings were observed on the condensate demineralizer when the plant was started up after refueling outages, particularly in older BWRs.

It was expected that cleaning the condenser and hotwell could remove considerable quantities of crud and prevent its input to the reactor vessel at conventional plants which used only a condensate demineralizer. From this point of view, as mentioned before, HFF was adopted by conventional plants.

If a HFF system having 30% capacity of the feedwater flow was installed upstream of the condensate demineralizer, the cleaning time necessary to obtain the target value of 200 ppb in the final feedwater was only two days compared in two weeks with only a condensate demineralizer at the same plant. In the case of the dual condensate polisher system, only prefilters were put into service for the purpose of pre-restartup flushing of feedwater and condensate water.

EFFECT OF RADIATION CONTROL MEASURES

All measures developed so far to reduce radiation dose rate, some important ones of which were discussed in the preceding sections, were adopted by the newer Japanese BWRs from their initial designing stage. Many of them also were applied to older, conventional plants. As the result of these measures, the general area dose rates in the drywell of newer plants are kept comparatively low, as a typical example shows in Figure 14.

This low dose rate, combined with efforts to reduce exposure time, have resulted in a record of low radiation exposure. Older, conventional plants also show noticeable decreasing trends in radiation exposure after adopting many of these measures.[9] Figures 15 and 16 give typical examples of radiation exposures of both older and newer BWR plants.

FURTHER REDUCTION IN RADIATION EXPOSURE

Older Plants

Although the radiation dose rate and, hence, the radiation exposure in older plants has decreased by adopting many of the countermeasures developed so far, it is not yet satisfactory. To find further remedies, extensive measurements to determine the amounts and morphologies of radionuclides deposited on the inner surfaces of various pipings and equipments were carried out in a typical old plant. The measurements revealed unexpectedly large amounts of radioactive soft crud or slightly adhered insoluble crud still existing, even on vertical pipings. Figure 17 shows the result of calculating the relation between radiation exposure and radiation sources in the drywell;[10] about 30% of the radiation exposure comes from the soft crud in the drywell, which means that a considerable amount of radiation exposure could be easily reduced by mechanical cleaning such as water-jet flushing.[11]

Newer Plants

Although the occupational radiation exposures of newer plants were very low at the first refueling and maintenance outages (Figure 18), they have shown a gradual increase year by year. It seems clear that this increase comes from Co-60 buildup on the material surface of the primary circuit (Figure 18). Two remedies are considered to mitigate this buildup; one is to further reduce Co and Co-60 sources by measures such as replacing the Stellite in valves, and using air-prefilmed Inconel for fuel spacer springs as described before, and the other is to develop methods to mitigate the buildup rate of Co-60. For this purpose, a more detailed and

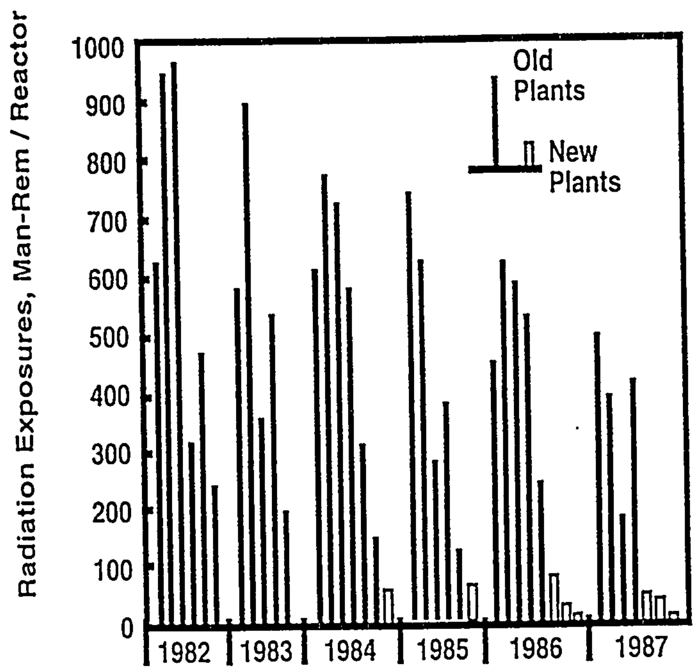

Figure 15. **Decreasing trend of radiation exposure from routine maintenance and inspection work during annual outage**

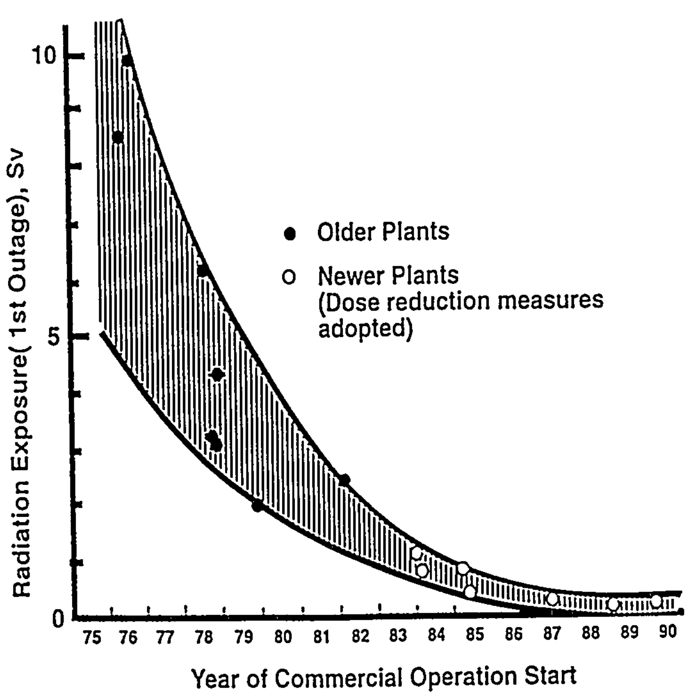

Figure 16. **Occupational radiation exposure during the first annual inspection outages of Japanese BWRs**

basic knowledge than we have now is needed about the interaction between oxide film and ionic metal elements.

CONCLUSION

The radiation control measures adopted in Japanese BWRs were qualified by the investigating data on operating water chemistry and radiation levels. Following the Improvement and Standardization Program established by electric power companies and plant manufacturers, the occupational radiation exposure was reduced this decade by factor of more than ten. Standing on the ALARA concept, however, more measures need to be developed to further reduce dose rate and radiation exposure.

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SESSION 1
DISCUSSION

Wood: I would like to comment on the subject of reduction of exposures through the control of radiation fields worldwide. I think two of the speakers this morning played a major role -- Krister Egner in raising the pH at Ringhals. It is kind of a rocky road with that work, but you can see the success of it. The equivalent in the BWRs is Bill Marble with the zinc injection, which he pioneered. Again, he had a rocky road with it, the problems with zinc-65 that you heard about, but it has been highly successful. We are lucky to have two of the pioneers of this work here today.

Baum: The first question this morning regarding work practices reminded me of my first job as a health physicist. I was in the Safety Department at Allis Chalmers, and the first thing they taught me there was that the safety principles are first to eliminate the hazard. If you can't do that then you guard against it. Then, after you've done both of those things, you develop procedures. It occurred to me as I thought about that question, that our sessions here today have been on the first principle of eliminating the hazard -- eliminating the problem -- and this afternoon we will get into the design of new plants and the "guarding against it" type of actions. Tomorrow we will get into some of the more applied procedural and work practice aspects of radiation protection. Hopefully, some of the questions asked this morning will be answered tomorrow.

The other point I would like to make was that in the introduction I mentioned that the ALARA Center exhibit is over to the right. We also have one over to the left that I neglected to mention, which deals with the DOE ALARA Center Exchange - DOEACE. Bruce Dionne, who is the person at Brookhaven most responsible for the DOE activity, will be out there showing that to you if you would like to explore that. I'd like to thank all the speakers for their very interesting presentations. I'm sure they put a lot of effort into them and we appreciate it.

SESSION 2

PANEL DISCUSSION ON RECENT RECOMMENDATIONS ON DOSE LIMITATIONS

Chair:

Charles B. Meinhold

SESSION 2

PANEL DISCUSSION ON RECENT RECOMMENDATIONS ON DOSE LIMITATION

Chair: Charles B. Meinhold

CHARLES B. MEINHOLD is a Senior Scientist and Deputy Division Head of the Radiological Sciences Division at Brookhaven National Laboratory. His field of expertise is the application of radiological physics and radiobiological data to radiation protection. He has served as President of the National Council on Radiation Protection (NCRP) since 1991, is the chair of the NCRP Scientific Committee 1 on Basic Radiation Protection Criteria, and was a co-author of the basic recommendations of the NCRP and ICRP. Mr. Meinhold is Vice Chairperson of the International Commission on Radiological Protection (ICRP) Main Commission, is member of the Main Commission since 1978, and Chair of Committee 2 on Secondary Standards since 1985. He is President of the International Radiation Protection Association (IRPA), and a member of the IRPA Executive Council since 1984. He has been Chairperson of both the Sievert and Admissions Committees since 1988. Mr. Meinhold serves on the oversight committees for Rocky Flats, Indian Point, Shoreham, and Pilgrim nuclear power stations and was appointed by the NRC to serve on the Blue Ribbon Panel of Three Mile Island 2.

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PANEL DISCUSSION ON RECENT RECOMMENDATIONS ON DOSE LIMITATIONS

Baum: I would like to introduce the chairman of our panel discussion on Recent Recommendations on Dose Limitations, Charlie Meinhold, who is Deputy Division Head of the Radiological Sciences Division at Brookhaven National Lab, and President of the National Council on Radiation Protection and Measurement.

Meinhold: We have a very interesting topic for this morning. I note that it is stuck in the middle of an ALARA session, which seems slightly inappropriate since I think one of the aspects of dose limitation that the ICRP and NCRP have been trying to point out is that dose limits are not based on ALARA considerations, and that you chaps are doing the work to establish constraint and reference levels that we aren't able to do. We can only set boundaries, and you all have to do the work of getting those exposures down to where they should be for everything to be the way both the NCRP and ICRP want to see the situation.

Some of you also realize the NCRP and the ICRP have been up to their usual devilment. As soon as NRC adopts a new set of recommendations, they find that they are again 15 years late, because the NCRP and ICRP have a new set of recommendations. I guess that's pretty much true throughout the country and I'm sure we'll hear it from our panelists as well. Of course, both of those organizations are merely reacting to the new information that we get from the remarkable studies conducted in Japan on the survivors of the atomic bombings.

Having said that, I think we have a very interesting panel who ought to be able to address a few of these issues. Clearly those recommendations start with a reduced dose limit. I thought that Dr. Cool picked it up pretty well when he pointed out that what we are really after is a lifetime limitation of about 1 Sv or 100 rem. It almost doesn't matter in a real sense in terms of the radiobiology how you do that, although you should keep the dose rate down certainly below 5 or 10 rem per year in order not to exceed the risk estimates that led you to the 100 rem lifetime suggestion in the first place. Given that, we have to look at how that will work. The other aspect of, particularly the ICRP recommendations, is because of the "intolerability of exceeding the dose limits," ICRP has suggested a system of dose constraints. They suggest regulatory authorities should impose dose constraints on various segments of their regulated organizations. This is indeed an optimization (ALARA) step made by the federal agencies as they look at each practice. Perhaps some of our panelists can react to that to some degree.

More importantly, of course, is the idea that everyone needs to understand that the dose limits as they exist are only acceptable because of ALARA. It is the distribution of doses below the dose limit which is so important. Perhaps we need to reflect on that as we look at some of the information that some of our panelists can bring us. The panelists this morning are each going to give us a few minute discussion on these recent recommendations.

Don Cool will decide whether or not he wants to do that, already having had about a half hour of your time earlier today, but he may feel that he needs to defend himself in some way. Don Cool is, of course, the Branch Chief, Radiation Protection Health Effects Branch, Division of Regulatory Applications, the Office Nuclear Regulatory Research. I'd also point out, because I'm a bit parochial, that he's a member of ICRP's Committee 4.

Mary Measures is the Director of the Radiation and Environmental Protection Division, Atomic Energy Control Board of Canada. I suppose a lot of folks don't think Canada is a foreign country, but I'm afraid that it is. I'm sorry about that Mary, but we'll consider you giving us information from another country for this particular talk. It is interesting that, of course, we work very closely. As a matter of fact, the NCRP has often had members of the Canadian organizations on some of our panels.

Christer Viktorsson is the Head of the Department of Nuclear Power Inspections and Emergency Preparedness at the Swedish Radiation Protection Institute. Sweden has had a long history in radiation protection. Actually, the first meeting of the ICRP was held in Stockholm in 1928, and Sweden has been very much in the forefront on the whole question of reducing exposures.

John Schmitt is a Manager of the Nuclear Energy Institute. His name tag actually explains it better as NUMARC. But in the binder you will find that he's under this new name of NEI, the Nuclear Energy Institute. John has been very active at NUMARC bringing the industry people together to look at recommendations as they come, particularly from the NCRP, and the input from them has been helpful.

Jacques Lochard is with the CEPN, I guess probably the interface between the regulators and the operators is sort of an ALARA Center for Europe. Now, I will also be parochial here in that Jacques is a member of the Executive Council on the International Radiation Protection Association, of which I am the current president, and I suppose we should try to tell you that we should all try to get to Vienna, Austria, for our International Congress in 1996.

Frank Rescek is here as defender of the faith, because he's the guy who is on the floor representing the utilities, the users. I have worked with Frank. He's helped me on a committee that's been looking at whether or not it's possible to live with new dose limits, and how we would do it if we had to.

I'd like to invite each of the panelists to give a few minute presentation on behalf of the force behind the things they do, and I'll ask Don if he wants another few minutes.

Cool: You know I can't resist taking one or two, although I don't intend to try and repeat the things that I said earlier, there are several things that we need to keep in mind as we consider changes to regulatory structures changes to operating systems and the other things that go along every time someone suggests that perhaps we're not providing the appropriate level of protection. I want to emphasize "suggests that perhaps we're not providing the appropriate" because the meaning of that particular phrase is truly in the mind of the beholder, and one of the things that we are faced with these days, more than ever before, is a clash between viewpoints of groups, organizations, members of the public, in terms of what is appropriate protection, and what is the way to achieve it. Here we have been talking about ALARA, the ALARA process of reducing exposures below a limit -- the classical radiation protection approach, and Charlie has reemphasized, appropriately, that those limits are an upper boundary, a suggestion of what might just be tolerable or something, and certainly not something that we would want to have over a long period of time. At least in the United States, that philosophical approach is in direct clash with another philosophical approach, which is the establishment of a very low goal, and then seeing how close you can come to achieving it. The typical approach that is used, at least in the United States, in regulating chemicals, regulating other hazardous materials. What you discover is that you have two boundaries. You have a limit on the upper end, you have a goal on the other end, and in the middle you have a process, which is exactly the same process whether you call it by ALARA, whether you call it achievement of goals or maximum tolerable levels above the goal. It is that same process. So one of the first things we need to consider is the philosophy and then the application of what we are really doing and what we are really about in either one of those philosophies in order to make it work. And that gets me to what I'd emphasized earlier in talking with some of you with regard to public acceptance. Because no matter what we do with the regulations, no matter what we do with our operations, if we do not have some measure of both public understanding and public acceptance of those operations, we would have really failed in the end despite all of our technological achievements. In terms of the impact of the recent recommendations, I would like to note -- I'll do a brief bit of advertising for Charlie here -- out on the table is a copy of a study that Charlie Meinhold did for the Nuclear Regulatory Commission on the "Impact of Reduced Dose Limits." In the U.S. NRC's great acronym vocabulary, it's NUREG/CR-6112. That gives you a lovely little identifier. But as a rather interesting first step in a study which we are pursuing here in the United States, an attempt to try and find out what would be the impact of changing occupational dose limits from the present 5 rem/year, the old system, 50 mSv/year value, to a variety of things, either the 20 mSv/year average, some combination of 50 and a one and/or otherwise averaging as NCRP has suggested. We found a rather disappointing response in terms of people wanting to think about it right now. I think Charlie would testify that he had a terrible time in trying to convince people that they should give him any sort of data. But we found that there is some impact out there, certainly, more with the perceived nature of *complying* with a limit, rather then the reality of being in *compliance* with a limit, and there's a world of difference there, too, between whether or not you are achieving the objective of controlling exposures within a certain criteria which I believe is already the case, clearly demonstrated from the charts that I put up earlier today, versus the feeling that I am sufficiently far below that in terms of my averages that when the NRC inspector shows up at my door that I am comfortable with the fact that he's there and that he's not going to find something and he's not then going to pick on me in some way. So those are some of the issues to start off with. Maybe we'll deal with some of the other ones later as we go through it.

Measures: I think just before I start, I'll mention that I will go through the dose limits, not ALARA. In Canada, we are of the feeling that ALARA isn't something new. It is something that's been part of the regulatory process ever since we started regulating. We maybe didn't have words for it, but it was the way to go. In Canada, it is not something that we add on as a special program, it's part of a good radiation protection program. As far as the new dose limits are concerned, I think ALARA is just part of it, just as it is part of your every day practice. Now what we did do in Canada, was in 1991 we issued a consultative document, C122, which stated the Atomic Energy Control Board's intention to follow to a large extent the recommendations of ICRP-60. One significant difference was we decided that we would probably go directly from 50 mSv/year to 20 mSv/year, without including the 5-year averaging period. However, during the consultation process, we had many comments from industry who found that this would be perhaps a bit too restrictive, not giving them the flexibility they thought they needed. From the nuclear power plants' perspective, they thought that this would be a problem, particularly for special maintenance. For example, in Canada there are problems requiring the change of pressure tubes that have to be pulled and reinserted, and also there are boiler cleaning programs going on. They felt that probably they would not exceed the limit of 20 mSv, but they would like to be able to approach it without worrying about legal consequences should they exceed it. In other words, they didn't want to unnecessarily restrict people from radioactive work. We received the same comments from the mines, because as you are aware, in Canada there are some very high grade ore uranium mines, who were also concerned about limiting to 20 mSv without the 5-year averaging. In fact, they wanted us to go directly to the lifetime limit that Charlie was mentioning before. We have not agreed to do that, but we have agreed now that we will institute the 5-year averaging. However, that is a little bit late from the nuclear power plants' point of view because the other part of the equation is the unions, and once the unions read C122, which said we weren't going to allow the averaging, they didn't want any part of it. At one power plant utility, the union has said strictly it will not allow above 20 mSv/year. For another one, it is, in fact, part of the collective agreement, that 10 mSv/year will be the dose limit, provided that the collective dose is not increased as a result. So the unions are doing what would be part of the ALARA equation.

Another important area where we are not going to follow ICRP recommendations was for the dose limits that they recommend for pregnant workers. They recommend 2 mSv to the abdomen, plus or and--they don't really qualify if it's a plus or a combining formula--.05 annual limit of intake of any radionuclide. During our consultative process we found that the women across the country were very concerned. One, it's going to be almost impossible to measure and demonstrate compliance with those kind of limits, and especially, there is a concern about loss of employment opportunities for women, especially in nuclear medicine. So we decided in Canada to listen a bit more carefully, and we held a series of meetings across the country. We, in fact, had a series of workshops that we held at seven cities and at one mine site, to get the input from management, from workers, from unions, to see exactly what the concerns were. The overwhelming response was that women are concerned about the loss of job opportunities. They felt that their fetus would be more at risk from their losing the jobs or not getting a high paying job in the first place, then they would from any additional risks from the radiation exposures. They felt that, just as they are allowed to make an informed decision about the safety of their fetus with respect to alcohol and cigarettes, they should be given the information and allowed to make an informed decision on whether or not they would continue to work in a radioactive area. The Atomic Energy Control Board is now looking at some limit above the ICRP recommendation, by which we assume the ICRP means 1 mSv to the fetus, but below our current limit of 10 mSv. We are looking at the number of 4 mSv during the duration of a pregnancy as a dose limit to the fetus. One other topic we considered was hot particles, because of NCRP's recommendations on specific limits for skin dose from hot particles. We had a good look at the problem in Canada, and we came to the conclusion that there just aren't enough hot particle incidences for us to even bother considering that as a regulatory concern, at least not as something that we have to specify in the regulations.

The final point that I would make is with respect to doses to members of the public. We find that ICRP recommendations are a regulator's nightmare with this respect. They started with 5 mSv/year, then they added, well that's OK, provided over your lifetime you don't exceed 1 mSv/year on an average. Then they changed it to, 1 mSv, but it's OK to go up to 5 mSv sometimes. Regulating sometimes is very difficult. Right now they are saying 1 mSv, but you could have a 5-year period of 5 mSv over 5 years under special circumstances. That's not a problem with respect to nuclear power plants in Canada, but it is with respect to children and other relatives of patients in nuclear medicine. ICRP has washed their hands of that saying that's medical exposure. We think that perhaps it is true for adults who could make an informed decision. We're not sure that in the case of children that it would be true. So that's just some of the problems that we are wrestling with at the moment.

Viktorsson: First of all I would like to congratulate our U.S. colleagues for their very nice efforts we have seen this morning concerning dose reduction. I have followed very closely the work done in the United States in recent years and now I think we see the fruits that you can harvest from the very, very hard work that has been done. From the Swedish point of view, we have seen in some plants, rather dramatic increase in the last two years concerning collective doses. This is, of course, of concern to us, and I totally agree with what Don Cool said that public acceptance is vital for this industry to survive. So we are doing our best to find the means to reduce these doses. Not all these are spelled out in regulations, but in very close discussion with the industry. As you mentioned, Mr. Chairman, we are very close to the ICRP in Sweden, and we have implemented the ICRP-60 in our new regulations from 1994, and some basic elements of those regulations are first of all the ALARA programs, we are now going to emphasize more than we have done before and one particular aspect is the commitment of management. We strongly believe that radiation protection is not an isolated process. It has to be integrated into the overall management of the power plants. We have also in the new regulations issued new dose limits. We are not going to change the annual dose limit, it will still be 50 mSv per calendar year for the individuals. However, we have introduced the ICRP concept of 100 mSv in 5 consecutive years. That will apply from the January 1, 1994. There is also in our regulations that were issued in the late 1980s a lifetime limit of 700 mSv. What we also think is rather important is a sort of ambition level or planning level on collective dose. This was already issued in the 1970s with the 2 person-Sievert per gigawatt installed electricity. In the new regulations we have emphasized this even more, but it must not be interpreted as a limit, it is a sort of planning level for the utilities. But we don't believe only in dose limitations, and as I said earlier, we believe very much in the optimization process and in the ALARA programs, and that should be the sort of focus for our dose reduction efforts.

Schmitt: A change in perspective now as we go to the licensees or the users' portion of the panel. The record of doses in the U.S. commercial nuclear power industry is that occupational workers generally receive less than 2 rem, or 20 mSv/year, a rate similar to the 10 rem in 5 years in the ICRP 60 and less than **n** rem lifetime where **n** equals age in years, which is part of the limitation system in NCRP-116. We saw this in the data that was displayed this morning and I think we will see it this morning as the panel and the workshop progresses. Therefore, the risks to workers due to their exposures to radiation in the course of their job is generally equivalent to the risks associated with the ICRP and the NCRP systems of dose limitation. The radiation protection approach in the industry which has produced this risk management is structured like this. Radiation protection programs actively practice ALARA and the programs are designed to assure that regulatory limits are not exceeded. The health physicists responsible for these programs are aware of the NCRP and ICRP recommendations on systems of dose limitation. These recommendations are generally considered in making decisions about the programs. Formal adoption of these recommended systems of limitation would be by way of regulation. In considering whether the current regulation should be changed, the potential benefits, such as risk reduction to individual workers, must be considered relative to potential impacts such as increase in collective doses for the population of workers. This consideration is best done by anticipating the performance to be achieved by programs redesigned to assure regulatory compliance with the changed regulation -- which is different than achieving the objectives without a regulatory mandate. Optimal management of the risk, via operational radiation protection programs, is the principal consideration in looking at whether formal adoption of the recommendations is more appropriate than less formalized recognition. Also, if formal adoption via regulation is selected, the transition would need to be carefully planned and managed to assure that the benefits of current radiation protection programs are preserved and the enhancements sought are fully realized.

Lochard: As Charlie said in his introduction, I am working in between regulation and operators and I will try to reflect a little bit on the topic of dose limitation from the two perspectives. From the regulatory point of view, we are in France at the moment in the middle of the discussion about the adoption of the new Directives of the Commission of the European Community, and there are some interesting elements to mention at this level. First, there is a unanimity in France to adopt the ICRP system as it is proposed in publication 60. There is, of course, an ongoing discussion about how to apply the flexibility, in practice, with respect to the 100 mSv in 5 years. From the operational point of view there is no real technical difficulty and there is a consensus about the way to proceed. The main difficulty is related to the question of the confidentiality of the information about individual doses. A point on which we also have discussions is the problem of the role of dose constraints, specifically their regulatory status. I think there is also a majority to say that dose constraints could be a good tool to force people to think more in terms of ALARA, but also that dose constraints, if they have to be operational, should remain a matter for operators. This is the situation at the regulatory level. From the practical point of view, I think you probably know, that after a long period of hesitation, to say the least, France has jumped into the ALARA culture two years ago with the leading role of EdF. Now there are a lot of ALARA programs in power stations and we already are seeing very good results. I think this will be shown by different speakers during the week. We had an increasing trend in collective exposure per reactor over the last ten years and since the ALARA programs have been set up we now have a clear reduction. But we have also to be aware that this effort is mainly done in the nuclear industry at the moment, and that in the medical as well as in the conventional industry fields we are far from these good results. Just to finish, I'd like to come back to one point mentioned by Don Cool in his introductory paper this morning about the crucial need to pass the ALARA message to the public. Beyond the technical aspects that will be discussed the during the week, we have to be aware there is a philosophy of how to deal with residual risks in our society. I think we have to give this message to the people, especially with all the implications from the economical point of view, but also from the ethical point of view.

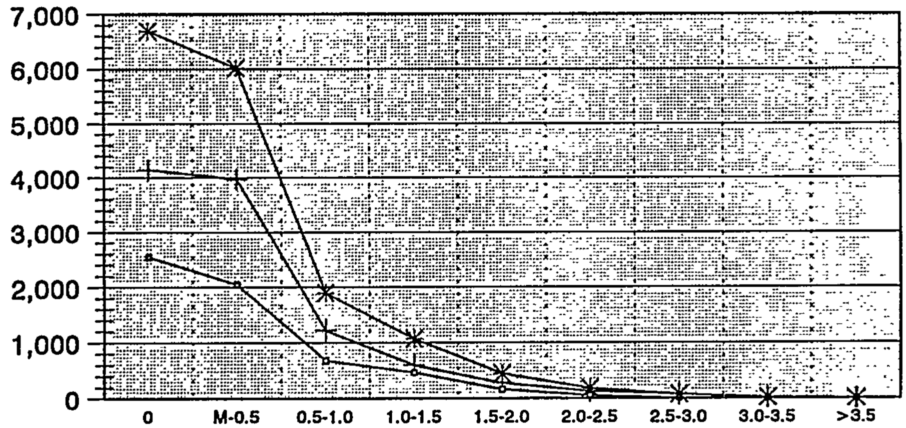
Rescek: I support the views and position expressed by John Schmitt. I believe that Commonwealth Edison, specifically, and the U.S. nuclear utilities, generally, are keeping individual doses ALARA and well below the regulatory limit of 5 rem/yr. Commonwealth Edison owns and operates twelve reactors (three two-unit BWR sites and three two-unit PWR sites). The 1993 year-end dose summary for all commonwealth Edison plants is shown in overhead #1. Note that there were 57 ComEd employees and 221 contractors who received greater than 2 rem last year. Furthermore, no Edison employees and only 18 contractors received greater than 3 rem. No one exceeded 4 rem in 1993. In contrast, overheads #2 and #3 show Edison employees and contractors dose summaries for the five-year periods 1989-1993 and 1984-1988. Only one Edison employee received greater than 10 rem total (average of 2 rem/year), but less than 15 rem total (average of 2.5 rem/year), in the last five-year period. Similarly, there were only 12 contractors who received greater than 10 rem total (average 2 rem/year) for the five-year periods 1989-93 at Commonwealth Edison facilities. For comparison purposes, although a fair number of Edison employees and contractors receive greater than 2 rem in 1993, only a very small number of individuals received more than 10 rem total (average 2 rem/year) over the last five years. Thus, our experience shows that having the flexibility to permit workers to receive greater than 2 rem in any one year does not hinder our ability to control individual lifetime doses. For example, ComEd plants are on 18-month refuel cycles. Consequently, one year out of three, each of our two-unit sites will have two refuel outages. during the years a site has two refuel outages, it would be very difficult to comply with a 2 rem/yr limit. Recently, ComEd reduced its administrative dose control level from 3.5 rem/yr to 3.0 rem/yr. Our analysis shows that this change would impact 33 contractor workers at a cost on the order of $200,000 to $700,000. It's important to note that this is an administrative control level and we have the flexibility to permit workers to exceed the 3 rem for critical situations with appropriate approvals. Similarly, I believe

Rescek - Figure 1

CUMULATIVE DOSE TOTALS - 1993

FOR PERSONNEL WORKING AT CECO NUCLEAR STATIONS
("HARD" DOSE VALUES FOR 1993)

NUMBER OF WORKERS WITHIN DOSE RANGE



	0	M-0.5	0.5-1.0	1.0-1.5	1.5-2.0	2.0-2.5	2.5-3.0	3.0-3.5	>3.5
EDISON	2,544	2,046	679	467	166	53	4	0	0
NON-EDISON	4,147	3,973	1,215	598	280	130	73	18	0
TOTAL	6,691	6,019	1,894	1,065	446	183	77	18	0

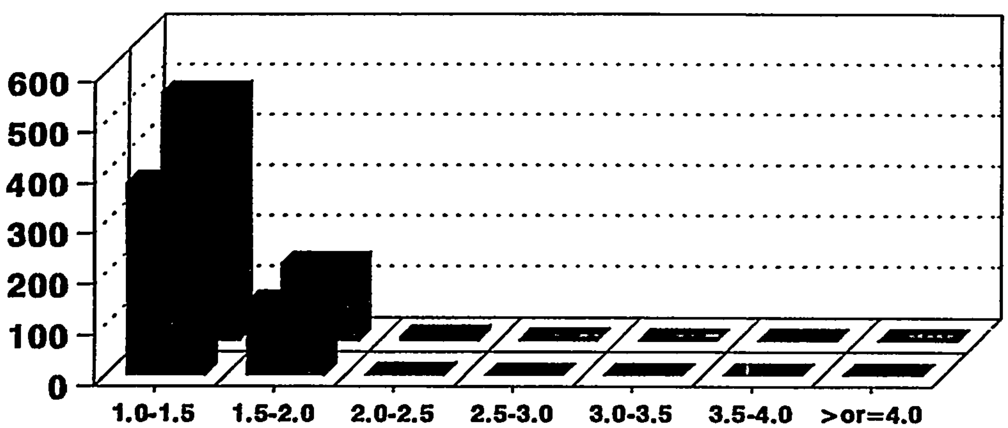
DOSE RANGE (REM)

NOTE: VALUES INCLUDE NON-EDISON AND SOME CALCULATED DOSES
ON THE X-AXIS, 0 MEANS BADGED BUT ZERO DOSE; M MEANS MEASURABLE (>0)
RACCE93.CH3 (Chart updated 01/24/94)

Rescek - Figure 2

DOSE RANGES (IN REM)
FOR FIVE YEAR PERIODS

# of INDIVIDUALS



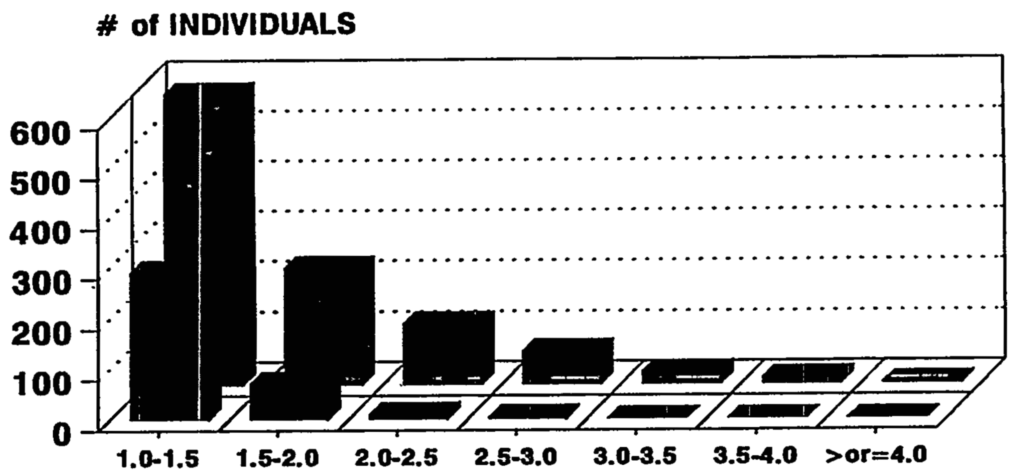
	1.0-1.5	1.5-2.0	2.0-2.5	2.5-3.0	3.0-3.5	3.5-4.0	>or=4.0
1984-1988	489	150	7	1	0	0	0
1989-1993	379	142	1	0	0	0	0

RANGE (AVG. REM/YEAR)

COMPANY PERSONS ONLY

Rescek - Figure 3

DOSE RANGES (IN REM)
FOR FIVE YEAR PERIODS



	1.0-1.5	1.5-2.0	2.0-2.5	2.5-3.0	3.0-3.5	3.5-4.0	>or=4.0
1984-1988	579	230	120	65	23	11	2
1989-1993	289	70	9	3	0	0	0

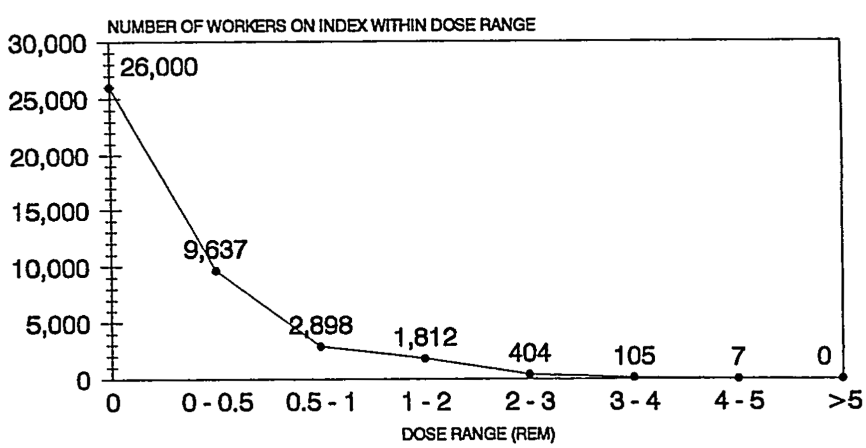
RANGE (AVG. REM/YEAR)

NON-COMPANY PERSONS ONLY

Rescek - Figure 4

1993 INDEX DATA

INCLUDES DOSE FROM ALL OCCUPATIONAL SOURCES



Includes data from 60,000 INDEX records as of 12/31/93
Data compiled by Nuclear Data Inc./INDEX
Chart prepared by Commonwealth Edison Co. 5/6/94

most utilities have set administrative controls well below 5 rem/yr to keep individual doses ALARA. Finally, my last overhead shows data on transient worker doses which I obtained from INDEX. INDEX, for those of you who are not familiar with it, is the integrated nuclear data exchange program. I believe there are 18 utilities representing 33 reactor sites and 60,000 total transient employees in the INDEX data base. For 1993 there were approximately 516 individuals who received greater than 2 rem and another 1,812 who received doses between 1-2 rem. If a 2 rem/yr regulatory limit were promulgated, then the industry would establish administrative levels on the order of 1.5 rem/yr to ensure compliance. Hence, the number of people impacted based on the INDEX data would likely be in excess of 1,000. Assuming INDEX represents about one-third to one-half of the total number of transient workers in the U.S., the total number of workers impacted and the total cost would be substantial.

Meinhold: Our panelists have set the stage for a discussion on the potential impacts of these new recommendations, and since this supposed to be a panel discussion, it is open for questions and comments. Please go to the microphone and identify yourself before asking your question. While you are all thinking up your questions, perhaps I can make a few comments. Some of the data that John and Frank talked about clearly demonstrates that they are doing a good job in terms of controlling the average dose to worker, but as ICRP laid out its rationale, the problem is that the "average" person is not the person we are concerned about when we set a limit. We made that mistake in 1977 when we justified our dose limit on the basis of an average, but in the Publication 60 and in NCRP's Publication 116, we're only talking about that very rare individual for whom the dose limit is acceptable based on a comparison for people whose jobs put them at the top end of safe industry (deep sea fisherman, etc.) ICRP said that there is an upper level of risk that people will tolerate, which is about 1 death per thousand workers per year. It is this criteria which applies to Rescek's 1,812 workers. It is the distribution below the limit that is truly an ALARA issue, and I can assure you even further that neither ICRP or NCRP could have adopted their dose limits if they thought that they were going to be the basis for controlling exposure. That's not the purpose of the dose limits and I think it is important to clear up any confusion. The limits are only a boundary condition for those who might be at the highest end of that risk level and not something which drives the average. I think it's clear that the ALARA and the dose minimization programs at the power plants drive the average down and have to continue to do that. So if any of the panelists would like to react to that, I'd be happy to respond.

Rescek: The number of individuals in the nuclear industry who tend to receive annual doses near the limit can be inferred from the data shown previously in Don Cool's graph. His graph showed that for the early 1980s, approximately 700 to 800 workers received greater than 10 rem in five years. However, in the last five years, Don's graph showed that the number had fallen to only 150 workers. Clearly the industry has improved its performance in lower individual doses since the ICRP and NCRP made their recommendations on controlling lifetime dose. I believe that the number of workers who exceed 10 rem in five years will continue to be reduced without reducing the 5 rem/yr limit. Furthermore, I strongly believe that we need to protect the lifetime risk to all workers, and the best way to achieve this is by establishing a separate lifetime dose limit consistent with the NCRP recommendations, including the grandfathering criteria for people who already exceed the lifetime limit.

Meinhold: Are there any questions?

Unidentified: This is more of a comment than a question. One of the things you talk about is that there are only 150 people at this point in time that are greater than 10 rem in 5 years. However, I don't know what the rest of the utilities are doing, but in our utility we are getting into a lot of pressure to reduce the crew sizes to perform certain tasks. By reducing the crew sizes, without reducing the dose that it takes to perform the job, you are actually increasing the exposure for each individual on that job. So with that in mind, if that is what is going on in industry, and I expect that it is with cost control measures, I think we may even see an increase in the number of people that are greater than 10 in 5. It's one other variable out there. I was reading through the NUREG report that you are talking about and what they said is that at 2 rem that is considered a safe industry, whatever that means, because even a safe industry is being redefined now as we talk about this. Safe industries are getting safer. On the upper end of the scale they say that the risk assessment is equal to that of a miner or deep-sea diver. Now in all these other safe industries, I would venture to say that they have people that work within their industry that it is publicly acceptable for them to take on riskier jobs. As a matter of fact, at the power plant, we have divers. Bringing a diver into the spent fuel pool reactor cavity is a little bit more risk that my sitting at a desk figuring out how much exposure he is receiving, but that is acceptable because of the fact that he is a diver. So one of the things that I would like to address, and this links back to Mr. Cool's comment about public education, is that it seems to me that it would be a lot more reasonable since we have a relatively small portion of people that are in the so-called high-risk category up with the deep-sea divers, that you would be better served if you would take all the money that we spent to try to get these few people less than 10 in 5, and take those resources and put them into educating the public to explain to them why it is OK to have people within the nuclear power industry have the same risk as a deep-sea diver or miner.

Measures: I would like to comment on that, especially with respect to the miners, it is an added problem where you have the miner worrying about a rock falling on his head, plus the radiation exposure. I think that when we are dealing with miners we have to add in all of these things so that we are looking at the total risk to the worker and not just one of the compartments. I think it is very important to not forget that these other risks are there.

Aldridge: I work for Westinghouse Hanford Company in Richland, Washington. I work for the DOD, I'm not in commercial nuclear power environment, but I would like to make a comment and ask you a question. In 1990 we reduced our administrative levels to 2 rem/year. We did a data search in internal dosimetry on all of the individuals that would be impacted. We also initiated at that time a 1 rem x age lifetime limit. We had 50 individuals out of roughly 1,200 employees that we had to take a serious look at their lifetime dose. We also had 3 individuals who exceeded the 2 rem/year due to old, internal deposition and exceeded, one in particular, his lifetime dose. The point I want to make is that we need to educate the public, but we also need to consider educating the worker. For years and years we have told these workers that the limits were fine, you were safe, everything was in control, they were not to worry. As health professionals we gave them this message. Then all of a sudden we impacted the workers, 50 individuals. A small amount of the total work force, but those individuals talked to other individuals and sometimes you can have problems in that area, particularly in the case of the three workers that were restricted. They can no longer work with radiation. They can no longer pursue their livelihood. One individual was only 32 years old. That is a very, very difficult situation to go through. That is my comment. I would like to ask a question of Frank Rescek. You said that you are under the legal limit right now of 2 rem/year? You are under that or you will be shortly?

Rescek: We have an administrative control level of 3 rem/year, not 2, at this time.

Aldridge: I thought I had heard you say you were going to 2 rem per year and then you were going to look at 1 and 1.5. We do currently have .5, 1 rem, 1.5 and 2 rem, and each one of those levels requires management signatures until the individual reaches two rem per year and then they can no longer work. So I guess I misunderstood that.

Rescek: To clarify that again, we have an administrative control level of 3 rem/year at which point to exceed that you need the station manager's approval. The number of exceptions is very rare, but the approval process to go above 3 rem on a rare case-by-case basis is available, if justified.

Schmitt: As we have heard, and as you alluded to in the question and comment, how we broach the subject of ALARA and how we manage it as we talk to the public is very important, but often we think of the public as those people outside our fence. I think we've got an important public in our workers and we really need to address them. We have to carefully consider what we are doing as we look at potentially lowering the limits. If we practice the recommendations and use it to help us by our administrative means to get the doses and the risks down to individuals it is very beneficial. If we also do that via regulation, where we say now it is not longer acceptable to this government agency or anywhere in this country, or whatever, to allow people to receive doses greater than this number or these numbers, this system of numbers, the we create perceptions among whoever is affected by those limits, any public, including our workers. There are all those social perceptions and perceptions about whether or not they have been protected in the past. Have they been safe? Will they continue to be safe in the future? How about their employability? How about their expectations, their families' expectations about their health in the future, those kinds of things, as well as the potential for litigation. The people may think that because we have now discovered some new level of risk that we haven't been protecting them and maybe they ought to come at us through litigations. There are also perceptions set up which could be dangerous and damaging to the licensees who have been protecting these people in the past, but who may be perceived as not having been providing them with an adequate level of protection. I think that needs to be carefully looked at as we formalize these recommendations.

Vikorsson: This adds on to what Mr. Schmitt said. I totally agree, lowering the dose limits causes concern among workers, particuarly, because they are going to ask themselves or us, have we been protected before or not? So we have had several discussions with contractors for example and they have these types of concerns. I think when you issue a new regulation they have to be accompanied with appropriate information, proper education programs, trainings, etc. Therefore, in Sweden we have asked the utilities to put more emphasis on training and on education of the workers.

Lochard: I'd like to speak on this topic because I think it's a crucial point when we discuss limits. Each time there were changes in the past with dose limits, it has been seen as a catastrophe by the industry in the beginning, just because there were these sort of considerations saying that limit is something like above the limit isn't safe and below the limit is safe. But, in fact, if you read carefully what has been written by ICRP, it has never been presented like this. What has been said all the time is that the limit is the upper bound of what is tolerable in the present vocabulary, and I think this is what we have to emphasize when we speak about education of the public or of workers. There are two different problems with radiological protection. One side is the problem of deterministic effects and in this case it is a matter of respecting some thresholds. It is safe under, it is unsafe as you go away from this threshold. As far as stochastic effects are concerned, this is just a matter of tolerability of risk and what society at a certain point in its development is able to cope with. What we have to tell people about this idea of residual risk is that we are all living every day with a set of residual risks. When we go to work every day there is a specific residual risk. Whether or not this risk is founded on scientific evidence is another business. This is the problem we have about the impact on low doses. Taking into account the doubt about the existence of a threshold for stochastic effects, we have assumed prudently that we do as if there was no threshold, and based on this assumption, we have to educate the public on the fact that we are living with a residual risk. When this is understood, a change in limit will not be seen as a catastrophe, but a general improvement in protection because societies are getting more resources and are able to reduce further residual levels of risk. I think this is a very important point in terms of the message to the public but also for the workers and we will not be confronted anymore with this type of attitude: don't change the limit because you are going to put panic among the workers. On the contrary, if you change the limit it's a very good sign. We've made a lot of progress. It's something like ALARA I think.

Meinhold: If I let the discussion continue, we will never be able to accept another question. We'll take the next question and return to this issue if we can.

Westbrook: I am from Oak Ridge National Laboratory, and Theresa Aldridge, who just asked that question, is one of my colleagues in the DOE system, and I would kind of like to know why some rems are more equal than other rems. Notably, Department of Energy rems and NRC rems. We, like Theresa's Westinghouse Hanford, some years ago after my former boss went to the previous Brookhaven ALARA conference and got all inspired, we went to the 1 x age limitation and we instituted a series of administrative goals. We, too, have had some workers confused by what was safe and what was not, but we mostly have been able to iron that out and educate the workers on it. We still have questions raised, but we are doing OK. We thought we were being very proactive to do that but the DOE has gotten the wind up and they have instituted through the Radiological Control Manual a de facto limit for Department of Energy facilities including all contractors and subcontractors, of 2 rem/year. In order for any worker at any DOE facility to exceed the 2 rem/year, application has to be made in writing to a program secretary, of whatever DOE program it happens to be, in Washington, D.C. Can you imagine, if you people at the utilities had to apply in writing, to say, Dr. Cool over there, for permission to give a worker over 2 rem. Even 10 mrem over 2 rem, if you thought he was going to get over 2 rem, you would have to apply in writing and wait and wait until Washington got back to you on that. I would like to know why some rems are more equal than others, considering the DOE system as a whole has a much lower dose curve than the NRC, why is that Dr. Cool? And the reason I'm asking you is a lot of the time when we ask questions of the DOE, they will say things like "Well we are doing in our new 10 CFR 835, which is sort of the analog of 10 CFR 20, we have certain provisions in here," and I'll say to the DOE folks, "Why is that in there?" And they will say, "Well that's the way the NRC does it." "Well how come that's in there." "Well that's in NCRP 60." Yet they have declined to adopt 10 CFR 20 in toto. The federal agencies have all declined to adopt ICRP 60. They take a "cafeteria" approach to 10 CFR 20 and to ICRP 60. What they like, they adopt and write into law, and what they don't, they sweep under the rug. So we have this regulatory inconsistency. Perhaps you would like to comment on that since the NRC seems to be the lead regulatory protection agency.

Meinhold: I want to hear this answer, too.

Cool: So do I. I think in essence what you have identified is the heart and soul of most of the discussion that we've had around here, which is the whole problem of establishing a limit and the legalities that go along any time you draw a line anyplace. Be that at 20 mSv, be it at 50 mSv, be it at 5 years with 100 mSv -- no matter where you draw a line you have then arbitrarily, but perhaps not capriciously, but certainly arbitrarily said that anything less than that can be treated in one way, and anything greater than that can be treated in a different way. Although if we assume for the moment that we really do believe in the linear nonthreshold hypothesis for purposes of laying this out, 20 mSv vs. 20.01 mSv only changed the incremental risk to that individual by some very small 10 to the minus, some number down there. Nevertheless, you have that problem within the regulatory schemes of things where we have decided for legal purposes, for control purposes, or whatever that might be, that there needs to be some framework laid out to provide some boundaries and that really gets to the biggest difficulty that I see in moving to things like the publication 60 or the NCRP publication 116 sorts of values, is the tradeoff between what would be a better system in terms of flexibility and perhaps a system which would more clearly recognize radiobiological realities vs. a legal system, which certainly the U.S. and the DOE and NRC live under, has gotten tremendously litigious, that we just wait for the next suit to come around the corner, we have our probability of causation tables, which are constantly changing, and how to try and provide enough flexibility so that the difference between 2 rem and 2.01 rem is recognized from its radiological standpoint perhaps separately from its legal standpoint in defining good practice. Because in the end that's really what we want to do. We want to define and carry out good practice, and ideally we would do that and we would never actually bump against the legal requirements. The short answer to your question, why are rems different? Because we had to draw a line in the sand. I don't like it either.

Meinhold: I think also one of the answers to this, of course, is that EPA has the overall responsibility for coordinating this and NRC is merely reacting to the 1987 guidance of the EPA. He doesn't like to hear that, but EPA hasn't even reviewed the ICRP or the NRCP 116 in a formal way. Now the difference between DOE and NRC is between the owner and the renter. The NRC has got licensees and all of the legal constraints that are involved in their putting more constraints on the licensees. The only interrelationship they have is the regulation. The DOE owns its facilities. It pays for the operation and can set up any rules it wants and if they react to the new data in a way that is more conservative or more up-to-date, if you like to use those words, that's something they have to understand that they are doing in terms of the additional costs and additional concern that it raises. But I think there's no inherent reason that they can't do that as anybody else. As a matter of fact the NCRP always expects individuals to look at our recommendations even more than the federal agencies because we like to have people thinking about them within 3 or 4 or 5 years after new risk information becomes available, and the federal agencies can't do it for 10 or 12. So we still think people ought to look at it and think what they should be doing now.

Cybul: I guess in listening to all the rhetoric going on, and listening to John Schmitt and his concern about expanding the normal number of people who get dose if we limit the amount of dose any one person gets, the question I have to ask the scientific community is why aren't we dealing with the total lifetime dose as the primary upper limit, the value you alluded would be 100 rem isn't the right answer. If you take 2 rem per year, and I assume most workers work 40 years in their lives, I would get 80 rem, if I take your age in rem NCRP I'll get 60 rem. So we've got three numbers here already. Why don't we come up with a reasonable risk based on a total lifetime dose and let the regulation be loose enough so that the people that have to live with them can work within a fairly good flexibility and manage their resources so that they don't exceed that total outer boundary.

Meinhold: I could react just briefly. One of the things that regulators have to worry about, and even the NCRP and ICRP, is a problem of exploitation. That is if I've just got a lifetime limit, say its a Sievert, 100 rem, how do I ensure that it's not being used by an unscrupulous person to deliver 25 rem in a year in order to get that job done faster and use him up, basically. So that's one of the reasons that there is some moderation in the way that it's delivered. The other side, of course, is that 100 rem delivered in increments at higher than 20 mSv a shot, has a different risk associated with it, about a factor of two high. We'd have to keep you below 200 mSv a year anyhow.

Cybul: My point was that we've got some reasonable regulatory limits now which address those issues. Let's just put a top cap on it and not fool around with changing any more regulations.

Meinhold: I guess you are preaching to the choir. Any comments?

Cool: You've certainly identified one of the possibilities and I will add to what Charlie Meinhold has already put out, a couple of perspectives which are probably unique to the regulator. One, of course, is the span over which you think you can exercise some sort of control. A year isn't too bad, most of us are still around next year. Five years perhaps gets a little more difficult because there's a greater turnover. As you start to expand out the time frame over which you will allow people to look at things, you begin to have a greater, uncertainty perhaps isn't the right word, but I will use it anyway, a greater uncertainty in terms of being able to keep track of it, know what has been achieved, deal with responding to changes. We are talking about a lifetime. We are talking about working 40 years or so. We're talking about looking at a sphere of control which is as long as our entire dealings in the modern area with radiation have been up until this point. And you look at the tremendous changes there and it leads you to some measure of uncertainty as to whether or not we can really do that. The other one comes to the point which Charlie had, which was simply the recording, reporting, tracking systems and the use of materials over the longer period of time where compliance gets to be extremely difficult. Up to now we have operated under a system where licensees, at least NRC licensees, had the primary responsibility for controls. We don't go to control of individuals. If you go to lifetime limits, you go to some of these averagings. That means you would have to change your sphere of control from a licensee and the focus on a licensee program to the individuals and that drastically changes, I'll suggest to you, the way in which you'll have to do business.

Meinhold: Except they use form 4 to do that now.

Measures: I'll just make a comment from a Canadian perspective. One, following the tracking isn't a problem because we have a National Dose Register, and everyone has an entry in the National Dose Registry which is one of the few times you can use the social insurance number as a linkage. So that isn't a problem, but it is a problem with some of the small licensees. The Industrial Radiographers, as Charlie was mentioning, like to dose people up and then put somebody else in, so that you have someone who quickly uses up their dose and is then unemployable. We find with the major utilities we don't expect that would be a problem because they seem to be very, very reliable and good corporate citizens. In fact, in Canada we have a much looser approach in that we don't have nearly the regulatory guides that you have in the United States. The licensee tells us how he is going to keep the doses ALARA and we review it, rather than we tell him how he has to do it, so it's a different approach.

Rescek: I believe the utility industry has good record keeping systems and can track lifetime dose. At Commonwealth Edison, we do this now and we have a separate annual administrative control level of 1 rem for Edison employees with high lifetime dose.

Borst: I work for Entergy Operations. We are the operating company for Grand Gulf, Waterford 3, River Bend, and the Arkansas Nuclear 1 Stations. We've got a fair number of employees. The question here from ANI landed on my desk last Monday, and I've been praying for this opportunity to let you all have it. And your original landed on my desk until Tuesday, so that's why it didn't get back to you. Mr. Viktorrson brought up a good point about conveying this new message to the worker about we've been protected before, what's the new limit now, why, is the lower limit safer? Where at the same time in the U.S. we've gone to the opposite extreme by telling them that now we're going to assign you 50 rem this year but it's OK because it's not really there or now we're going to assign you 50 rem to the skin or to an extremity or something like that. So we're going through that culture shift and if we try to do another culture shift from 5 rem down to 2 rem that will be another philosophy burden on them in that aspect Mr. Rescek brought out a good point that I figured out early on that the operational flexibility is really key to what the utilities need in all of this. The utilities could live with a 2 rem limit tomorrow by simply hiring more contractors and we could go from there. He showed on the slides the number of utility workers exceeding those limits now is extremely low, and we can get below that by simply adding extra contractor workers to take up that dose. I would like to urge any contractors in here to comment on this CR 6112 through NEI because I think it's the contractors who are going to be the key to whether this whole thing works or not. So I would like the contractors to do something to get their input into here. Three questions I had and you can answer any or all of them: Does our panel have any thought or insight on whether these new limits and ALARA incorporated into 10 CFR 20 will affect litigation in the future. Up until this point I know the other countries don't have the burden of litigation that the U.S. has, up until this point the bedrock of all of our defenses in the litigation cases has been that we have kept exposures below the legal limit. Do they think this new limit would have any bearing on how that proceeds. Secondly, of the 400 and something people who exceeded 2 rem based on the 1992 0714 report, if we turn those 400 people into 400 plant special exposures, how would the NRC view that. That's a possibility. Right now, I can't speak for the entire industry but I think a lot of plants are not planning on using PSEs very much at all, but with a 2 rem limit we may need to invoke that. I'd like to know the NRC's perspective to that. Thirdly, Ms. Measures brought up an interesting point I hadn't thought about, is this reduction from 5 rem to 2 rem, is that detrimental risk of a person losing his job less important than a reduction in a potential, theoretical cancer fatality?

Meinhold: We're running late so let's have short answers.

Cool: I think he gave me all three, and I'll try to deal with those really quickly. Will the new limits affect litigation? That's a very good question that I do not have the answer for right now, of course. History in the legal system would seem to indicate that it certainly may be used as a challenge. What the courts in their infinite wisdom determine after all that has been laid out is something I don't really want to speculate on as to which way they will come out because one of the things that the courts have proven is that they are not predictable. The second one with regard to how the NRC would view taking the 400+ and turning them into PSEs -- not very well. Because we also look at those as being very limited sort of uses, and you have to be aware that ICRP took that concept back out when it published publication 60. I would hope that would not become a routine way of getting around the flexibility, that there would be other approaches that could be taken and in terms of whether the risk reduction of going from 5 to 2 is sufficient when you balance it off the risk of losing a job, that is also a real good question and brings to the front one of the trade offs we haven't talked about very much which is immediate risk vs. long-term risk, and I'm not at all convinced how that would balance out either.

Schmitt: Let me try several of those. The history has been that the legal limit is regulation generally in the court. Now this is not predictable with great certainty, but generally it has been accepted that the legal limit is what is the legal duty owed. It is what the limit is in the regulation and not ALARA. That is an important point because if the legal duty owed becomes ALARA you can have an awfully hard time defending what you've done and you're at great risk. But I think the point that goes with this question is that there is a very large difference between whether these systems of limitations recommended by the bodies, there's a very big difference on whether they remain recommendations or whether they become the regulatory limits. A big difference -- they are both in perception, and I think when you go to the legal arena. On planned special exposures, etc., I think it's important in regulation where you've got the "you can do this" and "you can't do that" to maintain flexibility for limited situations. It's similar in my mind to the grandfathering provisions that were written into the NCRP situation. You use these under special limited circumstances in order to help better manage the overall risk situation and I think from that perspective they are beneficial. They have to be carefully used, however. On the question of whether we will increase the risk, I agree with Don, it's a needed tradeoff that needs to be carefully considered as we determine whether these should become regulation or not.

Meinhold: We are just about out of time, and I want to thank the panel very much. I thought it was a very valuable contribution.

Baum: I'd also like to thank the panel. It was very interesting.

SESSION 3

ALARA IN NEW REACTORS

Co-chairs:

Frank J. Congel
Robert A. Bari

ALARA IMPLEMENTATION THROUGHOUT PROJECT LIFE CYCLE

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ABSTRACT

A strength of radiation protection programs generally has been endorsement and application of the ALARA principle. In Ontario Hydro, which currently operates 20 commercial size nuclear units, great strides have been made in the last three decades in reducing occupational radiation exposure per unit of electricity generated. This paper will discuss specific applications of elements of the overall ALARA program which have most contributed to dose reduction as the nuclear program has expanded. This includes such things as management commitment, ALARA application in the design phase and major rehabilitation work, the benefits of the self protection concept, a specific example of elimination (or reduction) of the source term and the importance of dose targets. Finally, it is concluded that the major opportunities for further improvements may lie in the area of information management.

INTRODUCTION

[slide 1] Ontario Hydro has been operating nuclear power stations using reactors of the CANDU design since 1962 with the startup of the 22 MWe Nuclear Power Demonstration (NPD) station near Deep River, Ontario. This was followed by the startup of Douglas Point Generating Station, a larger scale prototype station (208 MWe) and the first nuclear facility to be built on the present Bruce site in 1967. Both of these stations are now shut down (1987 and 1985 respectively). Today, Ontario Hydro operates 20 commercial sized units with a total installed capacity of over 14,400 MWe, roughly half of Ontario's total electrical capacity **[slide 2]**. As nuclear units are normally operated as base load stations, total delivered energy from nuclear generation is currently on the order of 60%. These units are located at three geographical sites known as Pickering (8 x 515 MWe), Bruce (8 x 850 MWe) and Darlington (4 x 880 MWe). Pickering in turn consists of two stations (albeit under one roof), Pickering A, which came into service between 1971 and 1973, and Pickering B, which came into service between 1983 and 1986. There are two four unit stations on the Bruce site known as Bruce A and Bruce B which came into service between 1977 and 1979 and between 1984 and 1987 respectively. Darlington is a four unit station which came into service between 1990 and 1993.

A strong commitment to the management of radiation protection has been an essential feature of our nuclear program from the beginning. Corporate policy in radiation protection included a specific commitment to minimize and avoid unnecessary radiation exposure as far back as 1962[1].

The cost/benefit ratio of collective dose to energy produced has improved dramatically over the last three decades at Ontario Hydro stations **[slide 3]**. This is perhaps best illustrated by comparing total dose consumption in 1970, about 1585 person rem for 240 MWe installed capacity with total dose consumption in 1993, 1307 person rem for 14400 MWe capacity. So how has this been achieved? Clearly there are many factors but among the more important are: **[slide 4]**

* management commitment/corporate policy

* integration of operating experience with design of new facilities and rehabilitation of existing ones

* elimination of radioactive sources where possible

* training of personnel/self protection

* radioactive work planning

* use of protective equipment

* decontamination

* use of goals and dose targets

This paper will discuss a number of specific applications of some of these factors to Ontario Hydro's overall efforts at dose reduction and control.

HISTORICAL DOSE PERFORMANCE

A brief orientation to CANDU design is necessary before discussing ALARA application. Some of the major differences and essential features relative to a light water PWR or BWR include: pressure tube design, the use of natural uranium fuel, independent coolant (Heat Transport) and Moderator Systems both using heavy water and on power refuelling. The basic circuit is illustrated in **Figure 1.**

Collective dose from the operation of Ontario Hydro reactors for the period 1963 to 1993 is shown in **Figure 2.** Note that a significant component of the total is a result of internal exposure. This occurs from exposure to tritium oxide normally in the form of tritiated water vapour as a result of leakage from the Heat Transport, Moderator and auxiliary systems. Tritium is produced in large quantities in the core of CANDU reactors as a result of neutron activation of deuterium in heavy water. In general, there has been a downward trend in collective dose over a long period of time followed by a levelling off at about 1300-1500 person rem/a despite significant increases in nuclear generation and major maintenance activities in the last few years. Our collective dose target for 1994 has been set at 1250 person rem.

Average individual doses have also decreased over time levelling off in recent years at just over 200 mrem per exposed worker **[Figure 3]**. This is in fact an established target of Ontario Hydro Nuclear policy, ie 200 mrem per exposed worker. The most highly exposed work group within our operating personnel for routine operations are the mechanical maintainers. Typical average doses for that group are about 300 mrem/a.

MANAGEMENT COMMITMENT

Perhaps the most important and fundamental element of any corporate ALARA program is **management commitment.** This requires that senior management understand the ALARA concept, believe in it and actively support it in managing the business. In Ontario Hydro, the management of radiation protection has been governed by a set of internally generated Radiation Protection Regulations since 1962. These documents apply to all nuclear facilities in the corporation, to all phases of their life cycle and are authorized by our regulators, the Atomic Energy Control Board. Compliance with the Radiation Protection Regulations is a standard condition of each nuclear station's operating licence. **[slide 5]**

The present day Radiation Protection Regulations are prefaced by a set of RP Policies and Principles[2] including the following commitment to:

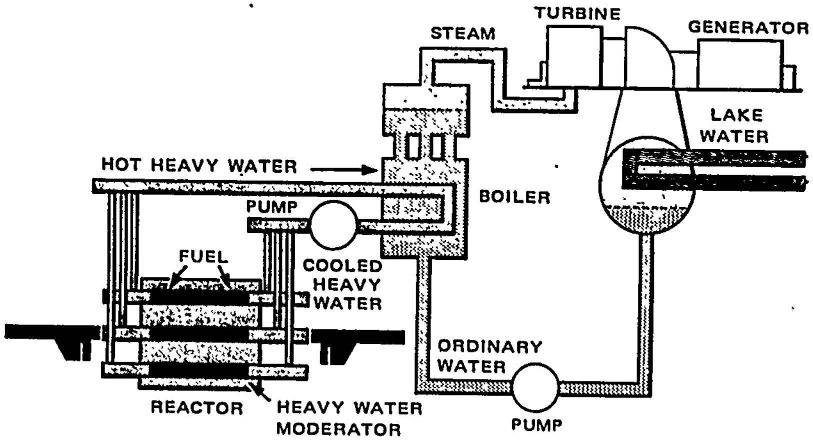

FIGURE 1 CANDU FLOW DIAGRAM

CORPORATE DOSE TOTALS
(1963 - 1993)

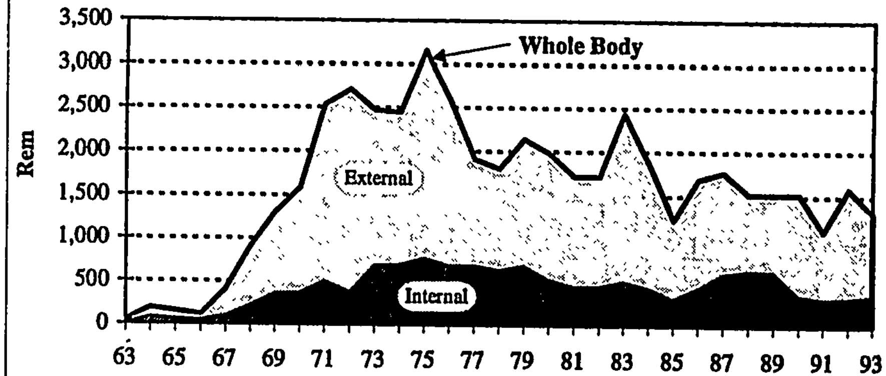

Figure 2

AVERAGE DOSE TO AN INDIVIDUAL
(1962-1993)

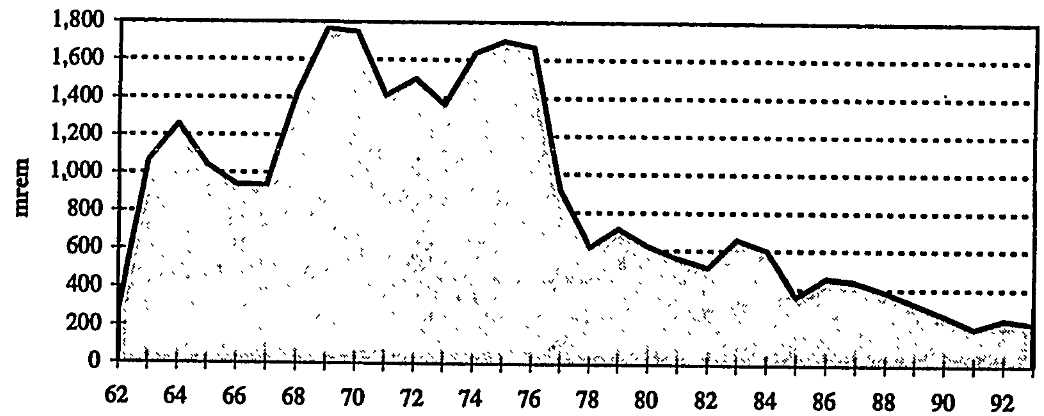

Figure 3

Limit detrimental stochastic health effects occurring in employees or members of the public to levels as low as reasonably achievable, social and economic factors being taken into account (ALARA).

This commitment is very real and is a primary driving force for many of the subsidiary RP programs in the design and operation of our plants. A tangible example of this commitment is the inclusion of collective dose targets for Ontario Hydro Nuclear as a whole and for individual stations into the performance contracts of nuclear executives. In short, *what gets measured gets done.*

ALARA IN DESIGN AND REHABILITATION

Design

Ontario Hydro recognized very early in its nuclear lifetime based on experience at the 208 MWe station known as Douglas Point that significant changes were required in the design of new stations to control radiation exposure. Specifically, it became obvious early on (late 1960's) that one of the major causes of high dose rates around reactor components was the presence of Co-60 due to activation of Co-59. It was too late to reduce Cobalt content in the design and construction of the Pickering A station but replacement components and later CANDU stations at home and abroad benefitted from this realization. This slide **[slide 6]** shows how Cobalt levels have been reduced over the years starting with the Bruce A station in the mid-70's.

In general, the feedback from Operations personnel to the Ontario Hydro designers has improved continuously over the years. In the early 1980's, the design organization established an occupational radiation safety engineering program in which a small group, with significant operational experience as Health Physicists and Engineers were fully integrated into the design process with a specific mandate to ensure that RP considerations were built into new system designs and modifications from the concept stage through to construction and operation. This kind of resource commitment is another example of management commitment.

This process had its greatest application in the design of the new Darlington Station (4 x 881 MWe) using a program known as the Occupational Radiation Management Program (ORMP), the objectives of which are to: **[slide 7]**

* emphasize occupational radiation dose as a parameter in the design process
* establish an occupational radiation dose target for the station in its design stage and to break this dose down into individual system design dose targets
* verify that the design achieves the dose targets through a four stage radiation management review process
* identify the normal operational, maintenance and inspection activities implicit in the station design that could be expected to involve significant dose expenditure and to estimate dose expenditures
* provide direction to designer's efforts to reduce the dose and to describe the methods available for achieving dose reduction in the design stage

The radiation management review process consists of four stages as described below for all relevant engineered systems:

Stage 1

In this stage, gross dose estimates are developed based on design data and prior operational exposure history at other stations. Areas are then identified where dose reduction can be achieved using techniques such as: **[slide 8]**

- elimination of equipment
- simplification and orientation of system components
- provision of adequate space for maintenance
- relocation of equipment into lower dose rate areas
- chemical control and purification of active systems
- extension of time intervals between scheduled maintenance
- use of radiation resistant materials
- provision of the means for quick removal of components for maintenance in low dose rate areas (eg shops)
- reduction in the amount of time for in situ maintenance including the use of special tools
- increased use of shielding

Stage 2

In this stage, detailed dose estimates are made accounting for operation, maintenance and in-service inspections. The designs are then checked for simplification, reliability, reduction and ease of maintenance to optimize dose reduction. Included in this stage of the process are the system design engineer, operations staff, the station Health Physicist, shielding specialists, the station layout coordinator, reliability and maintainability specialists and the design radiation safety specialists.

Stage 3

This stage consists of a followup on the decisions made in Stage 2 and an evaluation of the requirements of layout change and/or supplementary shielding to offset any problems encountered in Stage 2. A subset of the group involved in Stage 2 contribute to Stage 3.

Stage 4

In this stage, design changes resulting from the previous stages are confirmed to have been implemented and that the resultant changes to dose estimates have been accounted for. Where dose targets cannot be met, further consideration is given to dose reduction measures.

Consider application of this process to the Main Moderator Circuit at Darlington for example. The Moderator System is a low temperature, low pressure system whose primary purpose is to moderate fission neutrons in the reactor core. The core portion of the system is housed in a large steel tank known as the calandria. As the moderator is exposed to high flux, it is a source of activation products including Co-60, N-16 and tritium. Annual dose targets for the system (per reactor unit) were set in Stage 2 of the ORMP at 30 rem external (gamma + neutron) dose and 10 rem internal (tritium) dose. Some of the design requirements (dose reduction measures) that were established were that: **[slide 9]**

* all piping in contact with heavy water is stainless steel with low cobalt content (< 0.1% by weight)

* all welds to be butt welded

* all rooms to be connected to the confinement ventilation system (ie dried to remove airborne tritium)

* all penetrations have been checked for adequate gamma and neutron shielding

More specific measures are then applied to each part of the circuit.

Overall, for the total four unit station, the initial Stage 1 dose estimates were about 1570 rem per year for maintenance, inspection, operations and some allowance for contingency. At the end of Stage 4, this had been reduced to 550 rem per year, a reduction by about a factor of 3 due to dose reduction measures in the design stage.

REHABILITATION

Replacement of pressure tubes (the core of the Heat Transport (coolant) System) was always anticipated to be required at some point in the life cycle in CANDU plants. For Pickering A, this happened somewhat sooner than planned after discovery of a significant pressure tube leak (loss of coolant) in 1983 **[slide 10]**. The cause was ultimately determined to be delayed hydride cracking. As a result, a decision was made at that time to replace all pressure tubes (390 per unit) in all 4 units at the Pickering A station. The actual work commenced in 1985 on Unit 1 and terminated in 1992 with the completion of retubing work on Unit 4. This presented a major technical challenge since much of this work was to be done in high radiation fields. For a CANDU reactor, this is essentially a "heart transplant" conducted in a somewhat hostile atmosphere. The overall project is conducted in several phases including defuelling of the core, tube/component removal, inspections, new tube installation and waste removal. Since this involves highly activated components, it was recognized from the outset that a well planned RP program was required to maintain exposures ALARA. One of the first steps was to assign an experienced operational Health Physicist to the project team.

Again, integration of RP considerations with other engineering activities was a distinct advantage from the outset. Management commitment to maintaining doses to levels ALARA was crucial and consistent with corporate objectives for radiation safety.

This project involved a large volume of work in relatively high beams of gamma radiation and high general gamma fields. There were many elements of the RP program that contributed towards minimizing occupational exposure but some of the more significant ones were[3,4]: **[slide 11]**

* establishment of individual and collective dose targets at an early stage

* construction of a full scale mock-up facility on site of the reactor face and Fuelling Machine Vault (the containment). The facility was used to develop and test specialized tooling and train personnel in a nonradiological environment.

* training of multiple crews for a job series who worked as a team during training on a full scale mock-up and remained intact for the actual job execution.

* decontamination of the primary system (Heat Transport System) prior to retubing work using the Can-Decon process. Overall, this reduced radiation fields by a factor of between 5 and 6. The cost of this procedure has been estimated at roughly $4400 (Can) per rem saved for the overall retubing project[4] without consideration of future dose savings.

* design of a special 3 sided steel shielding cabinet behind which most of the reactor face work was done. A 1/4 inch thickness of lead was added to the front of the cabinet during the retubing of Units 3 and 4.

* extensive use of personal alarming dosimeters to prevent unplanned exposures

* an effective radiation safety organization and work planning process.

The collective dose for the entire project on all 4 units was approximately 1390 person rem, about 23% below the target of 1800 person rem [slide 12]. For units 1 and 2, actual collective doses had to be estimated since the appropriate mechanisms were not in place to distinguish between dose received from retubing activities and other sources in the station.

Individual doses were constrained by the following criteria:

* dose to be equalized to within +/-25% for equivalent trades/technical staff exceeding 500 mrem per year

* management and engineering review to be initiated for an individual dose total to exceed 1.3 rem/quarter or 3.0 rem/year

* 5 year rolling average to be below 2 rem/year

Typical average doses to the most exposed workers (those at the reactor face) throughout the project were 900 mrem per year and no individual exceeded regulatory dose limits of 5 rem per year or 3 rem per quarter.

ELIMINATION OF RADIOACTIVE SOURCES

A unique radiological hazard of heavy water reactors is tritium in the form of tritium oxide as water or water vapour. Equilibrium concentrations of tritium theoretically approach 60-80 curies/kg in the Moderator System. In some years, tritium dose has been as high as 50% of a station's whole body dose. As tritium concentrations in CANDU Moderator and Heat Transport Systems grew in the 1970's, there was increasing concern that tritium exposure to our workers would be difficult to maintain at acceptable levels without addressing the source term. As a result, a decision was made in the early 1980's to build a tritium removal facility (TRF) to service the needs of all Ontario Hydro stations. That plant has been built on the Darlington site at a capital cost of roughly $120 M (Can) and first started up in 1988 and achieved sustained production in 1990. [slide 13]

The objectives of tritium removal are to:

* reduce occupational dose from tritium exposure
* reduce public exposure as a result of tritium emissions

A secondary objective was to exploit the commercial value of tritium for non-military uses. The process basically consists of catalytic exchange of tritium in the vapour phase followed by cryogenic distillation to separate the hydrogen isotopes. The end product (> 99% gaseous tritium) is immobilized on titanium getter beds for long term storage. To date, less than 1% of stored tritium has been sold for commercial uses such as the tritium lighting industry.

The designed decontamination factor is 35 for the tritium removal process (once through). So far, treatment has primarily focused on removal of tritium from Bruce and Pickering Moderator System. More recently, some water has also been processed for the Point Lepreau Station in New Brunswick. Approximately 90 million curies of tritium (6500 Mg of heavy water) have been removed and immobilized to date.

Not only does this technology provide opportunities for reducing chronic dose consumption from tritium but also reduces the consequences of acute events such as heavy water spills. A Moderator spill, for example, can result in several thousand MPCa (DACs) of tritium in the air above the spill. Normally, our workers are dressed in air supplied plastic suits when working on the Moderator System which provides excellent protection against inhalation and absorption through the skin of tritiated water. However, if unprotected, a worker could receive a dose in excess of regulatory limits in a matter of minutes at such concentrations.

Although collective tritium dose corporate-wide has decreased in recent years, this is partly an artifact. Purely by coincidence, the startup of the TRF occurred at about the same time as a regulated change in the dosimetric model for tritium which has the effect of reducing the dose commitment by 27% for the same intake[5]. We will need to carefully monitor our tritium dose consumption in the years ahead to ensure that we are receiving good return on our investment in tritium removal.

TRAINING OF PERSONNEL/SELF PROTECTION PHILOSOPHY

Extensive radiation protection training of operating and maintenance personnel as well as supervisors, managers and engineering staff has been a cornerstone of Ontario Hydro's RP program from the beginning and is an element of the ALARA program that transcends all phases of project life cycle. **[slide 14]** This commitment to the self protection concept has no doubt been a major contributor to our performance in radiation protection. The overall program typically extends over a period of a few weeks with both generic and station specific components. The operating budget corporate wide for the radiation protection training program is of the order of $3 M (Can) annually. These are direct costs only, ie, it does not account for lost production time for the students which is of course significant. The program includes a mixture of classroom and skills training in such areas as: **[slide 15]**

* Radiological hazards and hazard levels associated with specific reactor systems

* Use of protective equipment and instrumentation

* Dosimetry

* Biological effects of radiation and risk

* Radiological work planning including contingency planning

* Emergency procedures

More advanced training is given to Shift Supervisors and Control Room Operators, in part to prepare them for RP examinations set by the regulator which are part of the licensing process for such staff.

To quote ICRP 55[6], **[slide 16]** "*The basic role of the concept of optimization of protection is to engender a state of thinking in everyone responsible for control of radiation exposures such that they are continuously asking themselves the question, Have I done all that I reasonably can to reduce these radiation exposures?*"

"Everyone responsible for control of radiation exposures" includes many people in our organization including the employees on the front lines doing the hands on work and their first line supervision. Some of the advantages to the self protection philosophy include: **[slide 17]**

* employees see radiation protection as a joint responsibility

* employees are able to integrate RP responsibilities directly into their work

* the individuals performing the work become a valuable source of information for future improvements to the radiological aspects of the work

* there is a reduction in the number of people required to do a task (normally no HP techs required)

Overall, this approach has worked well although clearly there are associated costs and problems. In our corporate culture, we have created an army of amateur Health Physicists in the field with the result that RP professionals and line supervisors are constantly challenged by employees and their representatives on the essential elements of ALARA application, ie what constitutes "low", "reasonable" and "achievable". Our labour unions, particularly, have become very active in the management of dose control and reduction largely through representation on joint committees at both the station and corporate level. This applies to both policy development and execution of work practices in the field. Although difficult to quantify, it is likely that labour influences often result in policies and programs which are not cost justified, ie go beyond reasonable efforts to reduce dose and minimize risk but these costs must be balanced against the benefits of having a radiologically informed workforce. Making these judgements on a day to day basis is more art than science.

GOALS AND TARGETS

A number of times in this paper, reference has been made to various collective and individual dose targets. Establishing such targets and integrating their use into the managed system is an important part of any ALARA program. To repeat, *what gets measured gets done.* In establishing such targets, it is essential that they are:

* challenging but attainable
* stated in measurable terms
* consistent with corporate initiatives and policy
* accepted and embraced by those directly responsible for achieving them

There must be real ownership for dose targets by facility management just as for production, reliability and cost targets. Dose targets must not be viewed as belonging to the RP Manager, ALARA Engineer or Health Physicist even though they may be responsible for deriving them.

In Ontario Hydro, dose targets are set on a broad basis by policy for the overall nuclear program for both collective and individual dose. Collective dose targets are pyramided up to the corporate level by summing contributions from each station or nuclear facility based on a detailed analysis of annual operating plans and outage schedules. Within those targets are targets for specific jobs and shift crews. It is important that adequate information systems exist to monitor and report performance in a timely manner.

Ontario Hydro Nuclear policy on exposure management[7] contains the following explicit limits. **[slide 18]**

Collective Dose Standard

≤ 85 rem/unit/year

In essence, this is intended as a long term average target for any given facility who are required to develop annual targets based on analysis of workload and diligent application of the ALARA principle.

Internal dose (tritium) ≤ 25% of collective dose

Individual Dose Limits

- 5 year dose limit - total dose averaged over 5 years ≤ 5.0 rem
- single year dose limit ≤ 2.0 rem or ≤ 1.0 rem if lifetime dose ≥ 50 rem

INFORMATION MANAGEMENT

The traditional technical tools of the RP specialists will continue to be challenged in the 1990's as dose limits and RP standards tighten and available dollars shrink. But perhaps the most exciting opportunities for further progress lie in the field of information management with the advent of personal computers and local area networks. As an example, for many years we have used a manual "Radiological Log" in our plants as a tool for all staff to record information on plant radiological conditions and events. This log is typically maintained in the main control room and reviewed by Operators and tradespeople prior to entering the field to do work. This system is gradually being replaced by computerized logs which will provide greater access to current and historical information at a number of locations throughout the plant.

Many other examples exist in dose control and work planning (eg, computerized Radiation Exposure Permits) all of which will ultimately contribute to dose reduction. The gap is also closing between dose control and dosimetry (dosimeter of record) with the emergence of electronic dosimetry. In Ontario Hydro, we are actively developing future strategy for our external dosimetry system and electronic dosimetry is certainly one of the options being seriously considered. We have been sponsoring and participating in testing of one such device for some time. Its clear advantage lies in providing real time data at the worksite. However, there are many issues to resolve yet in the use of this exciting new technology as a dosimeter of record including quality assurance, cost, user acceptance and regulatory approval.

SUMMARY

Major progress has been made over the past three decades in control of occupational radiation exposure in the Ontario Hydro nuclear program **[slide 19]**. This has resulted from the collective efforts of workers, supervisors, designers, RP specialists and has been driven by senior management commitment to the ALARA principle. Investment in the self protection philosophy has also been a major contributor by facilitating integration of RP considerations into all aspects of the work programs.

Economic pressures will provide further challenges in the future to maintain this performance while remaining cost competitive. It is likely that fully exploiting the advantages offered by information management will play a crucial role in meeting this challenge.

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Author Biography

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ALARA IMPLEMENTATION

THROUGHOUT PROJECT LIFE CYCLE

M.J. HAYNES
ONTARIO HYDRO

Slide 1

ONTARIO HYDRO'S NUCLEAR FACILITIES

- Bruce Nuclear Power Development
  Bruce NGS A 4 x 848 MW(e) net
  Bruce NGS B 4 x 860 MW(e) net
- Pickering
  Pickering NGS A 4 x 515 MW(e) net
  Pickering NGS B 4 x 516 MW(e) net
- Darlington NGS 4 x 900 MW(e) net
- Total Nuclear Capacity 14,556 MW(e) net

Slide 2

Year	1970	1993
Installed Capacity (MWe)	240	14,400
Collective Dose (person rem)	1585	1307
Rem/MWe	6.6	0.09

Slide 3

MAJOR CONTRIBUTORS TO ONTARIO HYDRO ALARA PROGRAM

- Management commitment/corporate policy
- Integration of operating experience with design and rehabilitation
- Elimination of radioactive sources
- RP training program/self protection philosophy
- Use of goals and targets
- Information management

Slide 4

ONTARIO HYDRO
RADIATION PROTECTION REGULATIONS

1962 - Principle of Minimum Exposure

"All doses shall be kept as low as is practicable and unnecessary exposure shall be avoided."

1993 - Objectives of the RP Program

"Limit detrimental stochastic health effects occurring in employees or members of the public to levels as low as reasonably achievable, social and economic factors being taken into account."

Slide 5

COBALT CONTENT
IN CANDU STEAM GENERATOR TUBES

STATION	MATERIAL	COBALT 59 CONTENT (%)
Douglas Point	Monel	0.1 - 0.15
Pickering A	Monel	0.07
Bruce A - 1,2	Inconel	0.024
Bruce A - 3,4	Inconel	0.014
Bruce B	Inconel	0.015
Pickering B	Monel	0.005
Darlington	Incalloy 800	0.015

Slide 6

ALARA IN DESIGN I
OCCUPATIONAL RADIATION MANAGEMENT PROGRAM

Objectives are to:

- **emphasize radiation dose as a parameter in design process**
- **establish system specific dose targets in design stage**
- **verify that design achieves targets through a 4 stage management review process**
- **targets include operations, maintenance and inspection**
- **provide direction to designers for dose reduction**

Slide 7

ALARA IN DESIGN II
DOSE REDUCTION METHODS

- **elimination of equipment**
- **adequate space for maintenance**
- **relocation into lower dose rate areas**
- **chemical control and purification**
- **extension of intervals between scheduled maintenance**
- **use of radiation resistant materials**
- **reduction in maintenance time in situ**
- **increased shielding**

Slide 8

ALARA IN DESIGN III
APPLICATION OF ORMP TO
MAIN MODERATOR SYSTEM - DNGS

- all piping in contact with heavy water to be stainless steel with < 0.1% cobalt
- all welds to be butt welded
- all rooms to be connected to confinement ventilation system
- all penetrations checked for adequate gamma and neutron shielding

Slide 9

LARGE SCALE
FUEL CHANNEL REPLACEMENT (RETUBE):

- CANDU Rehabilitation Program
- High Dose Expenditure
- High Risk

Slide 10

LSFCR
DOSE REDUCTION MEASURES

- establishment of dose targets at an early stage
- full scale mock-up facility
- training of multiple crews
- decontamination of primary system (CANDECON)
- shielding cabinet
- extensive use of PADs
- radiation safety organization

Slide 11

LSFCR
DOSE RESULTS

<u>Collective Dose</u>

- 1400 person rem vs 1800 planned for all 4 units (over ~7 yrs)

<u>Individual Dose</u>

- constraint - < 2 rem/year on 5 year rolling average basis
- average annual dose for "most exposed" workers ~900 mrem
- no regulatory limits exceeded

Slide 12

TRITIUM REMOVAL FACILITY (TRF)

- located on Darlington site
- reliable operation began in 1989
- objectives are to: a) reduce occupational exposure
  b) reduce public exposure
- secondary objective is to sell product for non-military uses
- to date, 93 MCi removed primarily from Bruce and Pickering
- < 1% has been sold

Slide 13

RADIATION PROTECTION TRAINING/ SELF PROTECTION I

- cornerstone of our RP Program is self protection philosophy
- all OH workers, supervisors and managers receive extensive training in RP
- program includes generic and station specific elements

Slide 14

RADIATION PROTECTION TRAINING

- RPT program covers
  - hazards and hazard levels
  - protective equipment/instrumentation
  - dosimetry
  - biological effects/risk
  - rad work planning
  - emergency procedures

Slide 15

RADIATION PROTECTION TRAINING/ SELF PROTECTION II

<u>ICRP 55</u>

The basic role of the concept of optimization of protection is to engender a state of thinking in everyone responsible for control of radiation exposures such that they are continually asking themselves the question, "Have I done all that I reasonably can to reduce these radiation exposures?".

Slide 16

RADIATION PROTECTION TRAINING/ SELF PROTECTION III

Advantages of Self Protection

- employees see RP as a joint responsibility
- employees are able to integrate RP directly into their work
- workers become a valuable source of information for future radiological work
- reduction in the number of people required to do a task

Slide 17

ONTARIO HYDRO NUCLEAR DOSE TARGETS

Collective Dose Standard

$\leq$ 85 person rem/unit/year

Internal dose $\leq$ 25% of collective dose

Individual Dose Targets

- Total dose averaged over 5 years $\leq$ 5.0 rem
- $\leq$ 2.0 rem per year if lifetime dose < 50 rem
- $\leq$ 1.0 rem per year if lifetime dose $\geq$ 50 rem

Slide 18

SUMMARY

KEY FACTORS IN DOSE REDUCTION

1. Management Commitment
2. Reduce the Source Term
   - Cobalt 60, Tritium
3. Integrate RP with other work programs at an early stage
4. "Empower" the workforce
5. Goals and Targets - "*What gets measured gets done*"
6. Exploit Information Management

Slide 19

HEALTH PHYSICS ASPECTS OF ADVANCED REACTOR LICENSING REVIEWS

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ABSTRACT

The last Construction Permit to be issued by the U.S. Nuclear Regulatory Commission (NRC) for a U.S. light water reactor (LWR) was granted in the late 1970s. In 1989 the NRC issued 10 CFR Part 52[1] which is intended to serve as a framework for the licensing of future reactor designs. The NRC is currently reviewing four different future or "next-generation" reactor designs. Two of these designs are classified as evolutionary designs (modified versions of current generation LWRs) and two are advanced designs (reactors incorporating simplified designs and passive means for accident mitigation). These "next-generation" reactor designs incorporate many innovative design features which are intended to maintain personnel doses ALARA and ensure that the annual average collective dose at these reactors does not exceed 100 person-rems (1 person-sievert) per year. This paper discusses some of the ALARA design features which are incorporated in the four "next-generation" reactor designs currently being reviewed by the NRC.

INTRODUCTION

The NRC has been actively reviewing "next-generation" reactor designs since the late 1980s. The term "next-generation" encompasses both evolutionary LWR designs and advanced reactor designs. Evolutionary reactor designs are essentially modified versions of current generation LWR designs. These reactors utilize conventional safety system concepts. Advanced reactor designs include passive and non-LWR reactor designs. Passive reactors employ greatly simplified designs, generally range from 300 to 600 MWe in size, and utilize passive means for accident prevention and mitigation.

All of these "next-generation" reactor designs incorporate lessons learned from currently operating LWRs. Many of the features in these new plants, such as standardization, simplified plant design, and modularization, will result in plants that will be easier to operate and maintain. This, in turn, will result in lower collective doses. Careful attention to material selection, such as the use of low cobalt- and nickel-based alloys in the primary coolant system, will also help to lower collective doses by reducing overall plant radiation levels.

BACKGROUND

Although the NRC has strongly encouraged the standardization of nuclear reactor designs for many years, it was the issuance of 10 CFR Part 52 (known as the Standardization Rule) in 1989 which served as the framework for consideration of future designs. The three parts of this Standardization Rule provide for the issuance of 1) early site permits, 2) standard design certifications, and 3) a combined construction permit and operating license. This rule is designed to streamline the reactor licensing process. The standard nuclear power plant final design approval which results from this licensing review is acceptable for incorporation into individual facility license applications.

CURRENT PLANT DESIGN REVIEWS

There are four "next-generation" reactor designs currently under review within the NRC. Two evolutionary LWR designs, General Electric's (GE) Advanced Boiling Water Reactor (ABWR)[2] and Combustion Engineering's System 80+ Standard Design,[3] are in the final stages of design certification. Two passive LWR designs, Westinghouse's AP-600[4] and GE's Simplified BWR (SBWR),[5] are in the early stages of staff review (Table 1). In addition to these ongoing plant design reviews, the NRC has completed its review of EPRI's Advanced Light Water Reactor (ALWR) Requirements Document.[6] The purpose of this document is to specify industry approved design criteria for evolutionary and passive ALWR standard plants. The design features described are those features that both utilities and industry would like to see incorporated into the next generation of nuclear power plants.

Table 1. "Next-Generation" Reactor Designs Currently Under NRC Review

Evolutionary	Passive
GE Advanced BWR (ABWR)	GE Simplified BWR (SBWR)
CE System 80+	Westinghouse AP-600

One of the objectives contained in EPRI's ALWR Requirements Document is to design a nuclear power plant that can operate with an average dose of 100 person rem (1 person-sievert) per year or less. The estimated annual doses for the four "next-generation" reactor designs currently under review range from 68 person-rem (6.8×10^{-1} person-sievert) for Westinghouse's AP-600, to 99 person-rem (9.9×10^{-1} person-sievert) for GE's ABWR. In contrast, the average annual dose for U.S. LWRs averaged 266 person rem (2.66 person-sievert) per reactor in 1992.[7] This average dose, however, represents a 62 percent drop from the U.S. average annual dose of 705 person-rem (7.05 person-sievert) per reactor just a decade earlier (see Figure 1).

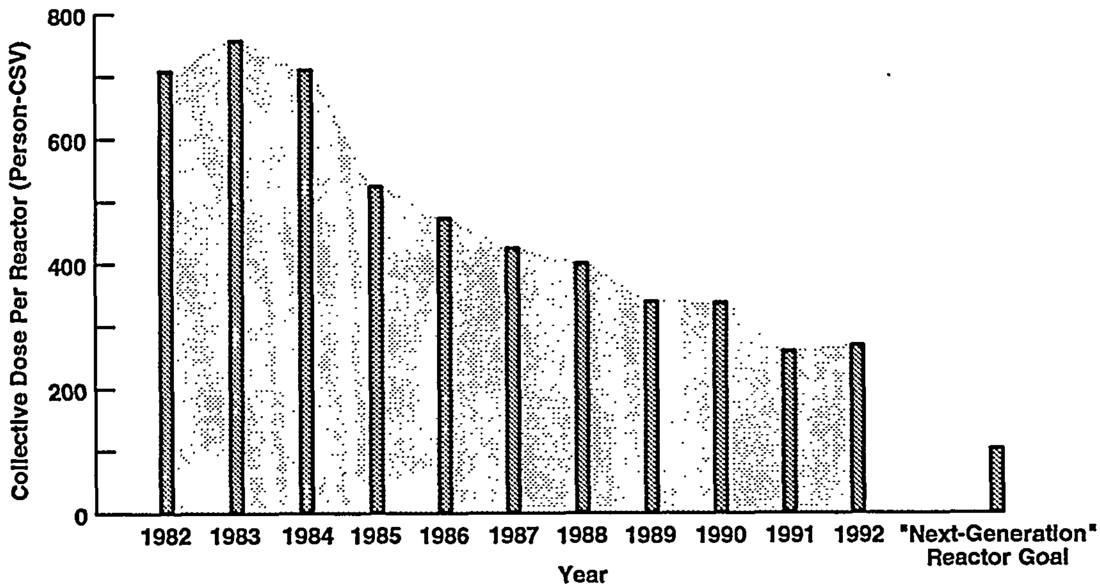

Figure 1. Collective Doses at U.S. LWRs and "Next-Generation" Reactor Dose Goal

In order to achieve an average annual dose of 100 person-rem (1 person-sievert) or less, the "next-generation" reactors will incorporate a number of ALARA design features. Some of these design features are based on the ALARA guidance provided in Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Is Reasonably Achievable." However, many of the ALARA design features utilized by the "next-generation" reactors are based on lessons learned from the current generator of operating reactors and employ the use of advanced technology.

Some of the ALARA design features described below will apply to all four of the "next-generation" reactor designs currently under review by the NRC. Others will apply only to designs by specific vendors or to a type of reactor design (i.e., evolutionary versus passive).

Material Changes

The primary source of radiation fields in nuclear power plants is cobalt-60. Cobalt is the major constituent of Stellite, a hardfacing material used in valve seats, pump journals, and other wear resistant components. The "next-generation" reactor designs will restrict the use of high cobalt alloys such as Stellite to those applications where no satisfactory alternate material is available. Where possible, the cobalt content of piping and other equipment in direct contact with the reactor coolant will be restricted to 0.05 wt %. The inconel steam generator tubes will contain no more than 0.015 wt % cobalt and will be fabricated to relieve stresses to reduce stress corrosion cracking. Main condenser tubes and tube sheets will be made of titanium alloys to minimize condenser tube leakage, thereby reducing the introduction of foreign materials (which can become activated) into the reactor system. The presence of antimony in reactor coolant pump (RCP) journal bearings has resulted in an increase in the number of hot particles at some current generation plants.[8] The RCP journal bearings in CE's System 80+ design will be designed to minimize the presence of antimony.

Component Design Features

The "next-generation" reactors will incorporate several innovative component design features to reduce area dose rates and minimize personnel doses. External recirculation pumps and recirculation piping were replaced by internally mounted recirculation pumps in GE's ABWR design. GE's SBWR design contains no recirculation pumps or recirculation piping since this plant employs natural recirculation as a motive force. In both designs, elimination of recirculation pumps and piping removes a major source of radiation in the lower drywell and should reduce general area dose rates in the drywell by 50 percent. The SBWR's simplified design will require significantly fewer safety relief and other valves in the drywell, thereby requiring less maintenance time to service these valves. The SBWR turbine design will utilize cross-over lines, thereby eliminating the need for moisture separators and reheaters and removing the most significant source of skyshine and turbine building operational radiation.

Both the CE System 80+ and the AP-600 designs will incorporate an integrated reactor head removal package. This package will facilitate reactor head removal and replacement during refuelings, resulting in lower personnel doses and less manpower requirements. Refueling doses for the GE BWR designs will be reduced through the use of a stud tensioner and an automated refueling bridge. The use of the automated refueling bridge, where no personnel are located on the platform itself, will cut refueling time in half and reduce the effective dose rate by a factor of ten.

The AP-600 steam generator design will include a sludge control system/mud drum which is designed to reduce the need for sludge lancing, and reduces tube and tube support degradation. The tube ends in this steam generator are designed to be flush with the tube sheet in the steam generator channel head to eliminate potential crud traps.

A system which has resulted in several overexposures in current generation LWRs is the Transversing In-Core Probe (TIP) system in BWRs and incore instrumentation in PWRs. GE's ABWR provides a shielded room for the TIP drive units. Automatic logic control and mechanical stops prevent the TIP or activated portions of the TIP cable from being withdrawn into the drive housings. GE's SBWR design eliminates the TIP system by using fixed in-core detectors. The CE System 80+ design prevents access to the reactor cavity housing the incore instrumentation chase by providing a posted and locked access door connected by electrical interlock to an area radiation monitor located in the reactor cavity. When the incore instrumentation are withdrawn from the reactor core, a warning light on the access door illuminates and the interlock prevents the access door from being opened.

Other component design features include the use of canned pumps in the Residual Heat Removal System and Reactor Water Cleanup System of the GE designs to minimize maintenance requirements. The RCPs in the CE System 80+ design will incorporate a cartridge type of RCP seal which is reliable and easily replaceable. The RHR heat exchangers in GE's ABWR and Westinghouse's AP-600 designs are designed with an excess of tubes in order to permit plugging of some tubes without losing system efficiency. The heat exchangers also are provided with drains to allow drainage of the shell-side water prior to maintenance. The CE System 80+ design will minimize the use of evaporators. Evaporators have historically required frequent maintenance and contributed to high personnel exposures. The CE design will also utilize mechanical snubbers rather than hydraulic snubbers in radiation areas to reduce maintenance and inspection needs. Liquid systems containing radioactive cartridge filters in the AP-600 design will be provided with a remote filter handling system for changeout and transfer to the drumming station of spent radioactive filter cartridges. In order to prevent the migration of noble gases and other airborne radionuclides between floors in CE's System 80+ design, floor drains connecting rooms that have significantly different airborne radioactivity levels will be separated or provided with water-filled loop seals to prevent cross-contamination.

Features To Facilitate Maintenance

The equipment selected for the "next-generation" reactor designs will have enhanced reliability and will be designed for low maintenance. These designs will make more use of modular components which can be easily replaced or removed to a lower radiation area for repair. The AP-600 design will have RCPs which can be unbolted for quick removal to a low radiation background work area for maintenance or replacement using a specially provided pump removal cart. The control rod drive (CRD) system in the GE ABWR and SBWR designs will have an internal CRD restraint feature which will facilitate CRD removal. This feature will result in lower radiation exposures than those seen in current generation BWRs, which have external CRD restraints. The lower drywell design in the GE design reactors will allow easy access to the lower reactor vessel head for CRD and reactor internal pump removal. A transport system will permit removal of these components to a lower radiation area.

Radioactive systems and components will be provided with taps for flushing with condensate or for chemically cleaning to reduce crud buildup and lower radiation levels prior to maintenance. Rooms housing these components will have epoxy-type floor and wall coverings to facilitate decontamination. Equipment and floor drain sumps will be stainless steel lined for ease of cleaning and to reduce crud buildup.

The "next-generation" reactors will be designed to facilitate accessibility to plant equipment. Adequate work and laydown space will be provided around components for maintenance purposes. In order to facilitate maintenance and improve worker efficiency, adequate illumination and support services (e.g., power, service air, water, ventilation, and communications) will be available at work stations. In the event that maintenance cannot be performed in-situ, rigging and lifting equipment will be provided to facilitate the removal, transport, or replacement of equipment (this rigging equipment can also be used for the installation of portable shielding).

Features to Facilitate In-Service Inspection

Approximately nine percent of the annual dose at U.S. LWRs can be attributed to in-service inspection work. The "next-generation" reactor designs will facilitate in-service inspection by making plant components more accessible and relying more on the use of robotics.

The CE System 80+ design will include permanent platforms around major equipment such as the steam generators and reactor coolant pumps. These platforms will facilitate access to these components for maintenance and in-service inspection, and will serve to reduce the overall plant collective dose by eliminating the need to erect temporary scaffolding around these components for maintenance/inspection purposes. In GE's ABWR and SBWR designs, permanent steel platform will be provided for in-service inspection of the reactor pressure vessel nozzle welds and associated piping. These steel platforms will also serve to provide shielding for inspection personnel from adjacent radiation sources.

All of the "next-generation" designs will utilize easily removable blanket or mirror type thermal insulation around piping and components. The sections of the reactor vessel insulation in the area of the reactor vessel nozzle welds for the AP-600 will have permanent I.D. markings to accommodate rapid reinstallation. GE's ABWR and SBWR designs will incorporate specific access panels and shield doors into required inspection areas permitting easy bypass of insulation areas and thereby reducing inspection time.

In order to reduce the inservice inspection time required for welds, all four of the "next generation" reactor designs under review will have forged ring instead of plate welded pressure vessels. Because forged ring pressure vessels have fewer welds, the total vessel weld length inspection will be reduced by 30 percent. The use of seamless piping in all "next-generation" reactor designs will reduce the amount of piping welds.

The reactor vessel nozzle welds in the AP-600 and CE System 80+ designs will be designed to accommodate remote inspection using ultrasonic sensors. The use of automated equipment for weld inspections in the GE ABWR and SBWR designs will reduce the required inspection manhours by a factor of two. The CE System 80+ design will utilize robotics, whenever practical, to perform maintenance and inspection activities such as remote pipe welds and inspections in high radiation areas. The steam generators in both the CE System 80+ and AP-600 plant designs will be designed to use automatic/robotic equipment for inspection and maintenance activities. In addition to having larger diameter manways to facilitate personnel access and the installation and removal of tooling, these steam generators will have an increased number of handholes and will be provided with platforms and adequate pull and laydown areas for inspection and maintenance purposes.

Plant Layout Features

The plant layouts for the "next-generation" reactor designs will be designed to maintain personnel exposures ALARA during normal and post-accident conditions. Radioactive systems will be separated from non-radioactive systems. Pipes or ducts carrying radioactive sources will not be routed through occupied areas. Redundant radioactive components will be separated and shielded from each other to permit maintenance on one component without being exposed to radiation from the other component. Labyrinth entrances will be provided to radioactive pump, equipment, and valve rooms. These labyrinth entrances will have sufficient space for easy access and for equipment removal. Adequate space will be provided for the storage and erection of temporary shielding.

Ion exchangers in the CE System 80+ design will be located in pits with the spent resin tanks located below the ion exchanger. This design will provide shielding around the spent resin tanks and will lower the dose rates to personnel working on other equipment in the area. The spent fuel transfer tube in the AP-600 design does not have a seismic gap and therefore will be completely enclosed in concrete. This design results in a spent fuel transfer tube which is shielded its entire length and eliminates the potential for personnel overexposures during refueling operations caused when spent fuel is transported through unshielded portions of the spent fuel transfer tube (this is a shortcoming and potential problem with conventional PWR designs).

Plant layout features will be designed to facilitate maintenance operations and minimize personnel dose. Adequate rigging and lifting equipment will be provided, where needed, to assist in equipment removal/replacement. The use of overhead tracks and in-place removal equipment (in the GE ABWR and SBWR plant designs) to transfer safety/relief valves, reactor internal pumps, and other valves in the drywell, will result in an estimated savings of 300 person-hours per year. The CE System 80+ design will provide for large staging areas both inside and outside the reactor building equipment hatch and personnel airlocks. This will allow for pre-staging prior to the start of an outage and the location of these staging areas will provide for efficient radiation controls and will minimize the potential for the spread of contamination. Hot tool cribs will be located in low radiation areas adjacent to maintenance areas to minimize waiting times in high radiation areas and to prevent the spread of contamination. The hot machine shop is located in a low radiation area adjacent to the equipment hatch to permit maintenance to be performed on equipment removed from containment in a lower radiation area. Dedicated change out areas are also located near airlocks in low dose areas to minimize personnel traffic flow and the potential for the spread of contamination.

CONCLUSION

The four "next-generation" reactor designs currently under review by the NRC all contain a number of innovative ALARA design features. Some of these features are simply modifications of ALARA design features used in currently operating U.S. LWRs. Others are based on design features used in foreign LWR designs. Still other design features, such as some of the features used by the passive design LWRs, will be used for the first time in the "next-generation" reactor designs. The use of these innovative ALARA design features, along with the minimization of cobalt and nickel in reactor coolant system components, should permit these "next-generation" reactors to operate within their estimated dose goals of 100 person-rem (1 person-sievert) per year.

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Author Biography

Charles Hinson is a health physicist in the Radiation Protection Branch, NRR at the U.S. Nuclear Regulatory Commission. Over the last three and a half years, Mr. Hinson has served as the health physics reviewer for the EPRI Utility Requirements Document review and the CE System 80+ and GE SBWR design reviews. Other current responsibilities include analysis of occupational exposure data for U.S. reactors and evaluation of radiologically significant events at U.S. reactors. Mr. Hinson has also served as a NRC project manager for the Fort St. Vrain and Big Rock Point nuclear plants. He has both a B.S. and a M.E. in Nuclear Engineering from the University of Virginia.

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PAPER 3-2
DISCUSSION

Dionne: In your review of these advanced reactors, you mentioned that there was one idea you had for improvement, that interlock, to the transversing in-core probe (TIP) room. Are there any other examples of design changes that were made by the NRC to reduce dose?

Hinson: Yes, there are. I was the radiation protection reviewer for the CE design, and the design change described in my paper was for the CESSAR System 80+ design. The radiation protection reviewer who reviewed GE's ABWR design found that there was inadequate shielding surrounding part of the TIP system which was located over one of the containment personnel access hatches. GE corrected this problem area by adding additional shielding in this location. The ABWR reviewer also noted that there existed a large gap between the reactor vessel shield and the drywell ceiling. In the event of a spent fuel bundle drop onto the refueling pool seal, the resulting dose rates in the upper part of the drywell would be in excess of 20,000 rad/hr. Personnel on the upper or lower decking during this event would have to transient through this radiation field to exit the drywell. GE corrected this potential problem area by the addition of several more feet of concrete and steel shielding in this area to reduce the size of the gap. The radiation protection reviewer who is reviewing Westinghouse's AP-600 design has had several confirmatory shielding calculations performed for various areas of the plant to determine the adequacy of the plant shielding design.

Rescek: We get into a lot of temporary hanging of lead and shielding and you talk a little bit now in your answer to Bruce about permanent shielding in place. Will these plants be designed such that the criteria for hanging temporary shielding may automatically be precalculated so we don't go through all of these special calculations to see how much lead loading we can put on various lines in the plant? Will all of that type of process be avoided here?

Hinson: Well these designs don't get into that level of detail because a lot of the operational concerns are left up to the individual utility. However, these plant designs will have places to hang temporary shielding and they will be designed for adequate space for storage and erection of temporary shielding.

Cybul: Based on historical data, BWRs consume more dose than PWRs. Is it reasonable for the next-generation design to have the same criteria?

Hinson: Based on the advanced reactor dose estimates, the two GE BWR designs have a dose estimate of 92 and 99 person-rem/yr vs. 68 and 79 person-rem for the PWR designs. These estimates are very close. Also, since the shielding design is not complete for these plants, these dose estimates are very preliminary, and it is really hard to say whether that trend will continue in the advanced designs.

Material Changes

- Where Possible, Limit Cobalt Content in Piping and Other Components to 0.05 w/o
- Limit Cobalt Content in S/G Tubes to 0.015 w/o
- Use Titanium for Main Condenser Tubes and Tube Sheets
- Minimize Use of Antimony in Reactor Coolant Pump Bearings (CE)

Component Design Features

- Elimination of Recirculation Pumps/Piping (GE)
- Refueling Improvements (Integrated Reactor Head Removal Package, Stud Tensioner, Automated Refueling Bridge)
- Modifications to TIP System (GE)
- Use of Canned Pumps
- Use of Mechanical Versus Hydraulic Snubbers (CE)

Features to Facilitate Maintenance

- Easily Removable Reactor Coolant Pumps (W)
- Internal CRD Restraint Feature to Facilitate CRD Removal (GE)
- Flushing Capability Provided for Systems Containing Radioactivity
- Adequate Work and Laydown Space Around Components
- Provisions to Facilitate Equipment Removal/ Transport (In-Place Rigging, Lifting Equipment, Overhead Monorails, Carts)

Features to Facilitate In-Service Inspection

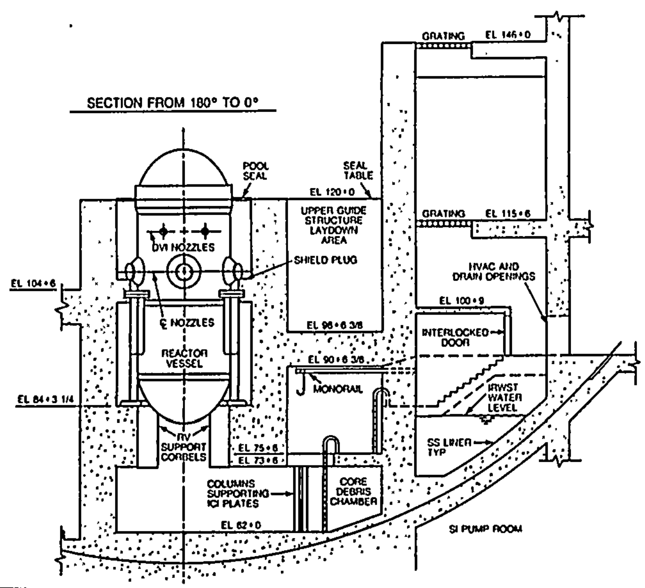
- Use of Permanent Platforms Around Major Equipment to Facilitate Maintenance and In-Service Inspection
- Use of Easily Removable Blanket or Mirror Type Thermal Insulation Around Piping and Components
- Use of a Forged Ring Pressure Vessel and Seamless Piping to Reduce the Number of Welds, Thereby Reducing Time Required for Weld Inspections
- Use of Robotic Equipment to Perform Remote Weld Inspections in High Radiation/Remote Areas

Features to Facilitate In-Service Inspection
(Continued)

- **Steam Generators Have Several Design Features to Facilitate Maintenance and In-Service Inspection (CE, W)**
  - S/G Designed to Accomodate Robotic Equipment for Tube Maintenance and ISI
  - Large Diameter Manways to Facilitate Personnel and Tool Access
  - Platforms and Laydown Areas for Inspection and Maintenance Purposes

Plant Layout Features

- **Redundant Radioactive Components Separated and Shielded From Each Other**
- **Labyrinth Entrances to Cubicles Sized to Permit Equipment Changeout**
- **Ion Exchangers Located in Pits with Spent Resin Tanks Beneath the Ion Exchanger to Reduce Area Dose Rates (CE)**
- **Spent Fuel Transfer Tube Completely Enclosed in Concrete (No Seismic Gap Which Requires Inspection) (W)**



Plant Layout Features
(Continued)

- **In-Place Equipment to Facilitate Removal of Safety Relief Valves, Reactor Internal Pumps, and Other Valves in the Drywell (GE)**
- **Large Staging Areas Provided Both Inside and Outside the Reactor Building Equipment Hatch and Personnel Airlocks (CE)**
- **Hot Tool Cribs Located in Low Radiation Areas Adjacent to Maintenance Areas (CE)**
- **Dedicated Change-Out Areas Located Near Airlocks in Low Dose Areas (CE)**

CONTROLLING RADIATION FIELDS IN SIEMENS DESIGNED LIGHT WATER REACTORS

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ABSTRACT

An essential item for the control of radiation fields is the minimization of the use of stellites in the reactor systems of Light Water Reactors (LWRs). A short description of the qualification of Co-replacement materials will be followed by an illustration of the locations where these materials were implemented in Siemens designed LWRs. Especially experiences in PWRs show the immense influence of reduction of cobalt sources on dose rate buildup. The corrosion and the fatigue and wear behavior of the replacement materials has not created concern up to now.

A second tool to keep occupational radiation doses at a low level in PWRs is the use of the modified B/Li-chemistry. This is practized in Siemens designed plants by keeping the Li level at a max. value of 2 ppm until it reaches a pH (at 300°C) of ~7.4. This pH is kept constant until the end of the cycle.

The substitution of cobalt base alloys and thus the removal of the Co-59 sources from the system had the largest impact on the radiation levels. Nonetheless, the effectiveness of the coolant chemistry should not be neglected either.

Several years of successful operation of PWRs with the replacement materials resulted in an occupational radiation exposure which is below 0.5 man-Sievert/plant and year.

INTRODUCTION

Radiation Field Control in Siemens designed Light Water Reactors (LWRs) will be the continuation of the efforts in the last 20 years. An example for these efforts: the average annual radiation exposure in Siemens designed PWRs was 5.25 man-Sievert per plant and year in 1980 and 1.73 man-Sievert per plant and year in 1990. A comparison of these values with statistics from other countries shows that the average personnel exposure of all Siemens plants is comparable to the values in most other countries.

However, taking into account only Siemens plants, which started operation after 1985, a decrease of personnel exposures of one order of magnitude can be observed compared to those plants which started operation before 1985. In order to achieve personnel exposures in such a low range, new concepts in shielding and material selection of these "recent" plants were necessary. The most important objective hereby was to reduce the radiation levels. To obtain this objective, the Co-60 had to be eliminated by eliminating its precursor Co-59.

Therefore a new material concept had to be developed. In order to enable the realization of this concept, the main cobalt sources had to be identified and suitable replacement materials had to be qualified. By using the new qualified materials a stepwise reduction of the identified relevant cobalt sources could be performed.

Simultaneously, a new concept for primary coolant chemistry was developed. The intention hereby was to achieve a pH in the primary coolant which would have a stabilising effect on the oxide layers. Especially the re-dissolution of nickel and cobalt and their transport was to be minimized.

IMPROVEMENT OF THE MATERIAL CONCEPT

During extensive research programs in the 70s, cobalt base alloys with a cobalt content of > 50%, especially Stellite 6, used in the reactor pressure vessel (RPV) area, in valves, and in pumps were identified as main cobalt sources. Furthermore, plating material and fuel assembly materials, located in the neutron field of the RPV and containing cobalt as an impurity, were taken into consideration as a Co-60 source.

Thus, the development and testing of replacement materials was started in the 70s, considering mainly the core internals made of cobalt base alloys, and was continued in the 80s, taking into account then the entire primary system. A list of replacement materials with a significantly reduced cobalt content (Table 1) and various mechanical characteristics (Figure 1) resulted. The replacement of most cobalt base alloys was now enabled by selecting these materials according to the location and mechanical load of the relevant components.

Table 1. Chemical composition of Co-free replacement alloys

		Chemical composition in weight %									
ALLOY	Hardness HRC	C	Si	Mn	Cr	Mo	Ni	N	W	V	Fe
EVERIT 50	47-54	2.5	0.4	0.9	25	3	-	-	-	-	bal.
EVERIT 50 SO	43-48	2.0	0.4	0.9	25	3	-	-	-	-	bal.
ANTINIT DUR 300	27-33	0.12	5	6.5	21	-	8	-	-	-	bal.
CENIUM Z 20	43-48	0.3	-	-	27	9	17.5	-	2	-	bal.
SKWAM	36-42	0.2	0.7	0.55	17.5	1.1	-	-	-	-	bal.
NITRONIC 60	20-29	0.1	4	8	17	-	8	0.13	-	-	bal.
Cr-STEELS	24-44	0.2	<1	<1	16.5	-	2	-	-	-	bal.

Detonation-gun-coatings: Coating type LC-1C of union carbide
Composition Cr_3C_2 with 20% Ni-Cr binder

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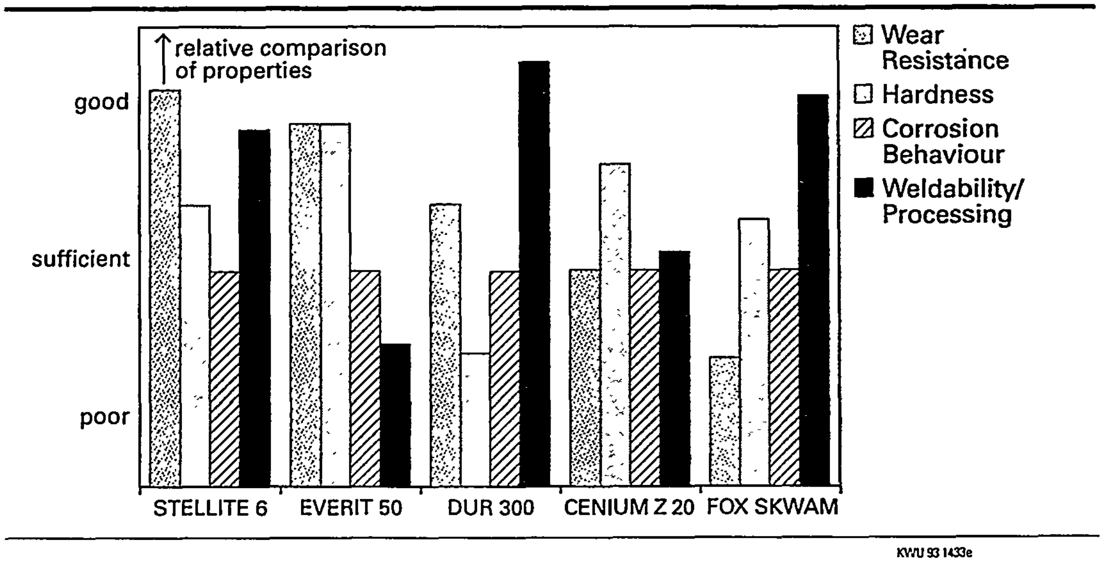

Figure 1. Comparison of main properties of hardfacing alloys

In parallel, the guide tubes, manufactured from stainless steel, and the spacer grids, manufactured from Inconel, were replaced during refuelling outages by those made from Zircaloy.

The first pressurized water reactor (PWR) designed by Siemens with Co-replacement materials started operation in 1985. At this PWR only stellites located in-core were replaced by "cobalt-free" materials. In the following plant the stellites located at the main coolant pumps and the main valves were included in the replacement program. All valves, even those in the auxiliary systems, were taken into account only in the Konvoi plants.

The transition from "older" to "recent" 1300 MWe Siemens designed PWRs can be quantified by comparing the surface areas of the individual materials listed in Tables 2 and 3. As additional information, these tables show the specified Co-59 content of the materials.

Table 2. Materials inventory of "older" 1300 MWe Siemens-designed PWRs

Group	Component	Material	Surface [m^2]	Co-59 Specification [%]
1	Fuel*)	Zircaloy 4	9600	~0
		Inconel 718	1220	<0.1
		Stainless steel	220	<0.1
	RPV-internals	Stainless steel	1124	<0.1
		Co-base alloys	1.1	63
2	Control rod assemblies	Stainless steel	340	<0.1
	Control rod drive	Stainless steel	220	<0.2
		Co-base alloys	1.54	≤67
	Steam generator	Incoloy 800	16276	<0.1
	RPV, Loops	Stainless steel	719	<0.2
	Main coolant pumps	Stainless steel	155	<0.2
		Co-base alloys	1.5	63
3	Auxiliary systems	Stainless steel	~500	<0.2
		Co-base alloys	6.5	63
	Total	Zircaloy	9660	
		Stainless steel	19554	
		Inconel	1220	
		Co-base alloys	10.64	

*) Material composition used before 1985, modifications per fuel cycle possible KWU 93 1436e

Table 3. Materials inventory of "recent" 1300 MWe Siemens-designed PWRs

Group	Component	Material	Surface [m²]	Co-59 Specification [%]
1	Fuel*)	Zircaloy 4 Inconel 718 Stainless steel	~10660 394 220	~0 <0.1 <0.1
	RPV-internals	Stainless steel Co-base alloys	1126 0.026	<0.1 63
2	Control rod assemblies	Stainless steel	340	<0.1
	Control rod drive	Stainless steel Co-base alloys	220 1.54	<0.2 ≤67
	Steam generator	Incoloy 800	16276	<0.1
	RPV, Loops	Stainless steel	719	<0.2
	Main coolant pumps	Stainless steel Co-base alloys	156 0	<0.2 63
3	Auxiliary systems	Stainless steel Co-base alloys	506 0.79	<0.2 63
	Total	Zircaloy Stainless steel Inconel Co-base alloys	10660 19563 394 2.36	

*) Modifications per fuel cycle possible

KWU 93 1435e

The components in these tables are divided into three groups depending on their location:

Group 1: Contains only components which are permanently in the neutron flux

Group 2: Contains components which are outside the neutron flux but within the main circuit

Group 3: Contains components from the auxiliary systems

Approx. 98 % of the cobalt base alloys located in-vessel were substituted by "cobalt-free" materials as can be seen when comparing the surface areas of Group 1. A reduction from 1.1 m² in "older" plants to 0.026 m² in "recent" plants was achieved. This in-vessel replacement had the greatest impact on radiation levels according to the fact that some of these hardfacings were placed close to the neutron flux area.

A comparison of the material concepts in Group 2 shows that the cobalt base alloys of the control rod drives were not replaced. This decision was based on the fact that the cobalt hardfacings in this case have almost no contact to the primary coolant. The release rates of Co-59 from this component into the coolant therefore cannot be very high. Nevertheless, the cobalt base alloys of the main coolant pumps were removed entirely. Thus, a reduction of approx. 50 % of the surface areas was achieved with an effectiveness much higher than 50 %, considering the Co-59 release into the coolant and hence the potential Co-60 buildup rate.

A major reduction (approx. 88 %) of the surface areas from hardfacings was also obtained in the auxiliary systems. However, the influence on the radiation levels was much less here than in Group 1 and also less than in Group 2, because of the reduced contact between these auxiliary systems and the primary circuit.

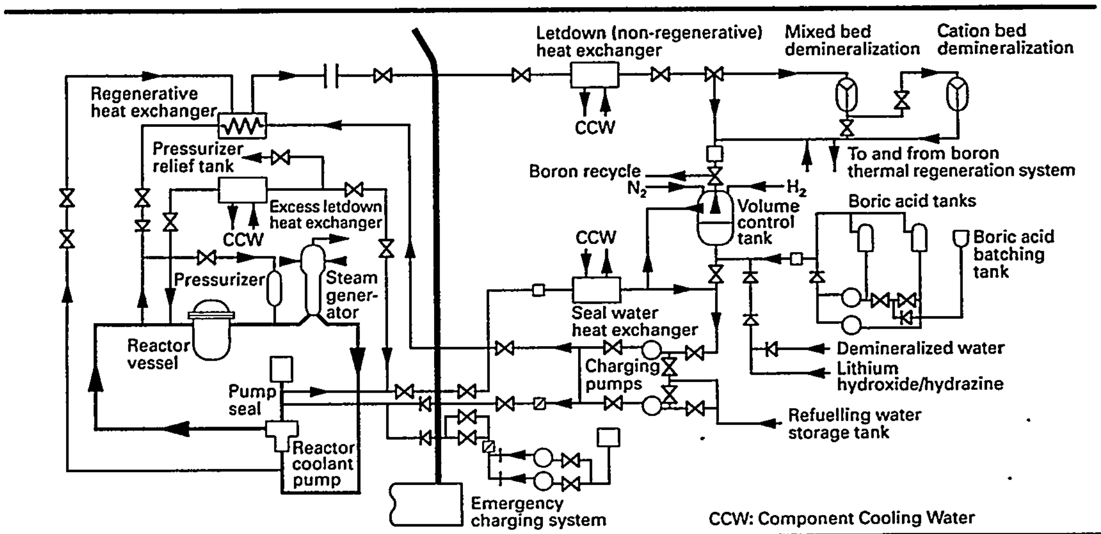

Figure 2. Example of main coolant circuit with CVCS (IAEA, Technical Reports Series No. 347, Vienna, 1993)

The schematic diagram in Figure 2 shows the chemical and volume control system (CVCS) and its connections to the primary circuit.

The replaced in-vessel components (Group 1) are shown in more detail in Table 4 and Figures 3 and 4, since the highest effectiveness on the reduction of the radiation levels was gained by these replacements. The table is valid for substitutions in Philippsburg-2 as well as in the Konvoi plants. The figures have been derived from Philippsburg-2.

Table 4. Substituted in-vessel stellites in Philippsburg-2 and Siemens Konvoi plants

Component	Amount	Gross Area, [m^2]	Substitute
Alignment pin with nut for support columns (control rods)	244	0.8	Cr_3C_2/ CrNi-binder
Grid plate centering bolts	4	0.08	Fox Antinit Dur 300
Hold down plates of the upper core support	112	0.24	Fox Antinit Dur 300
Centering bolts of the upper core support	4	0.04	Fox Antinit Dur 300
Total substituted		1.16	
Total Remaining		0.026	

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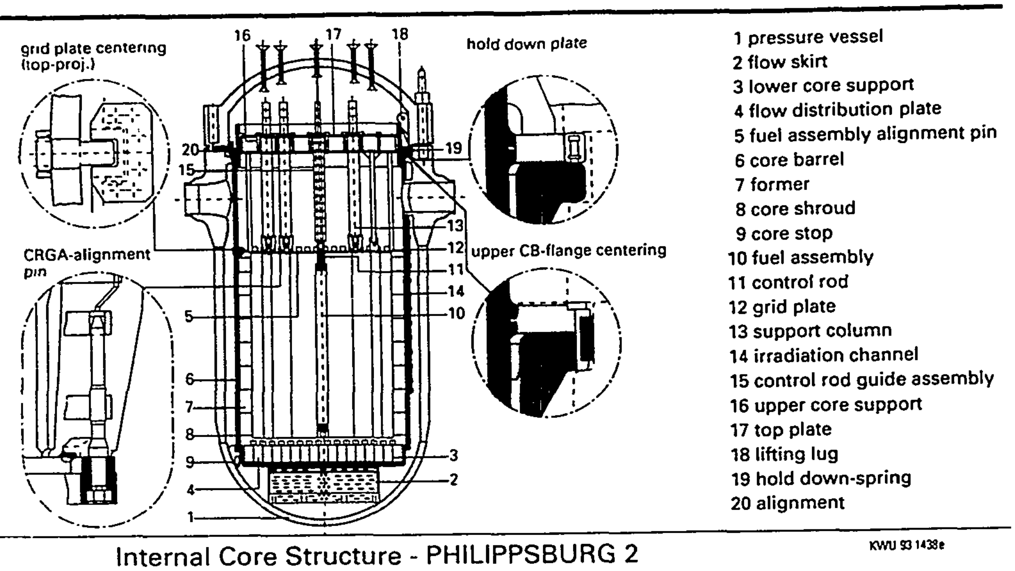


Figure 3. Reactor pressure vessel internals and fuel assemblies

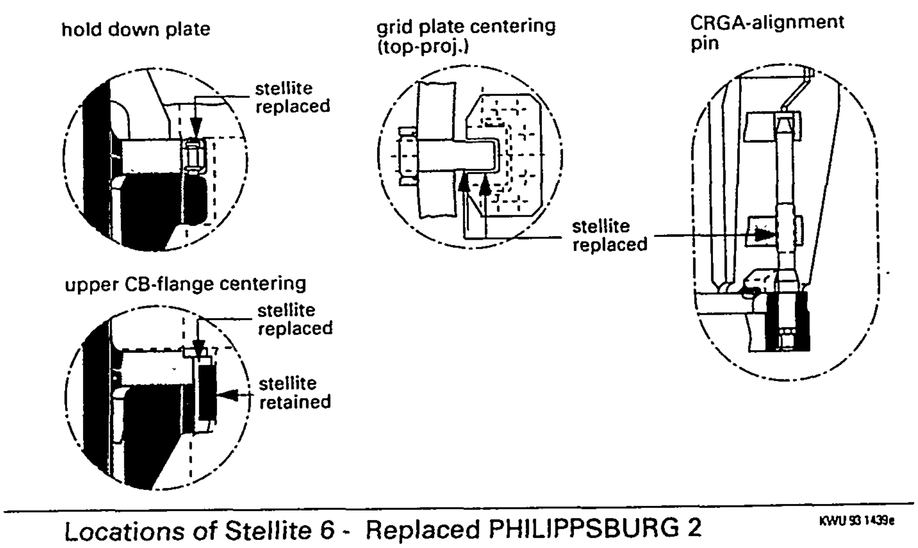


Figure 4. Reactor pressure vessel internals and fuel assemblies

MODIFICATION OF PRIMARY COOLANT CHEMISTRY

The concept of the coordinated coolant chemistry, pH (300 °C) = 6.9, was based on the assumption that the oxide layers in the primary circuit mainly consist of magnetite. However, further research work has shown that oxide layers as well as crud in the primary circuit are mainly composed of spinels in which the Fe(II) of the magnetite is substituted by other bivalent cations. Nickel and cobalt, respectively their activation products Co-58 and Co-60, are thus incorporated in the oxide layers and thereby increase the radiation field.

Considering this and knowing furthermore that, depending on the composition of the Ni- and Co-ferrites, the minimum solubility of these ferrites lies at a pH-level in the range of 7.4, a primary coolant chemistry different from the previous one must be recommended.

Preferably a chemistry with a constant pH, selected from the above mentioned range, should be chosen. However, the lithium concentration is limited to 2 ppm. Otherwise the material compatibility especially of fuel element material is no longer guaranteed. Since lithium hydroxide is used as the pH control agent, a compromise must be found. Therefore it is recommendable to first adjust a constant lithium concentration in the primary coolant until the desired pH is achieved and then to reduce the lithium concentration in dependence upon the boron concentration in such a way that a constant pH can be applied until the end of the cycle.

The effectiveness of this modified coolant chemistry can be experienced by comparing the dose rate development of plant A and plant B in Figure 5. Stellite replacement was not performed in either of these two plants, so that they differ mainly in the primary coolant chemistry. In plant A, with the higher dose rate, coordinated coolant chemistry was preferred with pH(300) = 6.9, whereas in plant B modified coolant chemistry was applied from the very beginning, in this case pH(300) = 7.4.

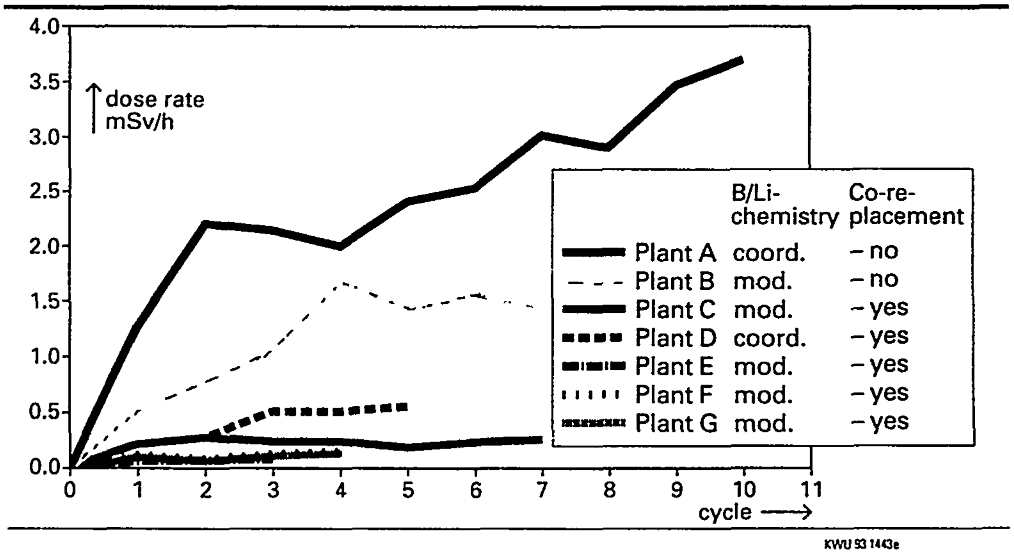


Figure 5. Dose rate development at main coolant piping of seven Siemens PWRs

RADIATION FIELD DEVELOPMENT

As a consequence of these measures taken, the radiation fields and the average occupational radiation exposures decreased in the Siemens designed PWRs. Figure 6 shows this development by classifying the PWRs into 3 groups.

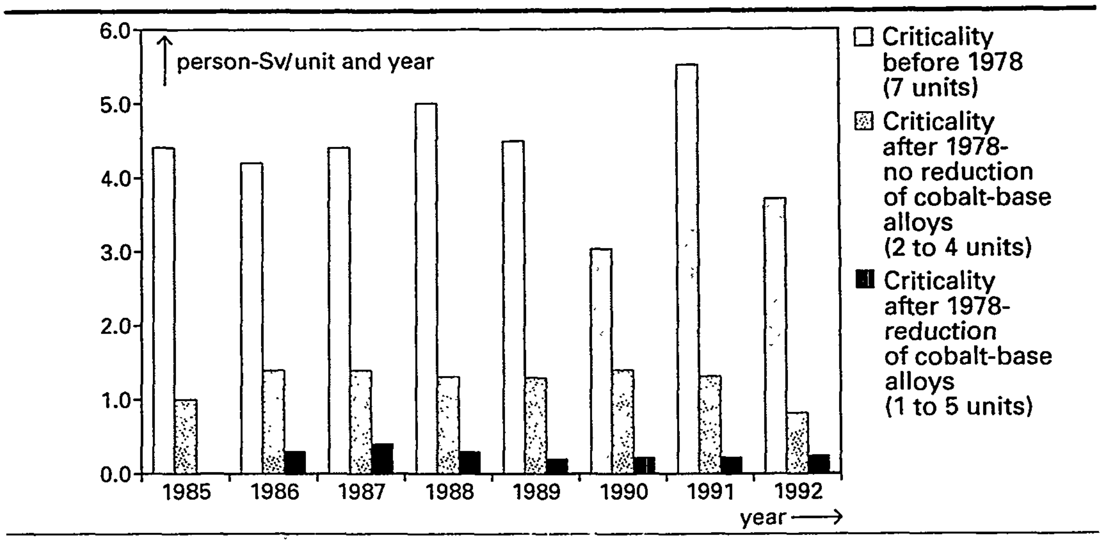


Figure 6. Average occupational radiation exposure per unit and year of Siemens PWRs

The PWRs in the first and second group had no reduction of cobalt base alloys. However, the plants in the first group (oldest plants), which had their first start-up before 1978, had generally uncoordinated or coordinated coolant chemistry during the early fuel cycles and only poor shielding, whereas the plants in the second group, which started first operation after 1978, applied mainly modified coolant chemistry from the very beginning and also had improved shielding. Yet, the third group consists of plants at which the cobalt base alloys were substituted either partly or in full extent as described in Table 3. This last group shows that the measures performed succeeded in an average occupational radiation exposure of less than 50 Rem per plant and year.

Figure 7 is an example for the dose rate development of various PWRs with Co-replacement activities but different coolant chemistry.

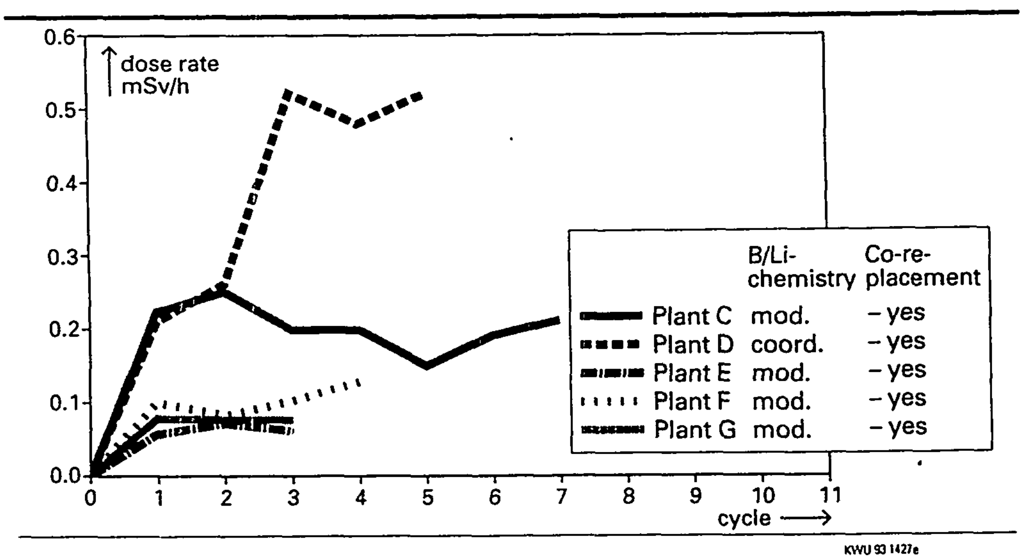
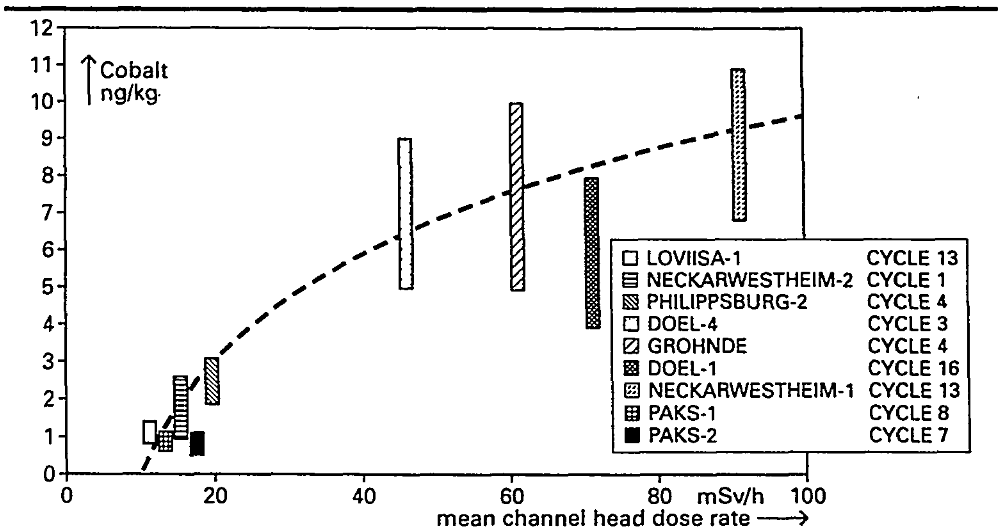


Figure 7. Dose rate development at main coolant piping of five Siemens designed PWRs

Compared to Figure 5, this figure shows that the influence of the coolant chemistry on the radiation levels is not as large as that of material replacement, but it still should not be neglected.

The overall objective of Co-replacement activities can be illustrated with Figure 8, where a comparison of soluble elemental cobalt-59 in the primary coolant is shown in comparison with channel head dose-rates. According to this figure the overall objective should be to reduce the soluble Co-59 to values less than 5 ppb in order to thus achieve distinctly reduced dose rate levels.



(Source: Nuclear Electric and AEA Technology) KWU 93 1445e

Figure 8. Comparison of soluble elemental cobalt in the primary coolant with channel head dose rate

Author Biography

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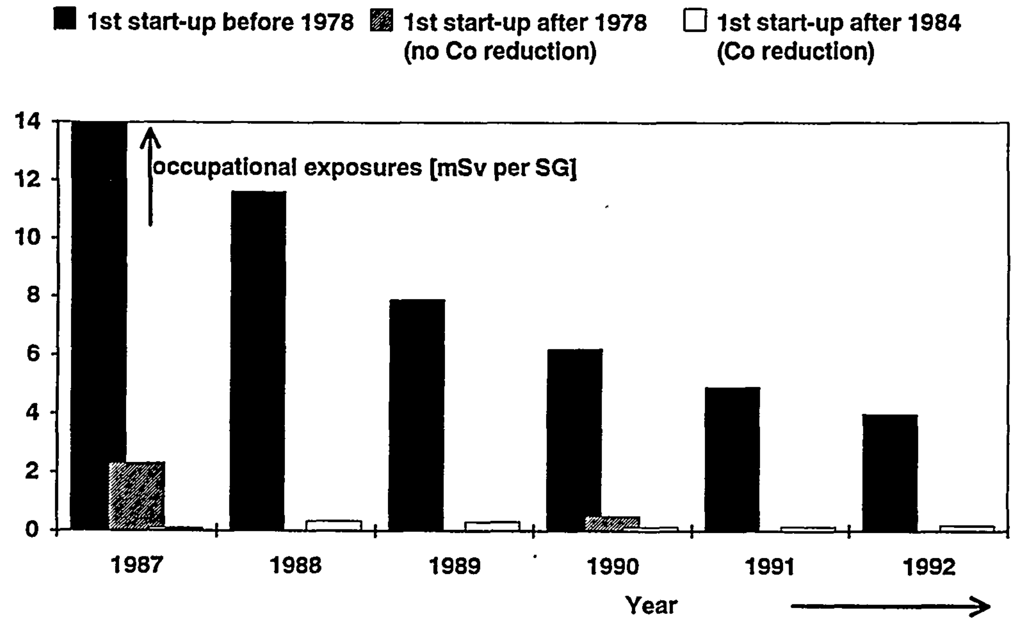
PAPER 3-3
DISCUSSION

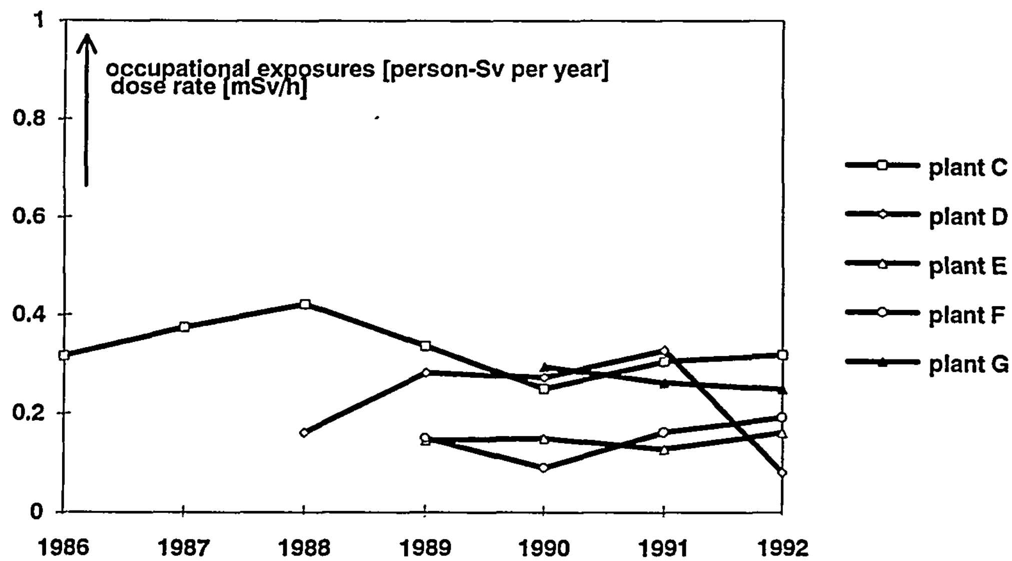
Khan: Rolf, you've shown that the doses for your plants are now extremely low. Do you think that we have reached a kind of a bottom line, or is there any more possibility of reducing doses even further in Siemens plants? Question number two, what kind of pH are you operating at in your newer plants?

Riess: As I said, our intent is to further reduce these exposure rates by those measures I mentioned. Take counteractions against nickel, that is the main target, further reduce the cobalt in the auxiliary systems, mainly in the SVCS equipment, of of course, provide decontamination as a routine measure against residual fields if you have to do specific works. These are the major actions that we have on the way. As to the second question about the pH, we introduced another feature about 10 years ago. We are operating at the upper level of the lithium concentration, in our case it is 2 ppm of lithium up to 7.4, and then follow this line. So except for three stations out of the 16, all are following the modified chemistry. Admittedly, this is a bit of a gamble, but we think we are in pretty good shape.

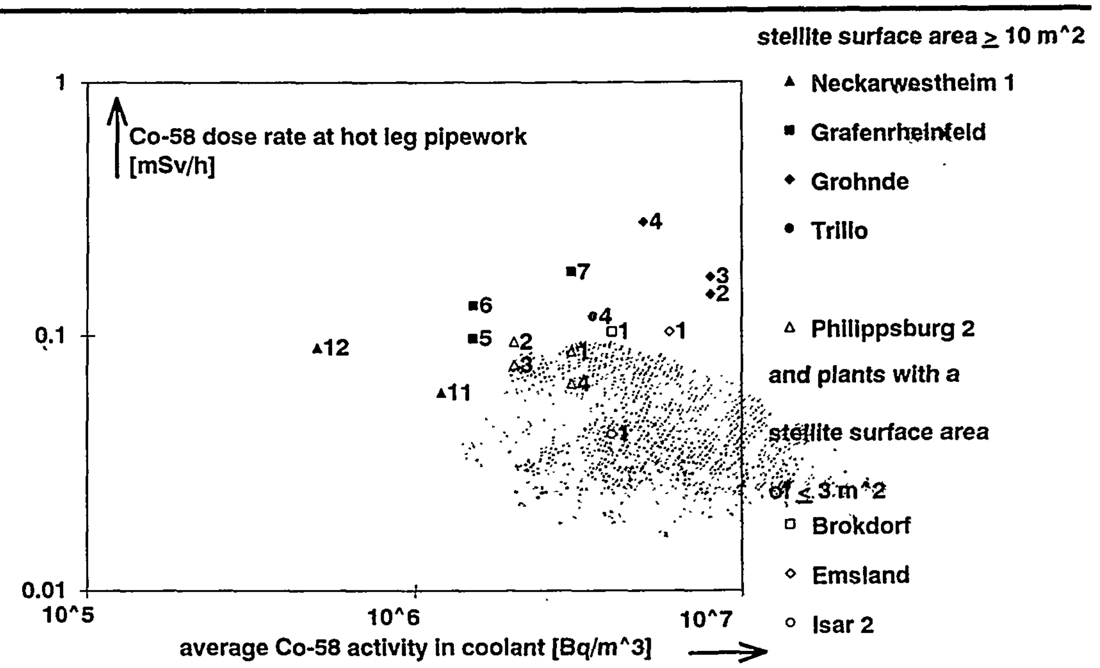
Personnel exposures during tube sheet lancing in steam generators of Siemens designed PWRs



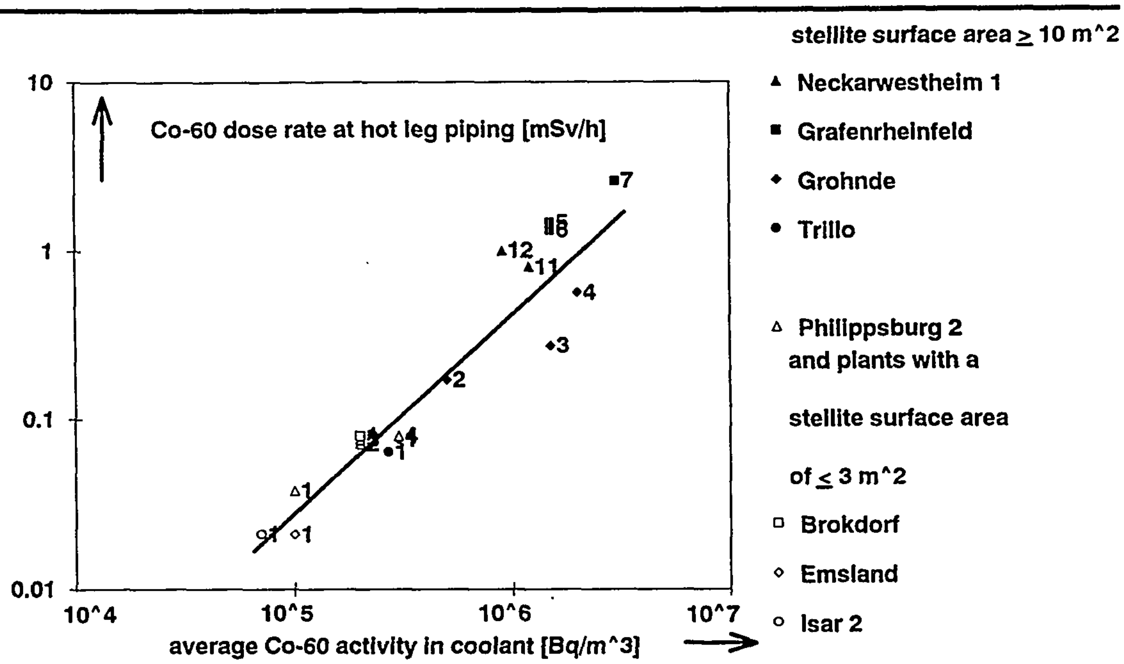
Individual occupational radiation exposures of "recent" Siemens designed PWRs



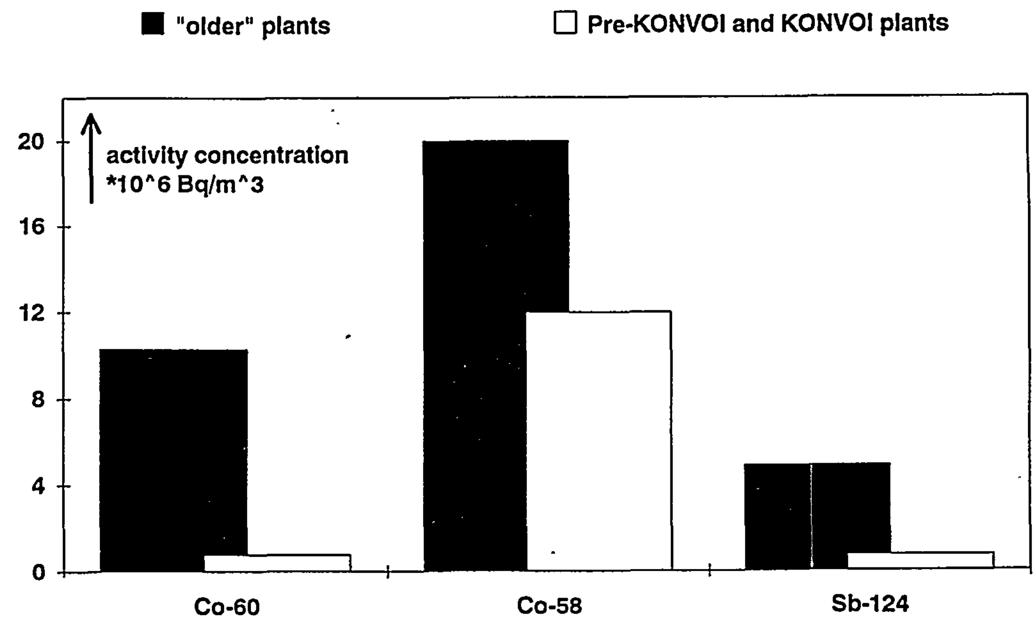
Comparison of Co-58 coolant activity concentrations and Co-58 hot leg dose rates for equivalent pipewall thickness (59 mm)



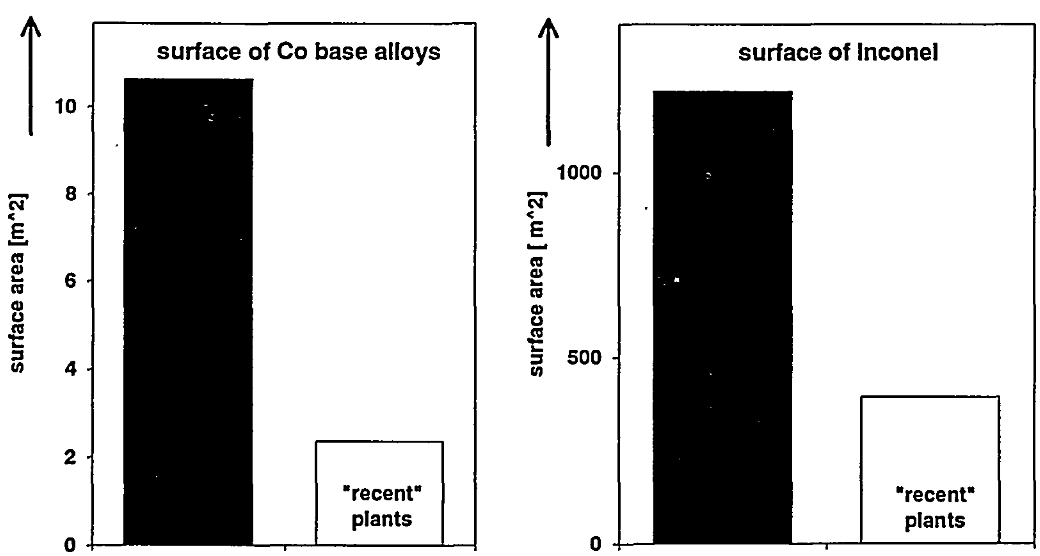
Comparison of Co-60 coolant activity concentrations and Co-60 hot leg dose rates for equivalent pipewall thickness (59 mm)

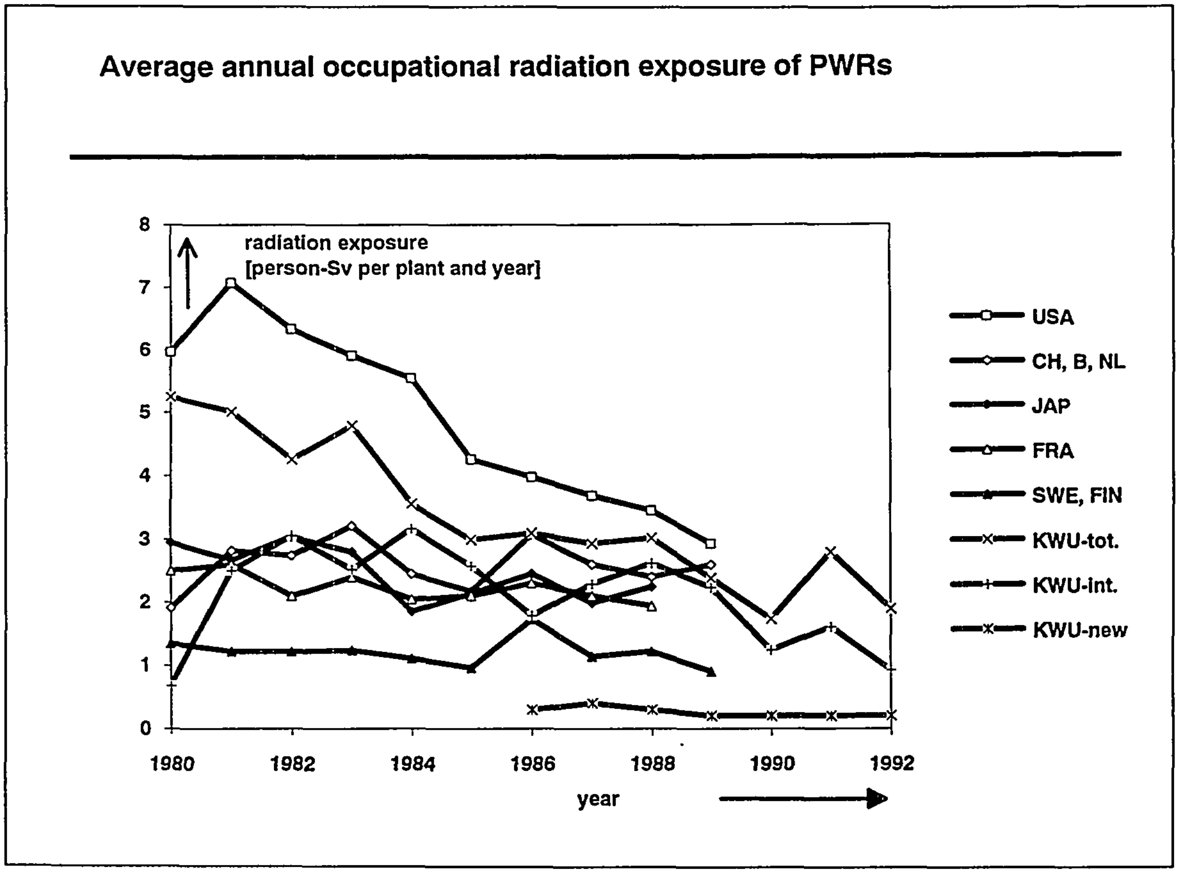


Influence of the material concept on the activity concentrations of Co-60, Co-58 and Sb-124 in the primary coolant of Siemens designed PWRs



Materials inventory of "older" and "recent" 1300 MWe Siemens designed PWRs




Average annual occupational radiation exposure of PWRs
radiation exposure
[person-Sv per plant and year]
8
7
6
5
4
3
2
1
0
1980
1982
1984
1986
1988
1990
1992
year
USA
CH, B, NL
JAP
FRA
SWE, FIN
KWU-tot.
KWU-int.
KWU-new

ALARA RADIATION CONSIDERATIONS FOR THE AP600 REACTOR

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INTRODUCTION

The radiation design of the AP600 reactor plant is based on an average annual occupational radiation exposure (ORE) of 100 man-rem. As a design goal we have established a lower value of 70 man-rem per year. And, with our current design process, we expect to achieve annual exposures which are well below this goal. To accomplish our goal we have established a process that provides criteria, guidelines and customer involvement to achieve the desired result. The criteria and guidelines provide the shield designer, as well as the systems and plant layout designers with information that will lead to an integrated plant design that minimizes personnel exposure and yet is not burdened with complicated shielding or unnecessary component access limitations. Customer involvement is provided in the form of utility input, design reviews and information exchange. Cooperative programs with utilities in the development of specific systems or processes also provides for an ALARA design. The results are features which include ALARA radiation considerations as an integral part of the plant design and a lower plant ORE. It is anticipated that a further reduction in plant personnel exposures will result through good radiological practices by the plant operators.

The information in place to support and direct the plant designers includes the Utility Requirements Document (URD), Federal Regulations, ALARA guidelines, radiation design information and radiation and shielding design criteria. This information, along with the utility input, design reviews and information feedback, will contribute to the reduction of plant radiation exposure levels such that they will be less than the stated goals.

RADIATION GUIDELINES AND CRITERIA

The URD is an important part of the design package, as this document contains the customers requirements for future nuclear plants. The document is several volumes in size and addresses plant features that current plant operators prefer in nuclear plant designs. An example of the URD requirements for reducing personnel radiation exposure is the stipulation that an ORE assessment of maintenance activities must be made to confirm that man-rem requirements and objectives can be met. Robotic analyses for key maintenance activities, in conjunction with a maintainability evaluation program that includes consideration for in-service inspections, are also required.

ALARA guidelines also provide information to aid the systems and plant layout designers in the implementation of methods that will maintain radiation doses ALARA. The guidelines provide typical radiation fields adjacent to plant components and systems and various methods to avoid or reduce the radiation exposure to workers during plant or system maintenance or repair. The guideline for each plant component provides guidance and recommendations regarding component accessibility, maintainability, material impurities, location and good design features. A checklist can be used by the designer to judge whether the design meets the requirement of ALARA.

A check on the effectiveness of the process and the ALARA guidelines is provided by periodic evaluations of the expected annual radiation dose from planned AP600 plant operational activities. A compilation of the predicted doses in 1991 indicated that the average annual ORE will be less than 70 man-rem per year. Much of the shielding arrangements and systems information available at that time has since been upgraded, thus this estimated dose has now become an upper limit (goal) for radiation exposure from plant operation and maintenance activities. An upgrade of the evaluation is in progress in order to reflect the more recent incorporation into the shield and arrangements design of additional design features and methods which will reduce the estimated ORE.

Radiation design information for the AP600 includes design basis radiation source values for the various plant components and for postulated accident scenarios. The source values are also used in design efforts for waste handling and disposal and in planning to minimize associated radiation exposures to plant personnel. The plant parameters and assumptions used to develop the source terms are chosen such that calculated results are realistic without being over conservative. It is important that the design basis data contain some margin for unforeseen future considerations, but it is also important that the plant costs not be greater than necessary because of unidentified conservatism.

The radiation and shielding guidelines and criteria for the AP600 includes the requirements of the Code of Federal Regulations, Regulatory Guides, the URD and the ALARA guidelines. The plant specific radiation zone and access requirements and the access control criteria also are a part of the designers guidelines and criteria.

CUSTOMER (UTILITY) INPUT AND FEEDBACK

Utility input and feedback is an important part of our ALARA program as it provides insight into the actual operation and maintenance of plant systems and components. It is provided, in part, through the URD, and also through the effective involvement of the Advanced Reactor Corporation (ARC). This type of information is also provided through utility participation in design reviews, solicited input regarding system designs and area layout efforts and through joint programs for the development of process and procedure improvement.

An example of a joint program is one currently in progress with a utility to develop an improved process for handling waste such as contaminated filters, resins and other contaminated articles. The extensive use of robotics and remote operations is an expected result of this program. Another example of utility involvement is their participation in the maintainability evaluation of plant equipment on a cubicle by cubicle basis. This evaluation will include the use of robotics and consideration for in-service inspections. Utility participation in plant area access control is also a vital part of the plant arrangement effort which will minimize exposure in a cost effective manner.

Although not a customer, the Architect Engineering Firms involved in the plant design also provide significant input to the ALARA program through their participation in design reviews, planning sessions and general comments.

PROCESS RESULTS THAT SUPPORT THE ALARA GOAL

Plant Simplification

Several considerations have been designed into the AP600 which will reduce radiation exposures to plant personnel in a cost effective manner. This includes a larger containment which allows space for equipment laydown without crowding. This will reduce or eliminate the need to perform tasks where radiation from adjacent equipment is contributing to the worker dose. It has also allowed the use of a clearly defined clean area which is distinctly separate from the radiation control area (RCA). The containment design also provides an equipment hatch such that a truck can be driven directly into containment, thus simplifying the removal or return of equipment or plant components.

The plant buildings are close coupled so that the movement of equipment is simple and direct. One unique feature in the Annex Buildings is the inclusion of a "hot" maintenance shop which is designed to accommodate reactor coolant pump repair as well as other tasks on contaminated equipment. The layout of this facility as well as of the building has been reviewed by the ARC and utility personnel to insure optimum locations for the various rooms and functions of the building.

Other simplifications of note is the significant reduction in the number of valves in the plant (by 60%), as well as a reduction in the feet of piping (by 75%) and the reduction in the number of pumps used in the plant (by 35%). The chemical and volume control system (CVS) has also been simplified to operate on differential pressure across the reactor coolant pump, thus eliminating high pressure pumps in the system. The only pump required will be to inject plant make-up water. All of these simplifications will result in reduced maintenance efforts and radiation doses.

Waste Handling Considerations

Extensive effort has been spent to reduce and minimize the radiation exposure from waste handling operations required for the AP600. This includes the robotic and remote operation processes being developed with a utility as well as the planned use of cameras and mirrors to avoid and minimize exposure. Resin transfer operations will be through piping using air handlers with the ability to flush the lines if local "hot spots" develop. The ability to install temporary shielding over local "hot spots", should they occur, will also be available.

Component Considerations

The use of bent piping in the reactor coolant system has eliminated welds which must receive in-service inspections, thus reducing a source of exposure in the plant. The reactor coolant pumps have also received significant attention with respect to minimizing exposure. This has resulted in the use of two pumps per loop, the use of highly reliable canned motor pumps and the specification of polished impeller and flow vanes, which will reduce crud buildup as well as improve pump efficiency. The pump considerations also include the design of a quick removal and transport system which will minimize personnel exposure if pump repairs are required.

The steam generators have also received attention to increase reliability and in-turn to reduce radiation exposure. This includes the specification of Inconel 690 tubing and the specification that the cobalt impurity in the tubes be less than 0.015 percent by weight. The minimization of cobalt in other plant materials has also been specified with consideration given to the cost benefit expected for the amount of cobalt allowed. Allowable values were based on the expected amount of cobalt that might be input into the plant by the component as well as the cost to reduce the amount of the impurity.

Air operated pumps will be used for various waste tank applications; however, the rupture of a pump diaphragm could result in room contamination. The solution for this concern was to pipe the air vent path back into the top of the tank being served. Other components that have received special attention are the plate type heat exchangers planned for use in the spent fuel cooling system (SFS) and the heat exchanger in the CVS. The SFS heat exchangers typically do not provide significant self shielding, thus local shielding was placed adjacent to this component. The CVS heat exchanger could require replacement during plant life, thus an equipment hatch was added above this component to facilitate removal with minimal radiation exposure.

Fuel Considerations

Several features have been considered with respect to the fuel and its effects on radiation exposure throughout plant life. One considerations for the fuel is the use of gray rods for reactivity control. These reduced rod worth, control rods can be moved to provide daily load follow without changes in the soluble boron concentration. This greatly simplifies the auxiliary systems used in processing the borated coolant.

In addition to the use of gray rods, the fuel will be assembled using zirconium grid straps to eliminate the input of cobalt from these components. A reduced power density will also be utilized which will reduce the activation of components and also the exposure for various component handling operations.

Another core related consideration to reduce radiation levels in the plant is the planned use of non-cobalt bearing material in the control rod latch mechanism. This component is one of the higher wear items which can introduce cobalt into the primary system.

Shielding Considerations

In order to minimize the exposure to workers performing maintenance on equipment adjacent to radiation sources such as other pumps, valves or waste tanks, various shield walls and local shields have been provided between the radiation emitting components. The means to install temporary shielding is also being provided where access requirements for adjacent equipment preclude the use of a permanent shield. The addition of temporary shielding is contingent on the ability to provide adequate space for equipment disassembly and reassembly.

A laydown area has been provided just outside the steam generator compartment. In order to reduce dose rates in the laydown area, steel shielding has been added above the reactor coolant piping in the steam generator compartment. This will also allow low dose access to the compartment for maintenance work. Steel shielding has also been added in the area of the north steam generator compartment to shield the pressurizer surge line and valve gallery outside the compartment. This is in addition to the shielding in the CVS to separate the valves and other components for maintenance purposes.

Initial primary shield design analysis has shown that significant neutron streaming will occur through the relatively large annuli around the reactor coolant piping and in the reactor cavity. A detailed primary shield analysis using three-dimensional techniques will be used in the final design of local shields which will address these and other concerns.

During a review of the plant shielding it was noted that access to some areas for maintenance requires passing by or through areas of higher radiation fields. Alternate routes will be identified or shielding provisions (permanent or temporary) are being provided so that exposures when accessing all areas of the plant will be minimized.

Since SECY-93-087 defined post accident sampling system (PASS) requirements for advanced light water reactors, the AP600 PASS requirements differ than those for existing plants. This system is currently being reviewed to insure that the final design will meet the requirements for sample time and frequency as well as for personnel dose limits.

Plant access control has been evaluated as part of a review of the plant radiation protection system. Access requirements were based on expected radiation levels rather than on the design basis values assumed for shield design analysis. In order to allow the plant operator as much flexibility as possible, doors have been provided at all locations which could require personnel exclusion or controlled entry. The decision of whether or not to install locks on these doors is left to the plant operator should plant radiation levels require locked barriers. In addition, if ventilation air flow was considered and most of the doors will be constructed of wire mesh rather than being solid.

CONCLUSIONS

Based on the ALARA considerations presented here and on the process identified for the design of the AP600 Plant it is concluded that the plant design will meet the requirement of ALARA and will have an annual average ORE which is less than the current goal.

Author Biography

Fred Lau is Manager of Radiation Engineering and Analysis for the Westinghouse Nuclear Technology Division. In this position he is responsible for the many aspects of radiation evaluation, attenuation and control as it applies to both personnel and equipment radiation exposure. He has 28 years of experience in the design and evaluation of reactor shielding through his assignments at the Bettis Atomic Power Laboratory. This included the design and evaluation and testing of shielding for nuclear powered ships and the design of the shielding for the Shippingport Atomic Power Station for both Core 2 and for the Light Water Breeder Core. Prior to his assignment at NTD in 1984, Mr. Lau was the Manager of Shielding Design and Manager of Core Mechanical Design for naval nuclear plants.

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PAPER 3-4
DISCUSSION

Wood: Do you have any plans to put fine filters in the system? We heard this morning from Tas Khan that filters seem to work to reduce exposures in the German plants, and I wondered if you had plans to put them in.

Lau: Are you talking about the .4 and below micron filters?

Wood: Yes.

Lau: There are plans to do that.

Wood: Just in the CVS system?

Lau: As it stands right now, that is about all we have talked about, but I'm sure that those kind of things can be fitted to the customer's desires.

Borst: The 70 rem/yr -- how many outage days does that include? Does that include standard refueling outage or is that normal operations?

Lau: That includes the standard refueling outages. We have allowed 25 days for refueling and our goal is to refuel in 17 days. As a matter of fact, I believe that better than 90% of the dose comes from refueling and maintenance work during a refueling outage.

Borst: You mentioned shielding component cubicles. I know Ringhals Unit 1 has short walls as their shielding cubicles. Were you having something like that or were you envisioning a complete cubicle with a door and/or labyrinth surrounding each component?

Lau: What we have in a lot of the tank rooms that I showed are complete walls that surround the tank except for a doorway or a labyrinth-type door; in some cases there is a ladder that would get you to an area where you need to do work.

Borst: What about smaller pumps and things as opposed to large tanks?

Lau: For the small pumps, we are planning a head-height, half-height if you will, shield wall made out of either steel or concrete. In some cases that is not going to be easy because you have to be able to get to the components to work on them, and if you put the wall between two of them it may just impede that. So we are making plans for temporary shielding that could be put in semi-remotely to allow work on one component or the other.

Ferguson: Does the AP-600 design require any kind of vital access post-accident? If so, do you have any type of ALARA features designed to protect post-accident operators?

Lau: With regard to post-accident, we have been reviewing our post-accident sample requirements and some of those requirements have changed in recent times. One of the things that we have done is to discuss this with about fifteen different utilities and have just spent some time at Commonwealth Edison reviewing with them their post-accident sampling systems and those kinds of requirements. What we hope to do is to design a system that will answer the questions and concerns all of the utilities that we have talked to have in regard to minimizing the radiation exposure. As a matter of fact, we performed a dose assessment for sampling after an accident, and came up a little bit over the 5 rem that we had planned. We would really like to be below that, so we are back to the drawing board.

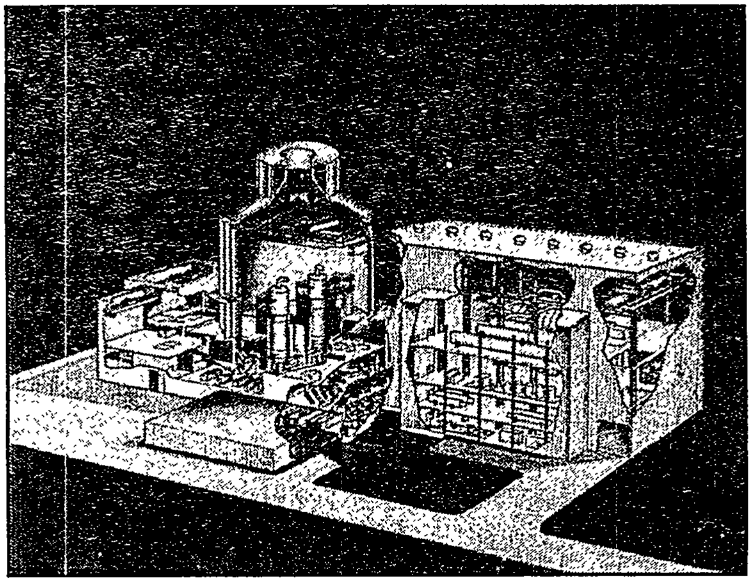
Ferguson: Would the shielding requirements then be reflecting the new source terms from NUREG 1465 or staying with the GID 14844?

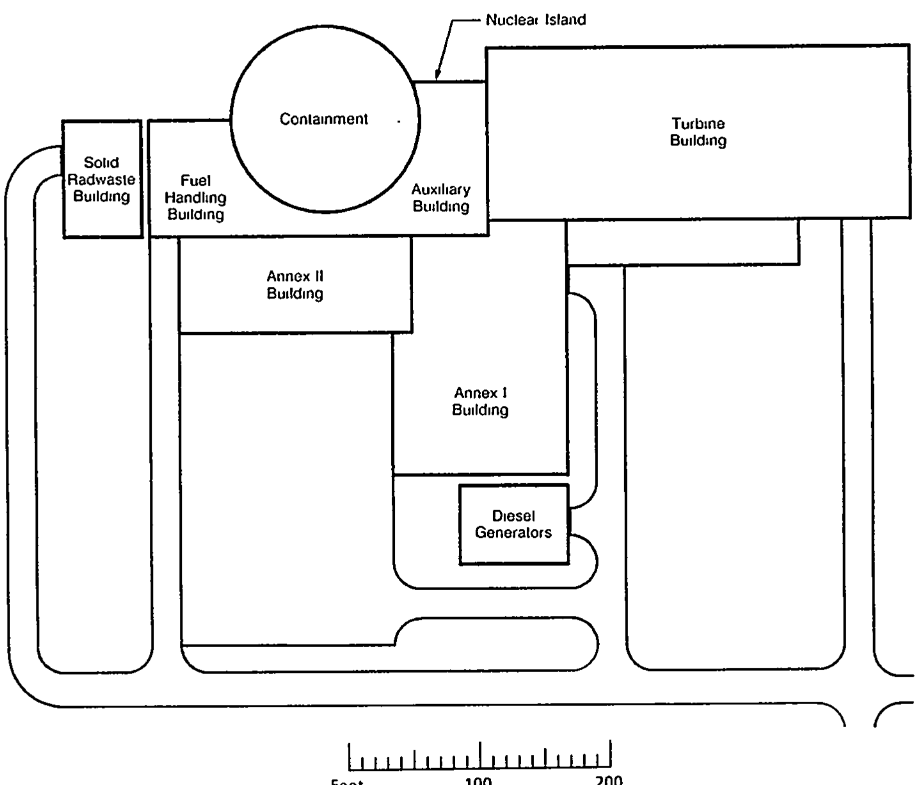
Lau: We are using the draft NUREG-1465.

Baum: I have three questions. This morning we heard from Mr. Terada from Japan that they are using automatic control for the chemical and volume control systems. I am wondering if you are considering that. Secondly, I believe at our international workshop five years ago the Japanese were speaking about using monorails to transport tools and equipment and perhaps surveillance equipment around a plant. Has any consideration been given to that. Thirdly, those of us who are parents and grandparents know how useful and cheap the camcorder and remote surveillance systems are. How much of that sort of thing is being built into these newer plants? Do you have cameras all over the plant? How many remote cameras would there be in a typical plant?

Lau: With regard to automatic pH control, we've not gone as far as the Japanese in our thinking, however, we do have, and I forgot to mention this, our chemical and volume control systems designed to operate on differential pressure, there are no pumps in that system. That doesn't answer the question in regard to pH, but I wanted to bring out the fact that the system is very simplified and certainly the idea of automatic pH control is something that we would consider. I have not heard about the automatic pH control feature, but I am sure that some of my other people have. In regard to monorails, yes there are areas which will be equipped with monorails, especially in the waste handling building, and as I mentioned, with regard to coolant pumps other areas that lend themselves to the monorail or other kinds of remote handling. I don't have first-hand knowledge of all of those things at the moment, but I would say yes, we are designing for remote handling such as monorails. Thirdly, we plan to have a lot of cameras in the waste handling building and we also plan to have electrical circuits that would allow multiplexing the camera, and other radiation dose monitoring throughout the plant. We are not going to dictate to the customer what he has to put where, but we are going to provide him with the capability to put things wherever he needs them.

AP600— The New Westinghouse Standard 600 MWe Plant





37625 1 Site Plan AP600

ALARA CONSIDERATIONS
AP600

INTRODUCTION

- o **Design Goal**
- o **Criteria**
- o **Guidelines**
- o **Design Information**
- o **Customer Input**
- o **Results**

ALARA CONSIDERATIONS
AP600

RADIATION GUIDELINES and CRITERIA

- o **Utility Requirements Document**
- o **Federal Regulations**
- o **ALARA Guidelines**
- o **Occupational Radiation Exposure (ORE) Predictions**
- o **Radiation Design Sources**
- o **Radiation Zone and Access Requirements**

ALARA CONSIDERATIONS
AP600

CUSTOMER FEEDBACK

o URD

o Design Reviews

o Joint Programs

o Evaluations

ALARA CONSIDERATONS
AP600

PROCESS RESULTS

Plant Simplification

o Large Diameter Containment

o Distinct Clean/Controlled Areas

o Large Hatch On Grade and On Operating Deck

o Close Coupled Buildings

o "Hot" Maintenance Shop

o Less Valves, Piping, Pumps

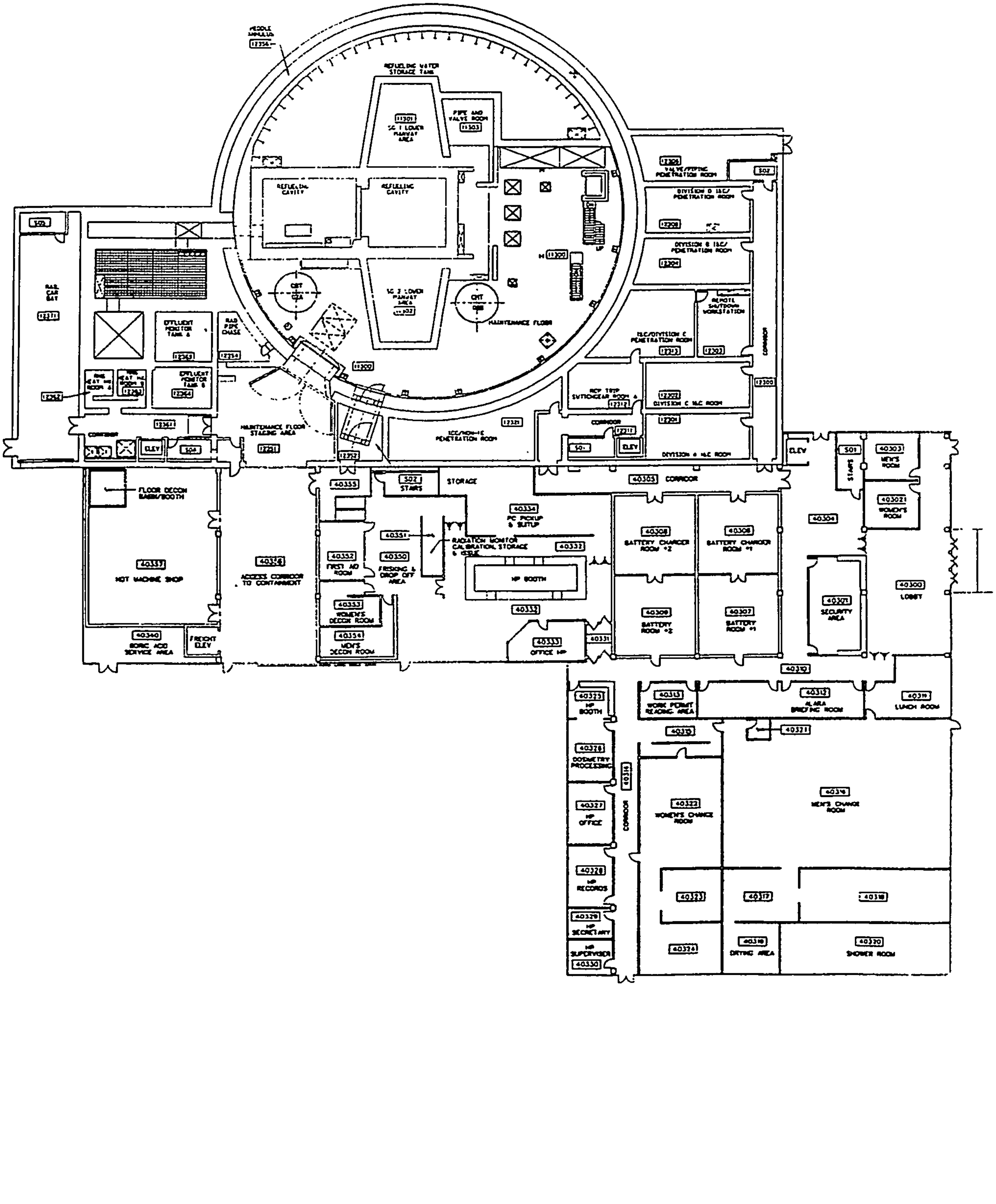
o Modular Chemical and Volume Control System

Containment Designed for Ease of Operation and Maintenance

- Improved access
  - Personnel airlock and maintenance hatch at both grade and operating deck level
- Truck access to both containment and annex buildings
- Size of containment increased to 130 feet diameter
  - Compared to 109 feet diameter of 600 MWe reference plant
  - Ample laydown space

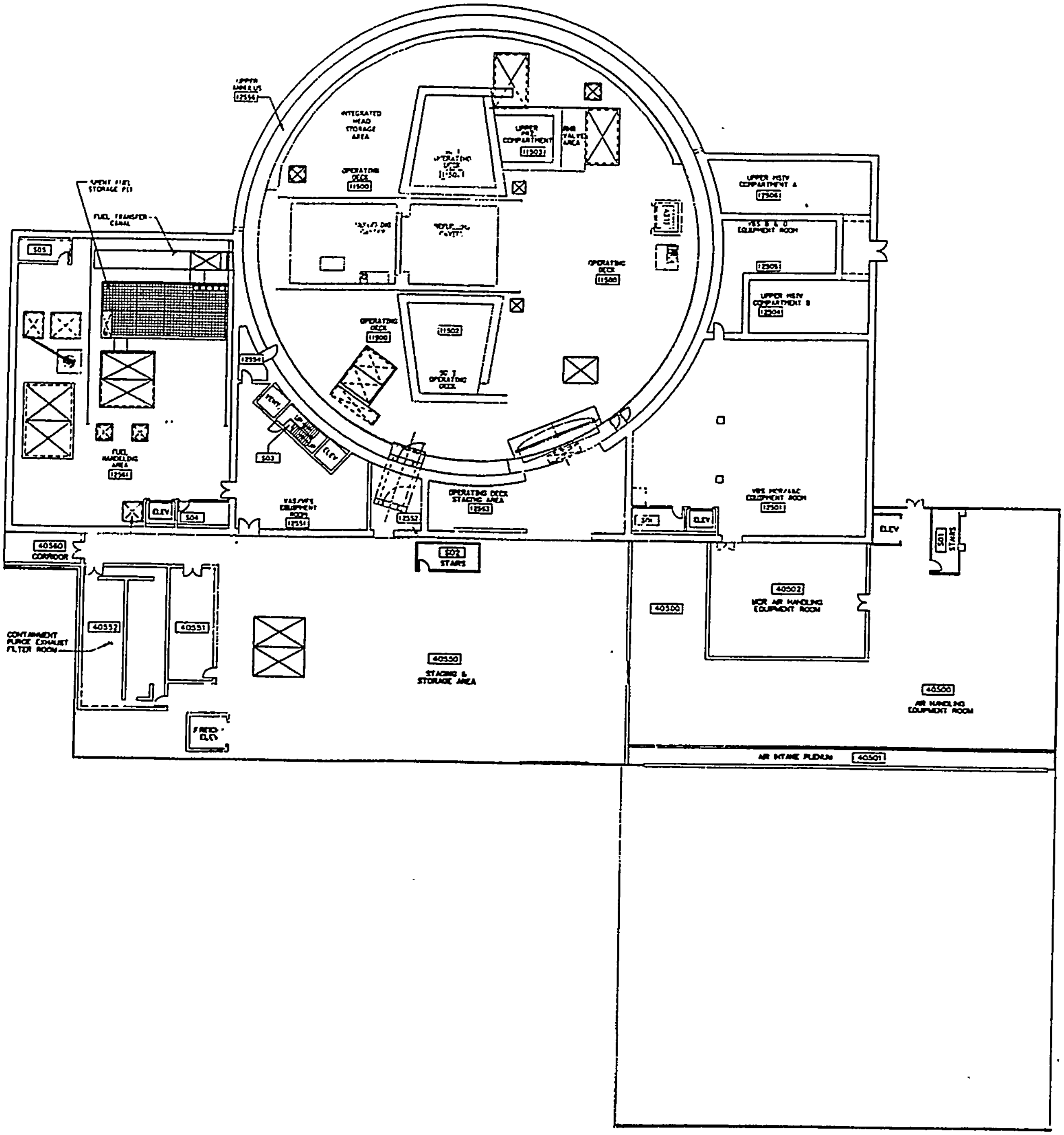
Maintenance/Laydown Areas

Elevation 100'/107'

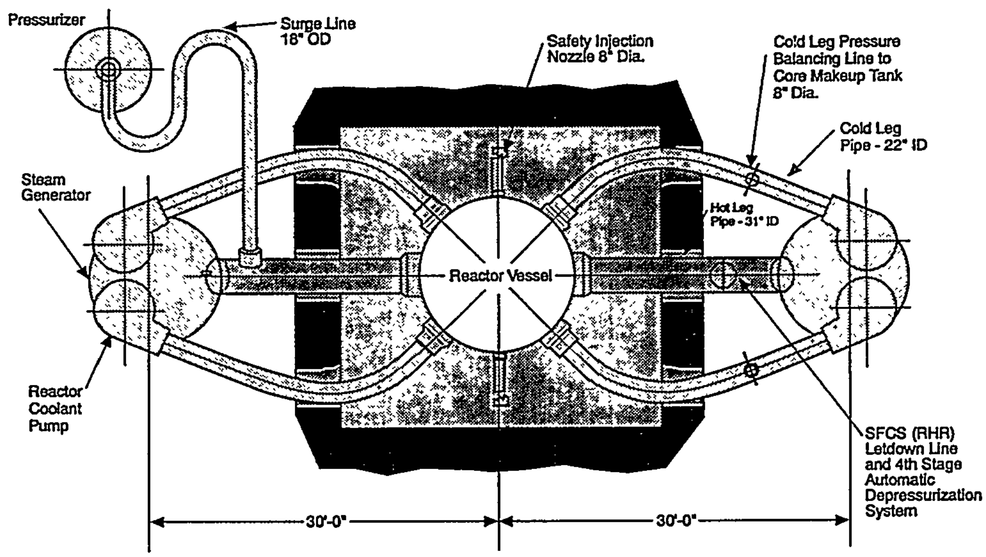


Maintenance/Laydown Areas

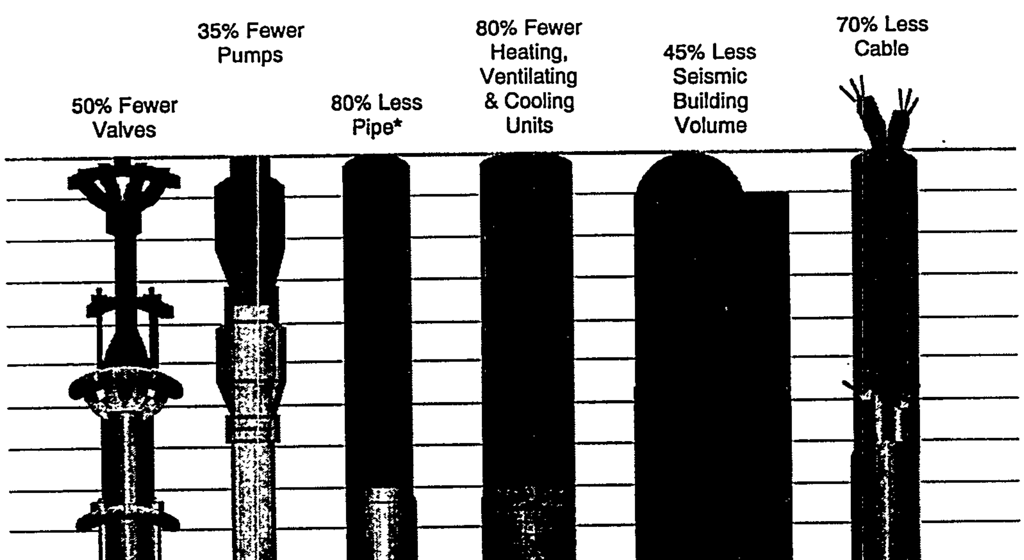
Elevation 135'-3"



AP600 Primary Loop Piping Plan View



AP600 — Simplified Design Based on Proven Technology



*Safety grade

Compared to a conventional, 2-loop 600 MWe plant

ALARA CONSIDERATIONS
AP600

PROCESS RESULTS

Waste Handling

- o Robotics
- o Remote Operations (Utility Co-op)
- o Cameras
- o Remote Resin Transfer
- o Local Flushing/Shielding Capability

ALARA CONSIDERATIONS
AP600

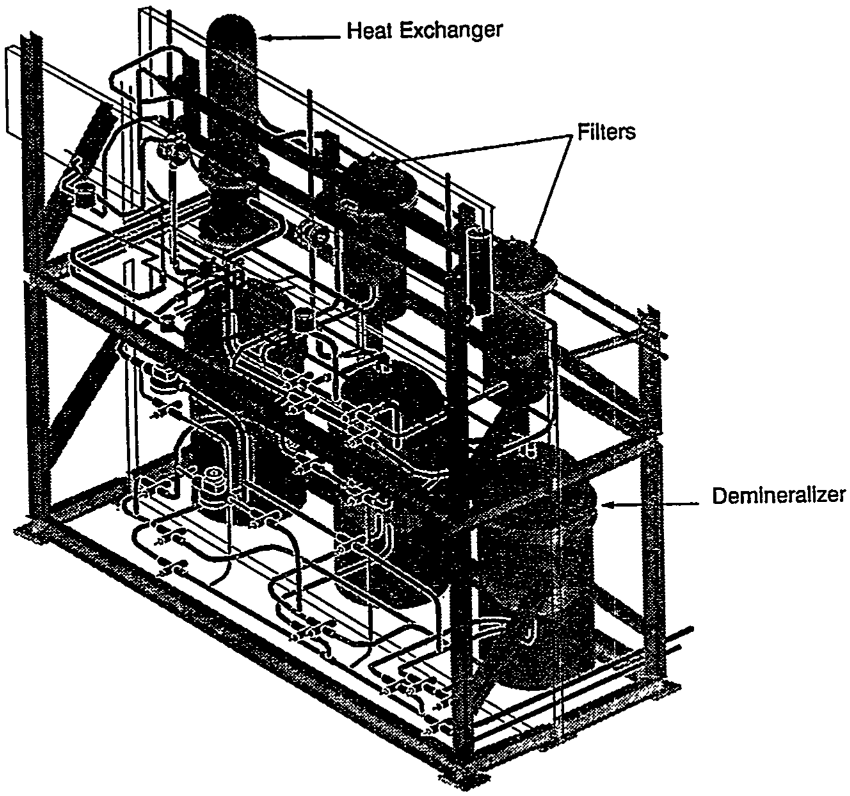
PROCESS RESULTS

Component Considerations

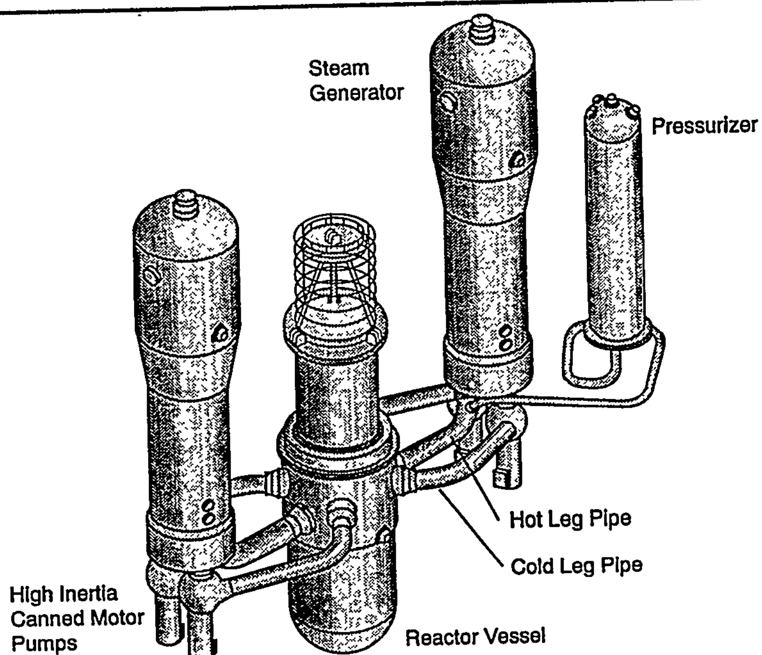
- o Reduced ISI
- o Reactor Coolant Pumps
  - Two per Loop (Canned)
  - Polished Impeller and Vanes
  - Rapid Removal/Transport System
- o Steam Generators
  -Inconel 690 Tubes
  -Low Cobalt Tubes (0.015 w%)
- o Air Handler Pumps
- o Spent Fuel Heat Exchanger Shield
- o CVCS Heat Exchanger Hatch

Innovative Construction Techniques Based on Pre-Engineered Modules

AP600 Chemical and Volume Control System Module



The AP600 Uses a Simplified Nuclear Steam Supply System



ALARA CONSIDERATIONS
AP600

PROCESS RESULTS

Fuel Considerations

- o Gray Rods
- o Zirconium Grid Straps
- o Reduced Power Density
- o Non Cobalt Rod Drive Latches

ALARA CONSIDERATIONS
AP600

PROCESS RESULTS

Shielding Considerations

- o Shielded Component Cubicles
- o Laydown/Access Shielding
- o Surge Line Shielding
- o Valve Gallery Shielding
- o Cavity Streaming Shield
- o Shielded Access Routes
- o Access Control Doors

ALARA CONSIDERATIONS
AP600

CONCLUSION

The AP600 annual average ORE will be significantly less than 70 Man Rem.

SYSTEM 80+™ STANDARD DESIGN INCORPORATES RADIATION PROTECTION LESSONS LEARNED

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ABSTRACT

Many lessons have been learned from the current generation of nuclear plants in the area of radiation protection. The following paper will outline how the lessons learned have been incorporated into the design and operational philosophy of the System 80+™ Standard Design currently under development by ABB Combustion Engineering (ABB-CE) with support from Duke Engineering and Services, Inc. and Stone and Webster Engineering Corp. in the Balance-of-Plant design. The System 80+™ Standard Design is a complete nuclear power plant for national and international markets, designed in direct response to utility needs for the 1990's, and scheduled for Nuclear Regulatory Commission (NRC) Design Certification under the new standardization rule (10 CFR Part 52). System 80+™ is a natural extension of System 80® technology, an evolutionary change based on proven Nuclear Steam Supply System (NSSS) in operation at Palo Verde in Arizona and under construction at Yonggwang in the Republic of Korea. The System 80+™ Containment and much of the Balance of Plant design is based upon Duke Power Company's Cherokee Plant, which was partially constructed in the late 1970's, but, was later canceled (due to rapid declined in electrical load growth). The System 80+™ Standard Design meets the requirements given in the Electric Power Research Institute (EPRI) Advanced Light Water Reactor (ALWR) Requirements Document. One of these requirements is to limit the occupational exposure to 100 person-rem/yr. This paper illustrates how this goal can be achieved through the incorporation of lessons learned, innovative design, and the implementation of a common sense approach to operation and maintenances practices.

INTRODUCTION

A common goal in the nuclear industry is to maintain personnel exposure as low as reasonably achievable (ALARA). The System 80+™ Standard Design has sought to incorporate those lessons learned by the current generation of nuclear power plants to achieve the EPRI ALWR Requirements Document's goal of limiting the personnel exposure to less than 100 person-rem/year.

The radiation protection philosophy of ALARA anchors a fundamental commitment to the safe operation of a nuclear power plant. This includes not only the protection of plant personnel, but also those who live and work in the surrounding communities. The concepts, outlined in Regulatory Guide 8.8, for maintaining occupational exposure ALARA are fundamental for the design, operation, and maintenance of a nuclear power plant. These concepts include time, distance, shielding and source term control.

The intent is to provide guidance for dose reduction. The concepts of time and distance are common sense. When performing an operational or maintenance activity, one should minimize the time spent in the radiation area. Conversely, one should maximize the distance between the personnel and the source of the radiation. Shielding or the placement of an absorbing material between the radiation source and the personnel should be used whenever possible.

Lastly, the control of the source term, can pay the best dividends for the reduction of dose. If the source term can be controlled through design improvements which reduce the sources of radioactivity and prevent the spread of contamination, then the exposure can be effectively reduced.

The following discussion will concentrate on these concepts and how these concepts have been implemented into the System 80+™ Standard Design.

WHERE ARE WE NOW?

Current operating nuclear power plants have developed ALARA programs for their respective plants. However, statistics have shown that the occupational exposures at Light Water Reactors such as McGuire, Catawba, and Oconee range between 167 to 310 man-rem/year on average per unit. A significant contribution to the overall occupational exposure is received while performing maintenance and station modification activities.

Reduction of occupational exposures could be achieved through the adherence to the ALARA concepts by an improved design, and better planning and execution of maintenance and operational activities. Unfortunately, redesign of the plant is not an option for most plants. However, the evolutionary plant designs ready for today's construction have implemented the wisdom from lessons learned in the current generation of nuclear power plants into their design.

DIRECTION FOR THE FUTURE

Design Features of System 80+™

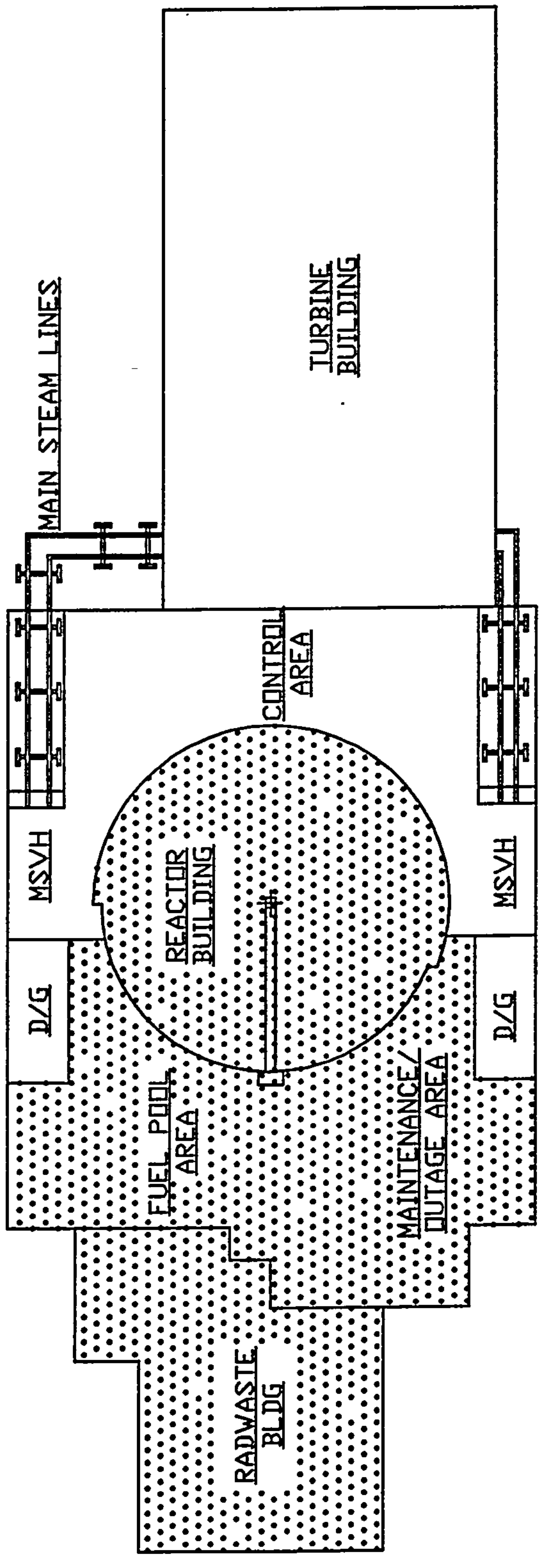
System 80+™ has incorporated many of the lessons learned by the current generation nuclear power plants. Each aspect of the ALARA philosophy of dose reduction, time, distance, shielding, and source term control has been considered in the design of the System 80+™ Standard Design. System 80+™ has unique design features that will result in a significant decrease in the occupational exposure.

General Arrangement

In the design of every plant careful consideration is given to the general arrangement of systems and their associated equipment. System 80+™ has incorporated some basic concepts into the general arrangement of the plant for dose reduction. These design features include the following:

Separation of radioactive systems from non-radioactive systems (See Figure 1) helps control the spread of contamination, minimize the necessity for routing piping containing radioactive fluids through personnel corridors, as well as the need for shielded pipe chases. It also simplifies the division of the plant into controlled and uncontrolled areas and aids in the unimpeded traffic through both the controlled and uncontrolled areas of the plant. Whenever possible, access to low radiation areas through high radiation areas is avoided; thereby, reducing the occupational exposure received.

FIGURE 1



SEPARATION OF RADIOACTIVE SYSTEMS FROM NON-RADIOACTIVE SYSTEMS

The Chemical Volume and Control System and the Fuel Pool Cleanup System are located in close proximity with the radwaste systems. This minimizes the pipe length and number of interconnections required to transfer of radioactive liquids, gases, or spent resin slurries, as well as the travel distance for transporting filters to the Solid Waste Management System. Again, this minimizes the length of piping that must be routed through access corridors.

Piping for radioactive systems is routed through shielded pipe chases. Ventilation, lighting, and adequate access area is provided for maintenance and inspection activities in the pipe chases. In addition, the number of active components located in the pipe chase has been minimized. This minimizes the frequency of access required into the pipe chase for maintenance activities.

The System 80+™ Standard Design has been designed to provide adequate spacing around equipment for easy access of equipment for maintenance. This includes provisions for an adequate laydown or equipment pull area, as well as a transport path for removal or replacement of equipment. Rigging and lifting equipment is also provided to facilitate the removal, transport, or placement of equipment or portable shielding during maintenance activities. This enables maintenance personnel to perform their task more efficiently reducing the time spent in a radiation area and therefore the dose. Space for maintenance laydown and access was one of the major reasons for selecting a large spherical containment for the System 80+™ design. The spherical containment with its 200 foot diameter provides considerable amount of floor space at the operating deck compared to cylindrical containments as illustrated in Figure 2.

Radioactive equipment are separated into compartments whenever possible. Equipment is compartmentalized based on frequency of access required, operational characteristics, and radiation level. For instance, ion exchangers containing resin beads are typically located in a separate compartment from the active components, such as pumps and valves. Valves are typically located in compartments called valve galleries.

Ion exchangers are located in pits with their associated spent resin service tanks located directly below the ion exchanger to minimize piping and the general area radiation. The compartment walls provide shielding. This enables personnel to perform operation and maintenance activities in a lower radiation area.

Hot tool cribs are located in low radiation areas adjacent to maintenance areas to minimize waiting times in high radiation areas, to help prevent the spread of contamination, and to decrease amount of decontamination work to be done; thus reducing radioactive wastes and personnel exposure.

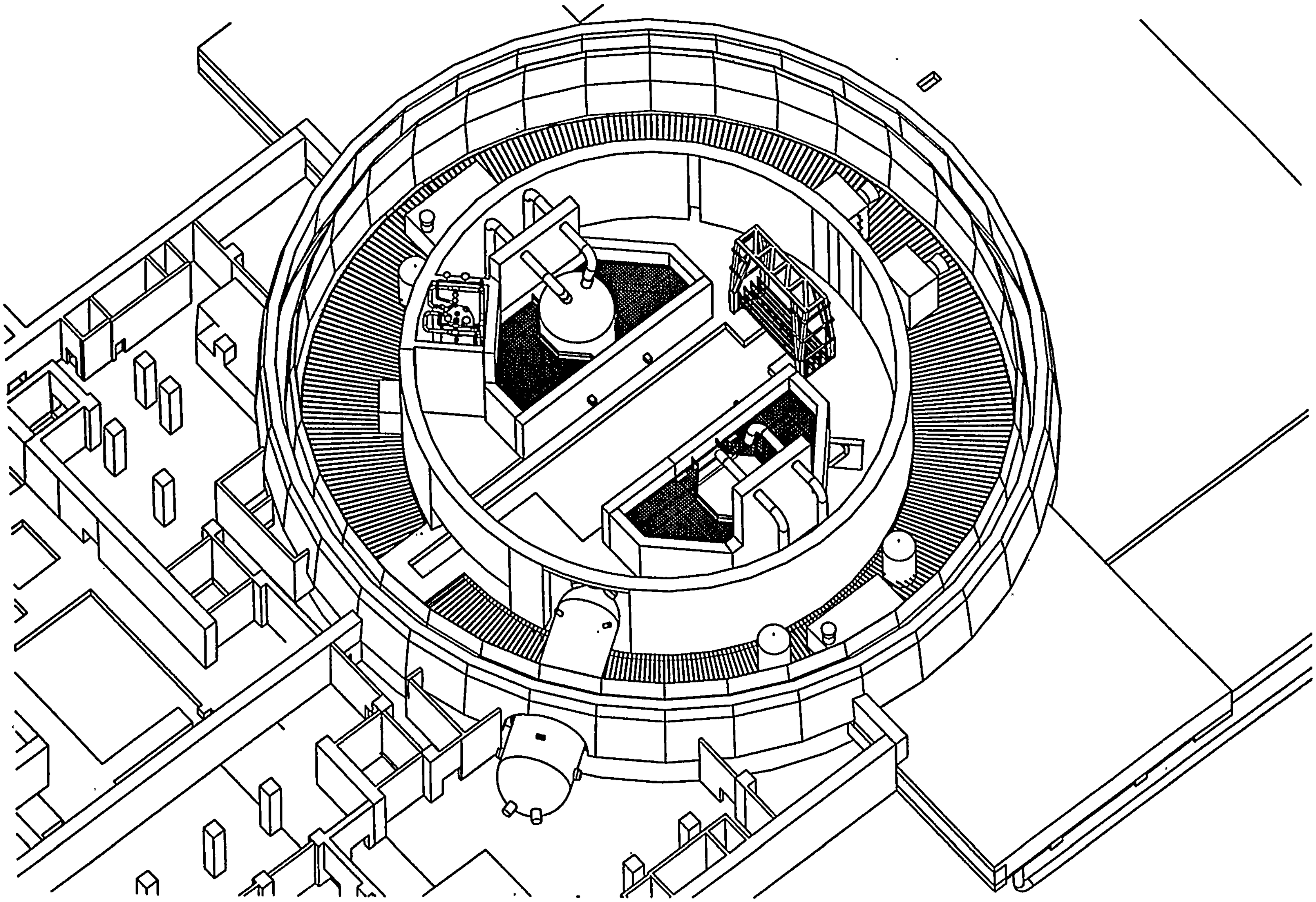
A hot machine shop is provided adjacent to the equipment hatch (See Figure 3). This enables personnel to remove equipment from containment and perform maintenance in a low radiation area, thus reducing the radiation exposure. Access from the hot machine shop is also provided to the truck bays and maintenance areas for ease of equipment movement.

Large staging areas inside and outside the equipment hatch and personnel airlocks allow pre-staging prior to the start of an outage as well as, provide space for efficient radiation controls in moving equipment in and out of containment (See Figure 3).

Change areas are located near airlocks to minimize personnel traffic flow, distance to the work area, and the potential for the spread of contamination (See Figure 3).

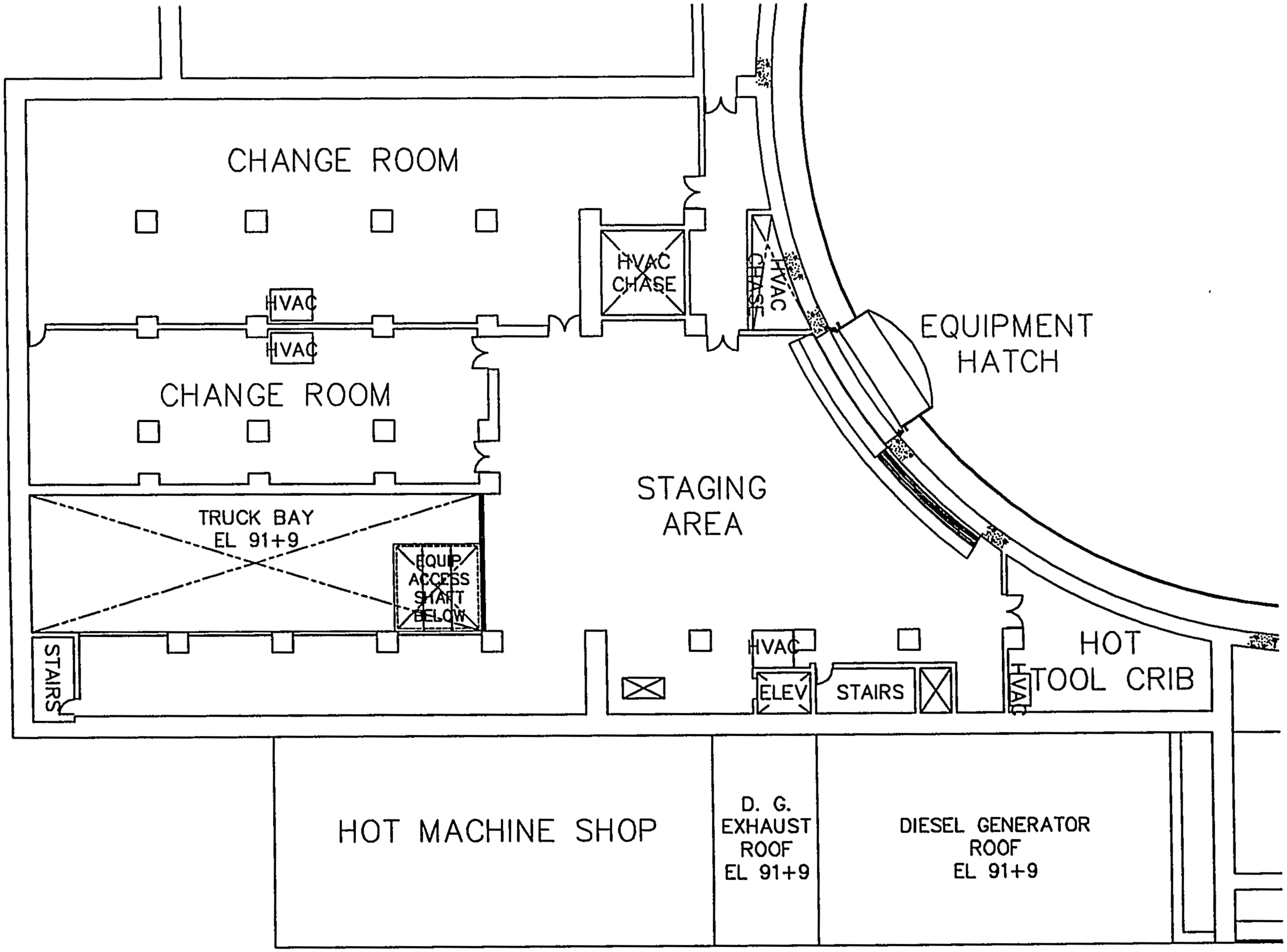
The access area to the Radiation Control Area (RCA) provides a flexible and adaptable layout, a large area (40' x 100') sufficient to accommodate outage work crews and the availability of immediate interaction with radiation protection personnel. This area provides a single point of access and egress for the RCA (See Figure 4).

Transient sources greater than 100 R/hr are considered in the System 80+™ shielding design to ensure adequate shielding is provided. One such source is a spent fuel assembly. During transfer of a spent fuel



FLOOR SPACE AT OPERATING DECK
ALLOWS ROOM FOR MAINTENANCE

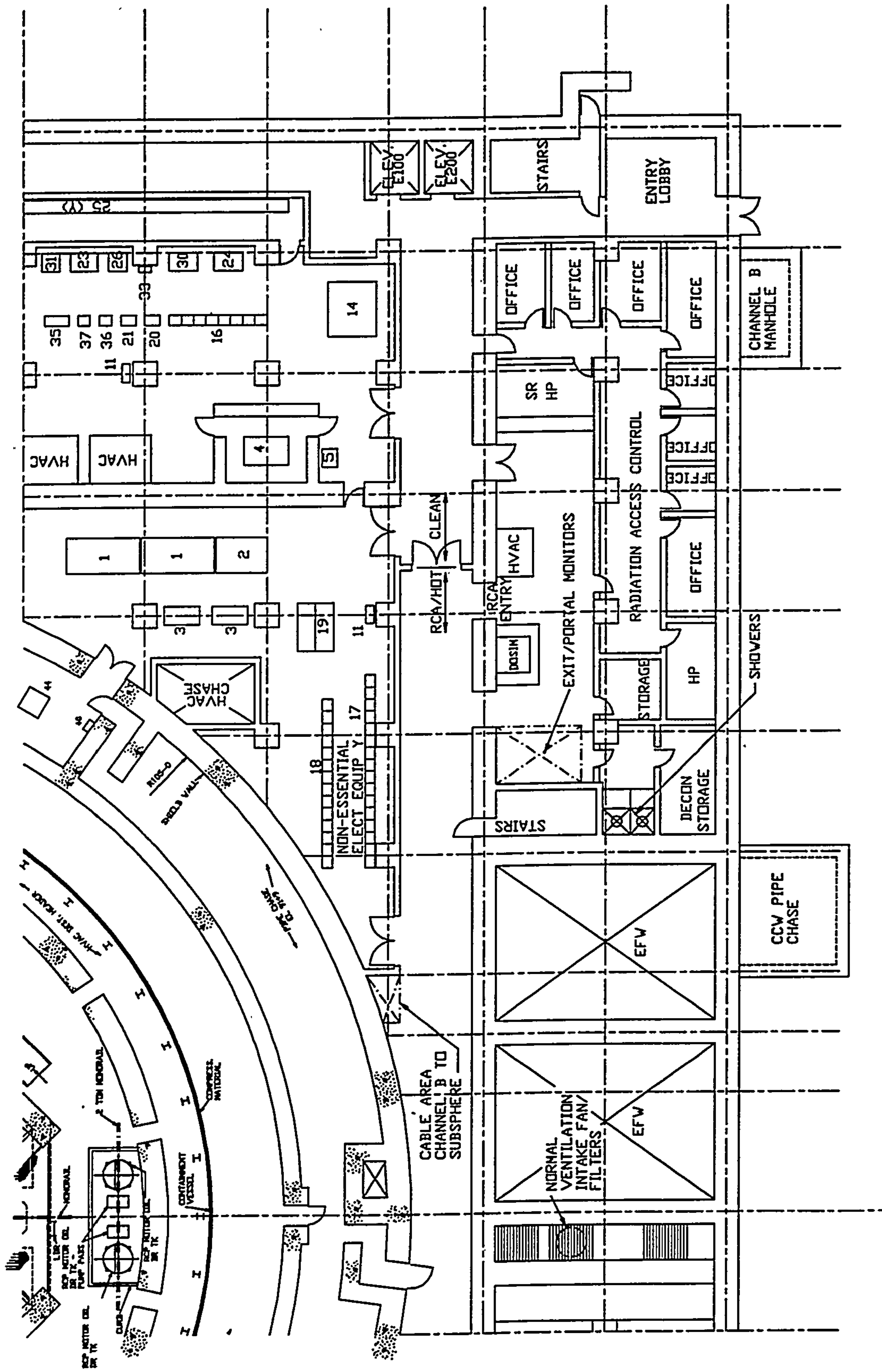
FIGURE 2



SYSTEM 80+ CONTAINMENT EQUIPMENT HATCH STAGING AND MAINTENANCE AREA

FIGURE 3

FIGURE 4



RADIATION ACCESS CONTROL AREA

assembly through the fuel transfer tube, adjacent corridors may experience elevated radiation levels. Streaming from this source up through the joint between the between the Reactor Building and the Nuclear Annex has been a concern for the current generation of nuclear plants. The System 80+™ design has the Reactor Building and Nuclear Annex on a common base mat which eliminates the shake space (joint) between the two buildings, thus the potential for streaming through this joint has been eliminated. In addition, a lead collar is provided around the fuel transfer tube, as well as several feet of additional concrete shielding to maintain adjacent corridor radiation level ALARA (See Figure 5). This permits personnel to perform maintenance and inspection activities in a lower radiation area and reduces the potential for adverse radiation zones from impacting refueling outage schedules.

The incore chase during incore instrumentation withdrawal is another potential area for a transient source greater than 100 R/hr. Positive access control is provided to this area during movement of the incore instrumentation. A lockable access door is provided with a warning light. During withdrawal of the incore instrumentation, the warning light illuminates providing indication that the incore instrumentation are being withdrawn. An area radiation monitor is located in the incore chase to provide indication of radiation levels and alarms when high radiation is in the area. An electrical interlock is provided between the radiation monitor and the access door to prevent access into the incore chase during withdrawal of the incore instrumentation. Emergency egress from the area is also provided.

Crud Control

Source term control is an important aspect of a nuclear power plant design. One-half to three quarters of the total dose received by personnel results from corrosion products or crud. Corrosion products result from activation of wear products or particulate in the reactor coolant as it passes through the core. Crud is then deposited in the reactor coolant system and interfacing system's piping and components.

Cobalt contributes significantly to the overall dose received during maintenance activities. Cobalt 58 and 60 are produced by the activation of materials containing cobalt and nickel impurities in primary system components.

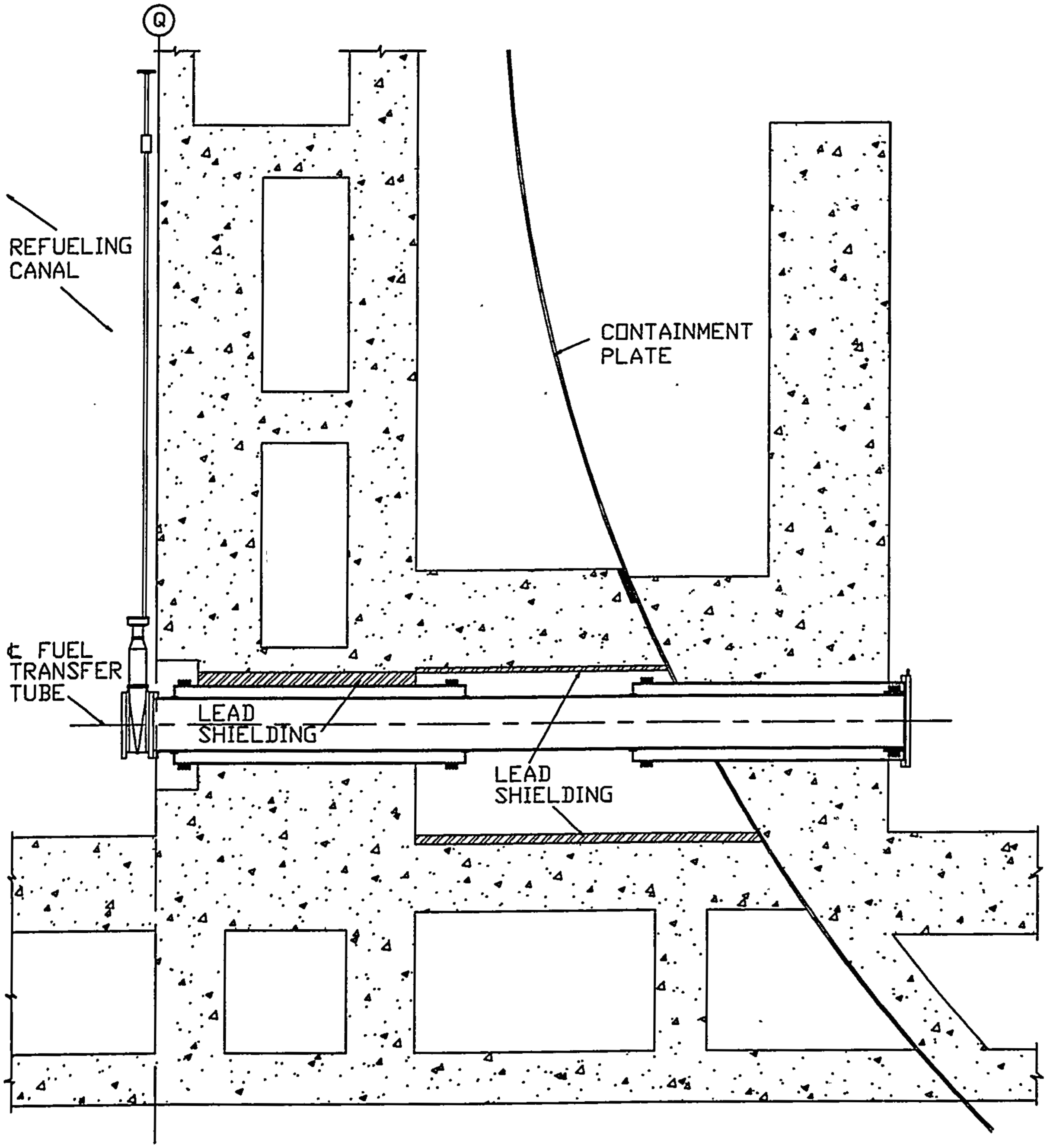
The presence of antimony in reactor coolant pump bearings has presented a problem with hot particles in the current generation of nuclear plants. The System 80+™ design minimizes the presence of antimony in the reactor coolant pump bearings.

To reduce the production of corrosion products in the primary system, System 80+™ Standard Design specifies that components in contact with the reactor coolant be fabricated from materials with low cobalt impurities ($\leq$ 0.020 w/o) and low corrosion rates. It has also been shown that an increase in reactor coolant water pH in the range of 6.9 to 7.4 reduces equilibrium corrosion rates and buildup of corrosion products on primary system surfaces. Therefore, the primary system chemistry will be operated in this range.

Equipment Selection and Design

System 80+™ specifies the use of more reliable and simplistic equipment. For instance, the use of evaporators in radwaste systems for decontamination of process flow streams in current plants has resulted in increased personnel exposure. The increase personnel exposure is primarily due to complexity of the system and the increased frequency of maintenance required by the system. The System 80+™ Standard Design has minimized the use of evaporators. With the exception of the Chemical and Volume Control System, which utilizes evaporators for boron recycle, ion exchangers are used almost exclusively for decontamination and purification of process flow streams. Ion exchangers are not only more simplistic in design, but are also more reliable and efficient. These design features reduce the frequency of maintenance and the dose received by personnel.

FIGURE 5



FUEL TRANSFER TUBE

Reactor Coolant Pump Seals

The System 80+™ reactor coolant pump seals are a cartridge type of design. This enables maintenance personnel to remove the seal as a unit and assemble and bench test it outside of high radiation areas. In addition, adequate platforms (See Figure 6) and space are provided for reactor coolant pump maintenance in addition to fixed lifting devices designed specifically for seal replacement.

Steam Generator Maintenance Improvements

Several improvements have been made in the System 80+™ design of the steam generator (S/G) to facilitate inspection and maintenance activities thus reducing the radiation exposure to maintenance and inspection personnel. These include the sizing and location of the manways, provisions for handholes and an internal hatch in the S/G, as well as the use of an improved S/G tube material.

The S/G tubes are Inconel 690 and are fabricated with the latest proven techniques to minimize residual stresses. This, along with maintaining appropriate secondary water chemistry, will greatly reduce the number of tubes required to be plugged; thus exposure time due to tube plugging is reduced.

The size of the manways are 21 inches. There are a total of four manways, two on the primary side and two on the secondary side. These manways provide access for eddy current testing and for inspection of the separator dryer, respectively. The steam generator manway locations are optimized for use of remote manipulators for inspection and maintenance. In addition, adequate platforms (See Figure 6) are provided to support S/G maintenance and inspection.

An internal hatch is also provided to access the top of the tube bundle for inspection and maintenance activities. At the tubesheet elevation, two eight inch handholes are provided. These provide access for tubesheet sludge lancing. They are also utilized to remotely inspect and retrieve loose parts. In addition, a twelve inch diameter access opening in the S/G support skirt is provided with removable insulation around that opening to facilitate inspection of the welds. Primary head draining capability is also provided which enhances accessibility for inspection and maintenance activities.

Reactor Vessel Head Vent

A vent nozzle and line is provided on the reactor vessel head to allow venting the gases to the pressurizer relief tank rather than the containment atmosphere; thus, reducing exposure during the head removal process.

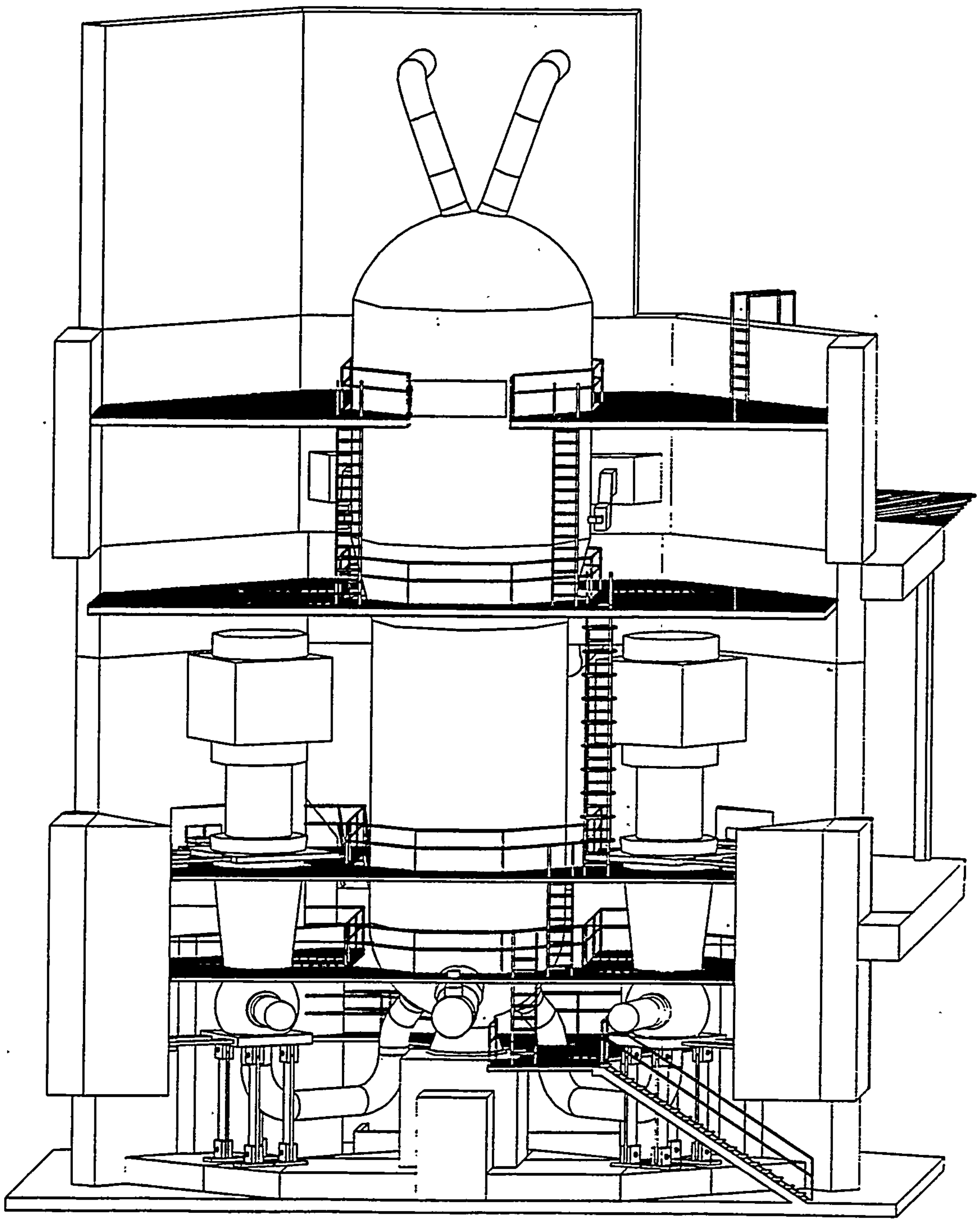
Ion Exchangers

Ion exchangers are designed for complete drainage. Spent resin removal is removed remotely by hydraulic flushing from the vessel to the Solid Waste Management System. Fresh resin addition is accomplished from a low radiation area above the shielded compartment housing the ion exchanger.

Filters

Filter housings are provided with vent connections and are designed for complete drainage. Filter housings and cartridges are designed to permit remote removal of filter elements. Cartridge filter seals are an integral part of the filter cartridge so that seal removal is accomplished during cartridge removal. Cartridge filter housing closure heads are designed to swing free for the unobstructed removal of the cartridge.

FIGURE 6



PLATFORMS FOR REACTOR COOLANT PUMP AND STEAM GENERATOR MAINTENANCE

Tanks

Tanks are designed for complete drainage and are free of internal crevices and pockets. This is accomplished by providing either a convex or sloped bottom to the tank with the drain connection located at the lowest point. Vents are provided to facilitate the removal of potentially radioactive gases during maintenance. Non-pressurized tanks are provided with overflows, routed to a floor drain sump or other suitable collection point to avoid spillage of radioactive fluids.

Valves

Except for modulating valve applications, packless valves are used on all valves two inches and under in diameter. Modulating valves and valves greater than 2 inches in diameter use live loading of the packing by conical spring washers or an equivalent means to maintain a compressive force on the packing where possible. Double stem packing with a leak-off between the packing is used for valves four inches in diameter and larger. Stem leakage is piped to an appropriate drain sump or tank. Valves utilizing stem packing are provided with backseat capability. Radiation resistant seals, gaskets, and elastomers are utilized, when practicable, to extend the design life and reduce maintenance requirements. Fully ported valves are utilized to minimize internal accumulation of crud.

Reduction of Linear Feet of Welds

A System 80+™ Standard Design goal is to reduce the total linear feet of welds in the reactor coolant pressure boundary. This reduces the time necessary to inspect these welds in radiation areas, thus reducing the radiation exposure to personnel.

System 80+™ Standard Design eliminates the use of longitudinal welds in reactor vessel through ring forgings. The in-service inspection of these welds in the past has been quite time consuming

Seamless piping is under consideration to be utilized in the System 80+™ Standard Design of the Reactor Coolant System to minimize the number of welds requiring in-service inspection. This would reduce the amount of time required to inspect the welds located in radiation areas, thus reducing the radiation exposure to personnel.

The inspection of the reactor pressure boundary can be done with remote equipment to minimize personnel exposure. The design of welds joining the reactor vessel nozzle to the reactor coolant pipe permits in-service inspection to be accomplished from the inside diameter of the reactor vessel. Automated equipment, operated remotely normally used for reactor vessel pressure boundary inspections, can be utilized in this area.

Blanket type thermal insulation is utilized, wherever practical, held in place by Velcro fasteners. A metal jacket around the insulation is provided and held in place by quick actuation type buckle fasteners. This insulation is easily removable and will facilitate the performance of in-service weld inspections, thus reducing the time spent in contaminated areas.

Radwaste System Design

Waste is segregated by radiation level, physical and chemical characteristics, and the type of waste (solid, liquid, or gaseous). By segregating the waste streams, the processes can be tailored to the unique characteristics of each waste stream. This improves the efficiency of the process and prevents the mixing of waste streams, thus minimizing the radiation exposure to personnel.

Radiation Monitoring

Radiation monitoring in the plant is an essential part of maintaining occupational exposure ALARA. System 80+™ Standard Design's Radiation Monitoring System provides early warning to station personnel of equipment, component, or system failures which may represent a potential radiological hazard via area radiation monitors and process system monitors.

Area radiation monitors provide essential information to the plant health physics staff planning operations or maintenance activities in radiation areas.

Process flow monitors provide for the continuous monitoring capability of gaseous and liquid effluent discharges from the plant. These monitors provide necessary information to estimate the dose consequence to plant personnel and the general public in the event of a design basis accident.

The design features of the Radiation Monitoring System enable the operator to monitor, assess, and evaluate information from a central location via a digital communications network. The digital communications network interfaces the Data Processing System (DPS) and Discrete Indication and Alarm System (DIAS) with each monitor microprocessor. Via the DPS and DIAS systems the operators can obtain detailed information of monitor readings, alarm setpoints, and operating status. The digital communications network enables the operator to access information on monitor configuration and historical trends, as well as diagnose problems from operation status alarms.

In addition, the Radiation Monitoring System design enables the operator to control monitor operation from dedicated operator control modules. Dedicated operator control modules are utilized to change microprocessor database items, initiate certain monitor control functions, and change monitor alarms setpoints. These control functions include starting and stopping sample pumps, manual checksource actuations, monitor purge initiation, and moving the filter paper advance. This design incorporates state of the art technology in the design of the Radiation Monitoring System.

Adequate shielding is also provided for each process, effluent, and airborne monitor. This ensures that the required sensitivity is achieved to provide an accurate radiation level readout at the design background radiation level for the area.

Airborne Contamination Control

The primary means of airborne contamination control is through prevention of its generation. Part of the System 80+™ operational and maintenance philosophy will be to minimize the generation of airborne contamination. This is accomplished by first recognizing the sources of contamination and then implementing airborne contamination control techniques. Airborne contamination can be generated by normal leakage of valves, seals, pipe flanges, etc., as well as during maintenance activities such as welding, machining surfaces, and grinding. Control of this contamination and reduction of occupational exposure can be accomplished through the proper use of containment devices such as drip containment, glove bags, and tents.

Plant ventilation systems are designed to prevent the spread of airborne contamination and minimize the exposures to both plant personnel and the public. In the System 80+™ Standard Design, plant ventilation systems are designed so that flow is from areas of lower to areas of higher potential activity. Potentially contaminated areas will be maintained at slightly lower pressure than non-contaminated or clean areas. Areas that have a potential for high airborne radioactivity, such as fuel storage areas, will be maintained at a negative pressure to ensure no outflow of radioactivity into a clean area.

Shielding

Permanent and temporary shielding is an integral part of the System 80+™ Standard Design. Permanent shielding is used where possible. For instance, System 80+™ Standard Design provides shielding between redundant components of an operating system. This reduces the dose to personnel performing maintenance on one component while the other component is operating. The location and design of labyrinths are also considered in plant layout. Labyrinths are provided for entrances in all high radiation rooms.

Portable shields, such as lead blankets and pigs, will be used during maintenance activities if the total exposure, which includes exposure received during installation and removal of the shielding, is reduced. Removable/portable shielding in the form of bricks and concrete blocks will be avoided. Rigging and transport paths are also provided in the design of the System 80+™ Standard Design for the removal of the shielding as necessary.

INFORMATION MANAGEMENT SYSTEM

The System 80+™ design is provided on an Information Management System (IMS). This IMS provides an effective means of acquiring, storing, retrieving and manipulating the documents and data necessary to design, construct, startup, operate, and maintain the plant. The System 80+™ IMS utilizes an existing computer program that is in operation within Duke Power Company called PASCE as its base program. PASCE currently store two dimensional drawing data in a hierarchical database called PLANT-SCHEMA. Three dimensional drawing data is stored in a hierarchial database call PLANT-VIEW. The System 80+™ plant layout is currently provided on the three dimensional graphics/data model within PASCE PLANT-VIEW. This model can be used in future plant operations for entering and developing three dimensional dose maps (See Figure 7). These dose maps can be generated within PASCE which integrates the plant layout graphics with specific area information, such as dose rate and source location(s) measured and entered by health physics personnel. This information can be readily used by health physics personnel to estimate the dose, as well as by personnel in the field to effectively implement the ALARA principles of time, distance, and shielding during maintenance activities. In addition the three dimensional model can be utilized for maintenance personnel in preplanning their maintenance activities and locating electrical and service connections. This allows pre planning without entering the radiation protection area, thus reducing occupational dose.

Duke Power has modeled their McGuire and Catawba Nuclear Stations on PASCE and currently utilize PASCE for developing three dimensional dose maps and for maintenance and outage planning. PASCE also interfaces with a program called PASSPORT which allows work request to be developed including the dose information for the area, dress out and dosimetry requirements. Duke Power Company experience has found that PASCE has considerably reduced the time personnel spend in the radiation protection zone for visual inspection for planning purposes and thus has reduce operation dose.

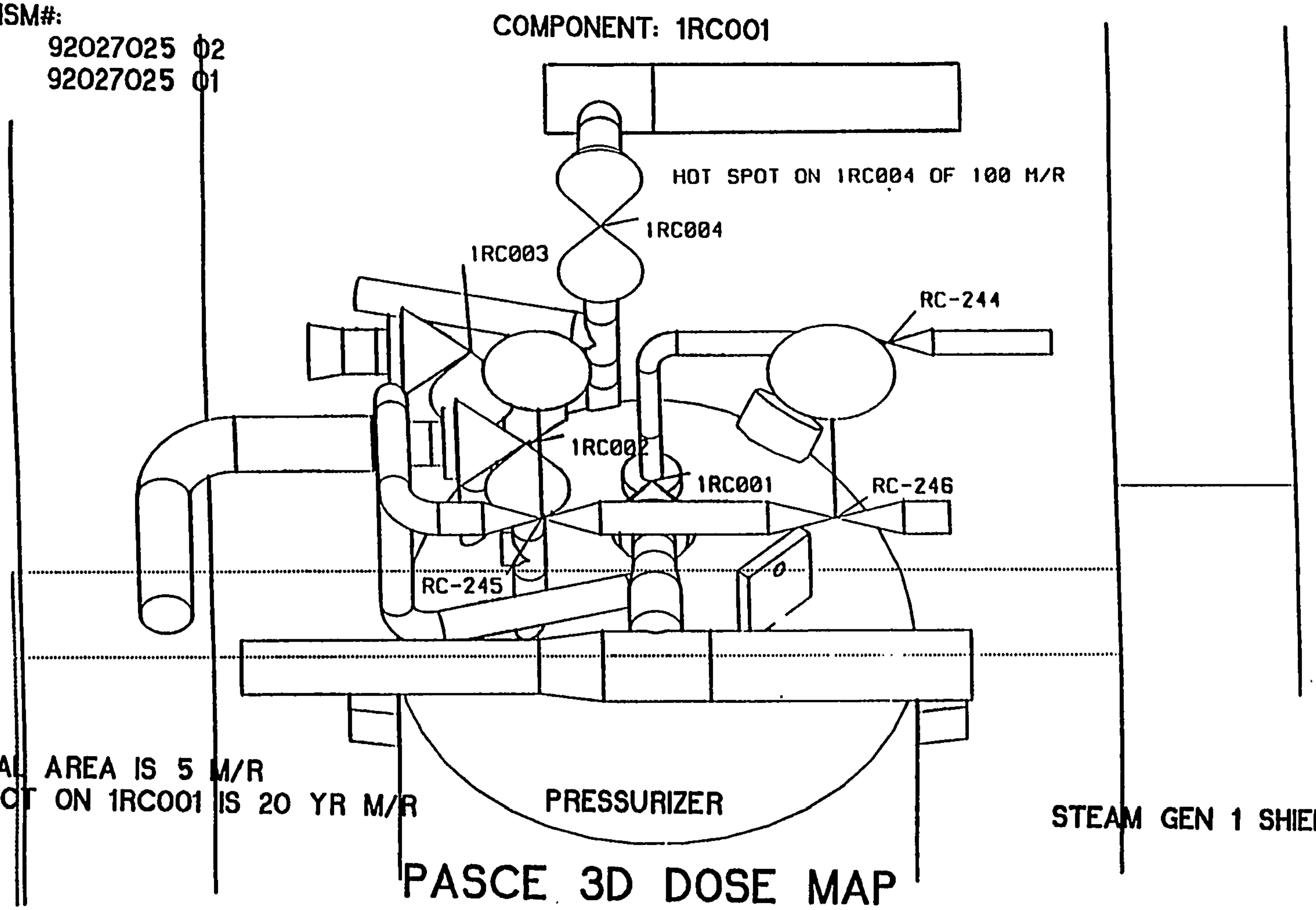
DOSE ASSESSMENT

Many design features which reduce operation exposure have been discussed above and implemented into the System 80+™ design. The two most important features which will reduce the annual exposure and allow the EPRI ALWR goal of 100 person-rem/yr to be meet are source term control through the reduction of cobalt and the steam generator material, fabrication, maintenance and inspection improvements that have been incorporated into the System 80+™ design.

The annual exposure at Duke Power Company's seven Pressurized Water Reactors (PWRs) for 1989 is shown in Table 1. This results in an annual average exposure of 235 person-rem per unit. This table also shows the percentage of annual exposure resulting from each major maintenance task.

SYSTEM 80+ NUCLEAR STATION
BASIC / DETAILED ALARA PLANNING WORKSHEET

MAINTENANCE MANAGEMENT PROCEDURE 1.9
ENCLOSURE 5.1
WO#/NSM#:
92027025 02
92027025 01



GENERAL AREA IS 5 M/R
CONTACT ON 1RC001 IS 20 YR M/R

PASCE 3D DOSE MAP
AND WORK OUTAGE PLAN

FIGURE 7

Table 1. PWR Reference Plant Data

	Oconee 3 Unit	McGuire 2 Unit	Catawba 2 Unit
Total Exposure (person-rem)	684	620	334
Average Per Unit (person-rem)	228	310*	167
Number of Refueling Outages	3	1	1
Refueling Exposure (% of Total)	77	67	67
Breakdown by Task (% of Total)			
Routine Operation and Maintenance	21	19	22
Steam Generator Inspection and Maintenance	20	36*	22
RV Head Inspection and Maintenance	19	5	8
Valve Maintenance	11	18	21
General Entry and Surveillance	8	5	11
Nuclear Station Modifications	5	6	1
Inservice Inspections	5	4	5
Reactor Coolant Pumps	3	3	5
Decontamination	8	4	5

*Impacted by Abnormal Occurrence, i.e., Steam Generator Tube Rupture.

One-half to three-quarters of all occupational exposures are related to exposure to activated corrosion products. Minimization of primary system corrosion and resultant dose rates will most effectively reduce total station occupational exposure, reduce effluent releases and radwaste activity. The combined effectiveness of proper material selection and chemistry control has reduced dose rate fields by as much as a factor of 5 in existing PWRs.

Leakage of fuel rod cladding accounts for the remaining one-quarter to one-half of sources of PWR occupational exposure. The fuel rod performance of the System 80+™ design is expected to reduce fuel leakage to below 0.1%. This is a factor of 2 to 4 better that average PWR fuel clad performance. This feature also is expected to reduce effluent releases and radwaste activity.

It is expected that proper material selection, chemistry control, and improved fuel cladding leakage will reduce the Annual Occupational Exposure by a factor o 2.5 compared to Duke Power Company's Oconne, McGuire and Catawba plants. These plants currently try to reduce operational exposure through pH control of the RCS and removal of fuel rods that have excessive leakage during refueling. However, these plants did not consider material selection of low cobalt content during their construction. As stated above the System 80+™ design is committed to selecting materials that have a cobalt content of less than 0.02 w/o for piping and equipment in direct contact with the RCS. The steam generator tubes will have a cobalt content of less than 0.015 w/o.

The design features described above for steam generator maintenance and inspection will greatly reduce time spent in performing this maintenance and inspection. This task represent approximately 25% of the total station dose for the average PWR system.

The dose received during an outage due to steam generator maintenance is dependent on the number of tubes requiring inspection and repair. Duke Power Company's McGuire Units 1 and 2 and Catawba Unit 1 steam generators have seen a considerable amount of primary side tube stress corrosion cracking. This results in a considerable amount of time inspecting and repairing the steam generator tubes by plugging or sleeving. In addition, McGuire had a steam generator tube rupture which has resulted in the performance of 100% bobbin coil testing of the hot leg tubes. This has also resulted in finding considerably more tubes requiring repair. The number of steam generator tubes to be inspected is dependent on the past history of the steam generator and previous problems. A plant in which the steam generators have not seen significant problems with tube cracking are only required to perform a 20% bobbin coil eddy current testing.

The McGuire and Catawba steam generator problems can be attributed to the lack of thermal treatment of the steam generator tubes and other manufacturing techniques. The System 80+™ steam generator design provides thermal treatment of the steam generator tubes and has avoided the manufacturing techniques that have caused past problems with steam generator tube cracking. In addition the steam generator tubes are fabricated from Inconnel 690 compared to Inconnel 600. Inconnel 690 has proven to be less susceptible to steam generator tube cracking.

The Palo Verde steam generator tubes are fabricated of Inconnel 600 and are thermally treated. Palo Verde has not seen the tube cracking problems experienced by McGuire and Catawba. This has resulted in a considerable difference in the annual dose received for steam generator tube inspection and repair. Palo Verde has seen an average refueling outage exposure due to steam generator maintenance of 39 person-rem compared to Duke Power Company's average of 62 person-rem. Therefore, a factor of 1.6 reduction in annual occupation exposure is expected for steam generator maintenance from these improved manufacturing techniques and material selection compared to the Duke Power Company average. The System 80+ has several additional features which are not provided on Palo Verde or the Duke Power Company units as described in Section 4.1.3.2. These features will reduce the time spent performing the steam generator maintenance and thus, are expected to reduce the annual occupational exposure due to steam generator inspection by a factor of 1.5. In addition a factor of 2.5 reduction is expected from reduction in cobalt, chemistry control and fuel performance as described above. Therefore all of the above System 80+ features will result in an overall reduction of the annual occupational exposure due to steam generator maintenance of a factor of 6.

The System 80+™ estimated annual occupational exposure is shown in Table 2. It is estimated that the System 80+™ annual occupational exposure will be 79 person-rem/yr or less with proper operational ALARA techniques. This is less than the EPRI ALWR goal of 100 person-rem/yr.

Table 2. System 80+ Estimated Annual Occupational Exposure

Task	PWR Avg. (%/man-rem)	ALARA Reduction	System 80+ Estimate (man-rem)
Routine Operation and Maintenance	21/48	2.5	19
Steam Generator Inspection and Maintenance	25/62	6	10
RV Head Inspection and Maintenance	12/27	2.5	11
Valve Maintenance	16/37	2.5	15
General Entry and Surveillance	8/17	2.5	7
Nuclear Station Modifications	4/11	2.5	4
Inservice Inspections	5/11	2.5	4
Reactor Coolant Pumps	3/8	2.5	3
Decontamination	6/14	2.5	6
Total	100/234		79

CONCLUSION

Many of the dose reductions techniques highlighted in this discussion are common sense. These dose reduction techniques are not new. However by applying these techniques early in the design stage, the cumulative effect of their implementation in the design of the System 80+™ Standard Design, as well as use of state of the art equipment, will have a long term beneficial impact on the occupational exposure and the radiation protection of the personnel and the public.

Author Biography

Thomas D. Crom is an Engineering Supervisor at Duke Engineering & Services, Inc. and is Duke Engineering's Project Manager for ABB-CE's System 80+ Design Certification Project. Mr. Crom has been directly involved with this project and the associated development of the Electric Power Research Institute Utilities Advanced Light Water Reactor Requirements Document since November 1985. Prior to working on advanced light water reactor projects he was involved in system design, pipe stress analysis and pipe support restraint design for Duke Power Company's Catawba Nuclear Station. He has a B.S. in Mechanical Engineering from Virginia Polytechnic Institute and is a registered Professional Engineer in the states of North and South Carolina.

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PAPER 3-5
DISCUSSION

Klazura: You mentioned that the System 80+ is the first plant licensed to the new source terms. Does that include the design basis normal operations source terms. In other words, is shielding designed to a quarter percent failed fuel instead of the one percent failed fuel?

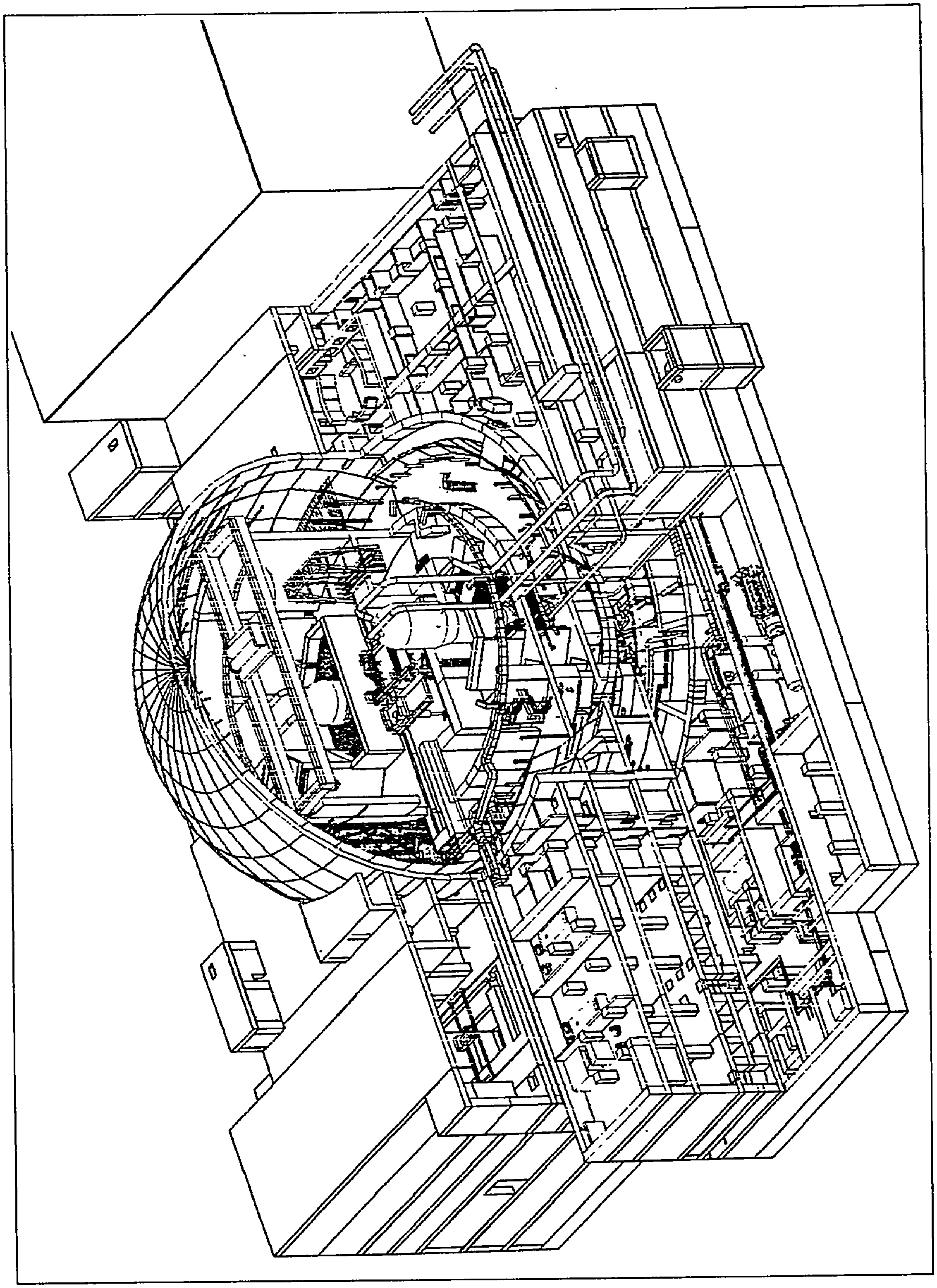
Crom: Shielding design is done to one quarter percent failed fuel. We had to do effluent analysis for rad waste systems to one percent failed fuel for the 10 CFR 20 limits. The new source term I was talking about is more for the design basis accident. System 80+ is the first plant to go through the licensing process utilizing the new source term for off-site dose for Chapter 15, and is also used for the shielding design post-accident.

Na: You have made a lot of improvements to steam generators. Why don't you consider, for the future, to replace steam generators. I saw your steam generators slide down and go in. Once you made it that way, it would be nice if you could make some configuration that can bring it back for future replacement. Did you consider that?

Crom: I was showing the removal, but the replacement would go through the same means.

Na: The same out?

Crom: Yes. Through the same equipment hatch. One piece.



EPRI ALWR REQUIREMENT DOCUMENT GOAL

FOR PERSONNEL EXPOSURE

<100 MAN-REM/YEAR

DUKE POWER 1989 OPERATIONAL EXPOSURES

CATAWBA NUCLEAR STATION	167 MAN-REM
OCONEE NUCLEAR STATION	228 MAN-REM
MCGUIRE NUCLEAR STATION	310 MAN-REM

System 80+™ Standard Plant
Radiation Protection Features

- Principles incorporated in System 80+™ Design
  - Lessons learned from current generation of nuclear plants
  - ALARA principles (e.g., time, distance, shielding, and source term control)

System 80+™ Standard Plant
Radiation Protection Features (Continued)

- System 80+™ Design Features
  - General arrangements
  - Equipment design and selection
  - Source term and contamination control
  - Shielding design
  - Radiation zone drawings and designations
  - Access control to transient high radiation areas (>100 R/hr)
  - Radiation protection design acceptance criteria

System 80+™ Standard Plant
General Arrangement Features

- Radioactive equipment separated into compartments such as:
  - Valves
  - Ion Exchangers
- Segregation of non-radioactive from radioactive systems
- Chemical and volume control and fuel pool cleanup systems in close promximity to radwaste systems
- Shielded pipe chases provided
- Penetrations located to minimize streaming
- Adequate rigging and lifting equipment provided

System 80+™ Standard Plant
General Arrangement Features (Continued)

- Adequate space for maintenace and inspection activities
- Hot machine shops and hot tool cribs located in low radiation areas adjacent to maintenance areas
- Large staging areas inside and outside equipment hatch
- Access area to RCA provides:
  - Single point access and egress to RCA
  - Immediate interaction with radiation protection personnel
  - Large area (40' x 100') for maintenance crews
- Change areas located near airlocks

System 80+™ Standard Plant
Equipment Selection Considerations

- Use of reliable and simplistic equipment (e.g., minimization of use of evaporators for decontamiation)

System 80+™ Standard Plant
Equipment Design Features

- Reactor coolant pump seal replacement
  - Use of cartridge type RCP seals
- Steam generator maintenance, tube inspection, and plugging
  - Location and size of manways adjusted
  - Addition of hand holes
  - Use of removable insulation
  - Improved material selection, Inconnel 690
  - Improved fabrication techniques of S/G tubing to minimize residual stresses
- Provision for platforms around major equipment (e.g., S/G, RCPs)

System 80+™ Standard Plant
Source Term Control

- Corrosion product control
  - Primary chemistry control (increase pH 6.9 to 7.4)
  - Material selection of components in contact with the reactor coolant with low cobalt impurities (<.050 w/o cobalt)
  - Cobalt-baes hardfacing alloys to provide wear resistance only used in applications where no proven alternative exists (EPRI NP-6737, Cobalt Reduction Guidelines followed)
  - Provision for flushing capability for slurry or resin transfer lines
  - Minimization of stagnant legs
- Improved fuel performance
- Minimize presence of antimony in RCP bearings

System 80+™ Standard Plant
Contamination Control

- Ventilation systems designed to provide air flow from areas of lower contamination to areas of higher contamination
- Containment of spills
  - Curbing
  - Sumps

System 80+™ Standard Plant
Shielding Design Features

- Adequate shielding to ensure personnel exposures are ALARA (i.e., total estimated annual occupational exposure <100 man-rem/yr)
- Shielding between redundant radioactive components
- Controlled access to high radiation areas:
  - Lockable access doors
  - Labyrinth entrances
- Use of portable shielding during maintenance (e.g., lead blankets)

System 80+™ Standard Plant
Access Control Features

- Access control features added to protect against transient sources >100 R/h (i.e., fuel transfer tube inspection area and incore instrumentation chase):
  - Area radiation monitors located at entrance to fuel transfer tube inspection area and inside incore instrumentation chase
  - Lockable access doors
  - Electrical interlock between area radiation monitor and access door to incore instrumentation chase

RADIATION MONITORING

AREA RADIATION MONITORS

PROCESS FLOW MONITORS

INFORMATION FROM CENTRAL LOCATION

- DATA PROCESSING SYSTEM
- DISCRETE INDICATION AND ALARM SYSTEM
- DEDICATED OPERATOR CONTROL MODULES

ADEQUATE SHIELDING

System 80+™ Standard Plant
Operational and Maintenance Techniques

- Pre-planning of maintenance activities
- Training on mockups
- Use of robotics, remotely operated equipment, and video
- Utilizaton of Information Management System for maintenance and outage planning
  - PASCE PlanView - 3D model
  - 3D Dose Maps
  - Interface with PASSPORT for work request development

PWR DESIGN FOR LOW DOSES IN THE UNITED KINGDOM: THE PRESENT AND THE FUTURE*

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ABSTRACT

The Pressuriser Water Reactor (PWR) design chosen for adoption by Nuclear Electric plc was based on the Westinghouse Standard Nuclear Unit Power Plant System (SNUPPS). This design was developed to meet the United Kingdom (UK) requirements and those improvements are embodied in the Sizewell B plant.

Nuclear Electric plc is now looking to the design of the future PWRs to be built in the UK. These PWRs will be based as replicas of the Sizewell B design, but attention will be given to reducing operator doses further.

This paper details the approach in operator protection improvements incorporated at Sizewell B, presents the estimated annual collective dose, and identifies the approach being adopted to reduce further operator doses in future plants.

INTRODUCTION

Nuclear Electric is a Government-owned utility which owns and operates all the commercial nuclear power stations in England and Wales. The bulk of these stations are gas cooled reactors but Nuclear Electric is currently building its first commercial PWR station, Sizewell 'B', in Suffolk, England. The station is now under commissioning and it is intended to be in full commercial operation in the second half of 1994.

Sizewell 'B' is based upon the SNUPPS design and there are two such plants already operating in the USA at Callaway and Wolf Creek. The Sizewell 'B' design has been developed to include additional safety features required to address UK licensing requirements. This includes additional fault mitigation equipment to address potential public release concerns and design improvements to reduce operator dose uptake.

With Sizewell 'B' nearing completion, Nuclear Electric is already looking to the design of any future station to be built in the UK. These future plants would be built as replicas of the Sizewell 'B' design, but attention would still be given to reducing operator doses further. This has the dual benefit of providing the maximum operational flexibility to the station in achieving the very low operator dose targets and also ensuring operator doses are As Low As Reasonably Practicable (ALARP). This paper outlines the approach in operator protection improvements incorporated at Sizewell B, presents the estimated annual collective dose and identifies the approach being adopted to reduce further operator doses in future plant.

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OPERATOR DOSE TARGETS

The whole body dose targets which have been adopted by Nuclear Electric for their PWRs have been derived from consideration of operator doses at their gas cooled reactors, from the requirements of the U.K. Nuclear Installations Inspectorate (NII) and the Public Enquiry for planning consent for Sizewell 'B'. The most important targets are the individual annual target of 10 mSv (1 Rem) and the station annual collective dose target of 2.4 man-Sv (240 man-Rem)

These targets were set when average PWR collective doses were about twice this value and hence were extremely demanding targets for a PWR at the time.

OPERATOR PROTECTION IMPROVEMENTS IN SIZEWELL B

A number of changes from the SNUPPS design have been incorporated in the Sizewell B design to reduce operator doses. The approach adopted in identifying the improvements was to reduce the radiation sources and source strengths, improve systems and layout to reduce the impact of the radiation sources on operator doses, and introduce remote equipment.

Examples of the improvements are:

(a) minimisation of radiation sources and source strength:

- reduction of Ni and Co impurities in materials; e.g.
    - replacement of stellite in the Chemical and Volume Control System (CVCS) control valves
    - use of Inconel 690 in Steam Generator (SG) tubing
- control chemistry at all stages
- remove crud traps
- etc.

(b) system and layout improvements:

- introduction of a permanent Reactor Pressure Vessel (RPV) cavity seal
- introduction of shielding in the Reactor Coolant Pump (RCP) seal change platform
- location of major items in individual shielded cells
- replace the wet Resistance Temperature Detection (RTD) system with alternative N-16 based measurements
- introduce platforms to improve access for maintenance
- carry out system and layout ALARP reviews to identify further improvements
- etc.

(c) use of remote equipment:

- Multi Stud Tensioner for the RPV stud removal/installation
- RPV flange and stud hole cleaning equipment
- all RPV and nozzle volumetric In Service Inspection (ISI) will be carried out by a submersible robot
- all volumetric ISI of welds in high dose rate systems will be carried out by automated equipment
- SG inspections eliminate the need for SG bowl man-entry
- etc.

The attention paid to operator protection improvements benefitted significantly from international operational experience and helps in ensuring that operator doses will be ALARP.

SIZEWELL 'B' OPERATOR DOSES

The Pre-Operational Safety Report (POSR) for Sizewell 'B' was submitted to the licensing authority in November of 1992, and it included an assessment of the operator doses for the station. The assessment was based on dose-rate data from the two SNUPPS plants Wolf Creek and Callaway and similar plants operated by Electricite de France (EdF). It then utilized the expected operator residence times to compute the operator doses for all the well defined tasks on the station. For the other tasks like Health Physics work and unplanned maintenance, overall reported dose information from relevant plants was used.

The assessment which was based on dose rate data for the coordinated 6.9 pH chemistry, concluded that for the planned 12 month fuel cycle the annual collective dose would be 1.97 man-Sv (197 man-Rem). The maximum individual annual dose was calculated to be 8.5 mSv (0.85 Rem).

The collective dose assessment was recognized as a conservative estimate because whilst the most applicable operational dose-rate information had been used it did not reflect all the source reduction steps taken for Sizewell 'B'. In particular, the benefits from the selection of Inconel-690 material for the SGs and the removal of stellite from CVCS valves were not included.

It was estimated that these improvements could reduce the annual collective dose to as low as 1.32 man-Sv (132 man-Rem) for the planned 12 month fuel cycle operation. Consideration of the dose savings from design and equipment improvements on Sizewell 'B' and from the adoption of higher pH chemistry would further reduce this value.

Following the dose assessment it was decided to implement the modified chemistry (2.2 ppm Li; 7.4pH_{300}) to benefit from the operator dose savings associated with this chemistry regime. The dose saving is estimated to be about 23% (see later sections).

FUTURE PLANT OPERATOR DOSES

The Sizewell 'B' design was effectively frozen in the mid to late 1980s when the majority of components were ordered. However, since then there have been significant steps forwards in both the operation of PWRs and in the reduction of the radioactive source terms. The following are considered to be the most important features for consideration for future plant:

Adoption of an 18 month fuel cycle

Adoption of Zircaloy fuel grids

Adoption of higher pH chemistry

Stellite removal

All of these are being considered for any future plant in the UK, as they can provide both significant financial advantages and greater operational flexibility in achieving lower dose targets.

An assessment of the operator doses has been carried out for the above changes to detail the potential dose savings. The assessment is again based upon operational data from the SNUPPS plants and has used the collective dose data for the periods of 18 month operation at Wolf Creek and Callaway. It then removed any dose not applicable to a Sizewell 'B' replica design due to design and equipment improvements. The benefit from the source term improvements on Sizewell 'B' in terms of material selection have also been recognized. The assessment then went on to recognize the benefit from the improvements beyond those already included in the Sizewell 'B' design.

Adoption of 18 Month Fuel Cycles

Wolf Creek and Callaway both started life with 12 month fuel cycles but Callaway changed to 18 month cycles from cycle 3 and Wolf Creek followed in cycle 4. The outage and operational doses for these plants have been averaged over a 3 year period which includes two refuelling outages, to provide an estimate of the operator doses. The annual average collective doses are 2.5 man-Sv (250 man-Rem) for Callaway and 1.8 man-Sv (180 man-Rem) for Wolf Creek.

These values are higher than would normally be expected, because they include one-time-only design changes to improve operation and operator doses. In particular both plants have replaced the Resistance Temperature Detection (RTD) System with an improved system. These replacements incurred doses of 1.5 man-Sv (150 man-Rem) at Callaway and 1.0 man-Sv (100 man-Rem) at Wolf Creek.

Design and Equipment Improvements

The SNUPPS dose records have been assessed to identify any dose incurred which is not applicable to a Sizewell 'B' type of design due to design and equipment improvements. The dose was reassessed where Sizewell 'B' did not have that type of equipment or when the task was carried out differently. Due to the way plants collect their dose information, it was not possible to identify all the dose for work not applicable to Sizewell 'B'. Therefore, operator dose was only removed in those cases where the dose was clearly identifiable and an exact comparison could be made.

The most notable saving is the deletion of the RTD system which was mentioned above and was replaced on Sizewell 'B'at the design stage.

The derived average annual collective doses for a Sizewell 'B' type plant recognizing the design and equipment improvements and operating an 18 month fuel cycle are 1.71 man-Sv (171 man-Rem) based on Callaway and 1.43 man-Sv (143 man-Rem) based on Wolf Creek. These are considered to be similar values and reflect the normal variability of operator doses and hence the two have been averaged to provide a representative value of 1.57 man-Sv (157 man-Rem). It is considered that this value will still be conservative as it has not been possible to identify all the dose savings for the Sizewell 'B' improvements.

Sizewell 'B' Material Improvements

The SG tube material for Sizewell 'B' (Inconel-690) is an improvement over the Inconel-600 used in the SNUPPS plants. The Inconel-690 has a reduced Nickel content and a very low Cobalt impurity specification (0.015 against 0.1 wt%). In addition, Sizewell 'B' has had the Stellite removed from 14 CVCS valves.

The impact of the SG material change was extensively reviewed by Westinghouse and it was concluded there will be a 35% improvement in dose-rate and hence operator doses. This results in a reduction of 0.55 man-Sv/yr (55 man-Rem/yr). Similarly the impact of Stellite removal in the CVCS valves has been assessed to be a 5% reduction in operator doses. This reduces the average doses by 0.05 man-Sv/yr (5 man-Rem/yr).

The two savings together give a reduction of 0.6 man-Sv/yr (60 man-Rem/yr) and would result in an average annual collective dose of 0.97 man-Sv (97 man-Rem). This applies for a Sizewell 'B' type plant operating an 18 month fuel cycle and recognizing all design, equipment and material improvements that can be readily and reliably identified.

Adoption of Zircaloy Fuel Grids

Historically, fuel grids have been made from Inconel which has a significant Nickel content and Cobalt as an impurity. However, more recently plants have been using Zircaloy fuel grids that have negligible Nickel and Cobalt content. At the time of ordering the first charge fuel for Sizewell 'B' it was considered that the Zircaloy grids were not a proven design and hence Inconel grids were selected. It is now considered that Zircaloy grids are a proven design and hence they will be adopted in any future design.

The improvement in operator doses from using Zircaloy has been estimated by Westinghouse and it was shown that plants with Inconel grids have dose-rates and hence doses which are typically 24% higher. This improvement relates to plants with dose values equal to or greater than the SNUPPS plants. Hence using the minimum value for a SNUPPS Plant in Section 4.2 above (1.43 man-Sv/yr) results in a saving of 0.27 man-Sv (27 man-Rem) each year. An independent assessment by Nuclear Electric assessed the saving to be 0.22 man-Sv/yr (22 man-Rem/yr). In order to be conservative a value of 0.22 man-Sv/yr has been used and this reduces the annual average collective dose to 0.75 man-Sv (75 man-Rem).

Adoption of Higher pH Chemistry

The main options for operating a primary circuit chemistry above the standard Coordinated 6.9 pH regime is either a Modified or an Elevated regime.

For the 12 month cycle the Modified regime is where the initial Lithium level is 2.2 ppm and it is held constant until a pH_{300} of 7.4 is achieved. The Lithium levels are then reduced in line with the Boron reductions to maintain a constant pH_{300} of 7.4. The Elevated regime is similar but the initial Lithium level is 3.35ppm and again this is held constant until a pH_{300} of 7.4 is achieved and then the Lithium is reduced to maintain a constant pH_{300} of 7.4.

For an 18 month cycle and depending on the initial Boron concentration, the Lithium level at the beginning of the cycle may need to be increased beyond the values above in order to keep the pH_{300} above 6.9. The choice of the precise chemistry regime for 18 month cycles is kept under review as operating experience becomes available.

The Modified regime has been successfully implemented at many plants and has shown dose improvements with no adverse effects on the plant. The Elevated regime is known to have been implemented at 13 plants. While these plants have shown higher dose savings, in the majority of cases, the plants have reverted to either a Modified or standard Coordinated regime. At this stage it is therefore considered that the Elevated regime is not a proven option and hence the Modified regime has been selected.

There have been independent assessments of the benefits of adopting a Modified regime by Westinghouse and Nuclear Electric. In both cases, operating plant data was used to estimate the benefit in dose-rate and hence operator doses. The Westinghouse assessment indicates an improvement of 25% and the Nuclear Electric assessment indicates an improvement of 23%.Based on this data, it is concluded that the adoption of modified chemistry regime will reduce the annual average collective dose from 0.75 man-Sv (75 man-rem) to 0.57 man-Sv (57 man-Rem).

Stellite Removal

Stellite is a major contributor to operator doses and substantial savings in dose can be made by removing the Stellite hard facings. This is strongly supported by the German Konvoi plant experience where the significant Stellite sources have been removed with no adverse impact on operation. The significant sources of Stellite are considered to be:

Stellite inside the RPV including the Control Rod Drive Mechanisms (CRDMs)

Stellite on all valves in circuits which can return water to the Reactor Coolant System (RCS). Exceptions are those valves with high stress loads such as the containment isolation valves

Stellite from Reactor Coolant Pump (RCP) bearings

The assessment of the impact of Stellite removal on operator doses indicates a saving of 0.22 man-Sv/yr (22 man-Rem/yr) further reducing the annual average collective dose to 0.35 man-Sv (35 man-Rem).

CONCLUSIONS

The design of the Sizewell 'B' plant has benefitted greatly from operational experience. Its anticipated annual average collective dose will be well below the POSR value of 1.97 man-Sv (197 man-Rem) when the benefit is recognized from the design, equipment and material improvements and from the adoption of the modified chemistry. It will then have one of the lowest operator doses of those plants with Stellite and operating 12 month fuel cycles.

However, there are still improvements to be considered in any new plant that is being designed. The above assessment shows that on a conservative basis it is possible to achieve annual average collective doses of 0.35 man-Sv (35 man-Rem) within the current understanding and technology.

The potential to achieve these low operator doses is supported by the German Konvoi Plant experience where there has been a progressive improvement in operating pH, incorporation of Zircaloy fuel grids and Stellite removal. There are similar improvements in operator doses as those described above and the Konvoi plants have an annual average collective dose of 0.18 man-Sv (18 man-Rem). These plants have not removed the Stellite from the CRDMs and they only operate a 12 month fuel cycle. This implies that the operator dose assessment above is conservative.

ACKNOWLEDGMENTS

This paper describes the potential steps that can be taken to improve operator doses on future PWR plant. The paper has only been possible because of the help of several people. Initially Russ Taylor of Wolf Creek and Ron Roselious of Callaway provided the details on operator doses. Then the experts in the areas of operator doses and chemistry provided the assessment of worldwide information on dose savings. In particular, Carl Bergmann of Westinghouse and Keith Garbett and Vic Polley of Nuclear Electric. I extend my grateful thanks to them all for their help in providing and analyzing the data and in developing the approach adopted.

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1. Willcock A., and Zodiates A. M., "Designing in Radiation Protection to Sizewell B", *Nuclear Engineering International, April 1992*, pg. 44-46, 1992.

Author Biography

Anastasios M Zodiates is a Radiological Protection (Operator) Engineer with the PWR Project Group, Nuclear Electric plc., England. He is involved with the development of the radiological protection aspects of the design and licensing of the Sizewell B PWR. Before joining the Sizewell B team he worked as a Reactor Physicist at the company's Heysham 2 Advanced Gas Cooled Reactor. He was born in the island of Aprodite, Cyprus, and attended university in England. He has a B.Sc. in Electrical Engineering from Southampton University, an M.Sc. in Nuclear Reactor Physics and Technology, and a Ph.D. in Physics from Birmingham University. He is a UK registered Chartered Physicist and Engineer.

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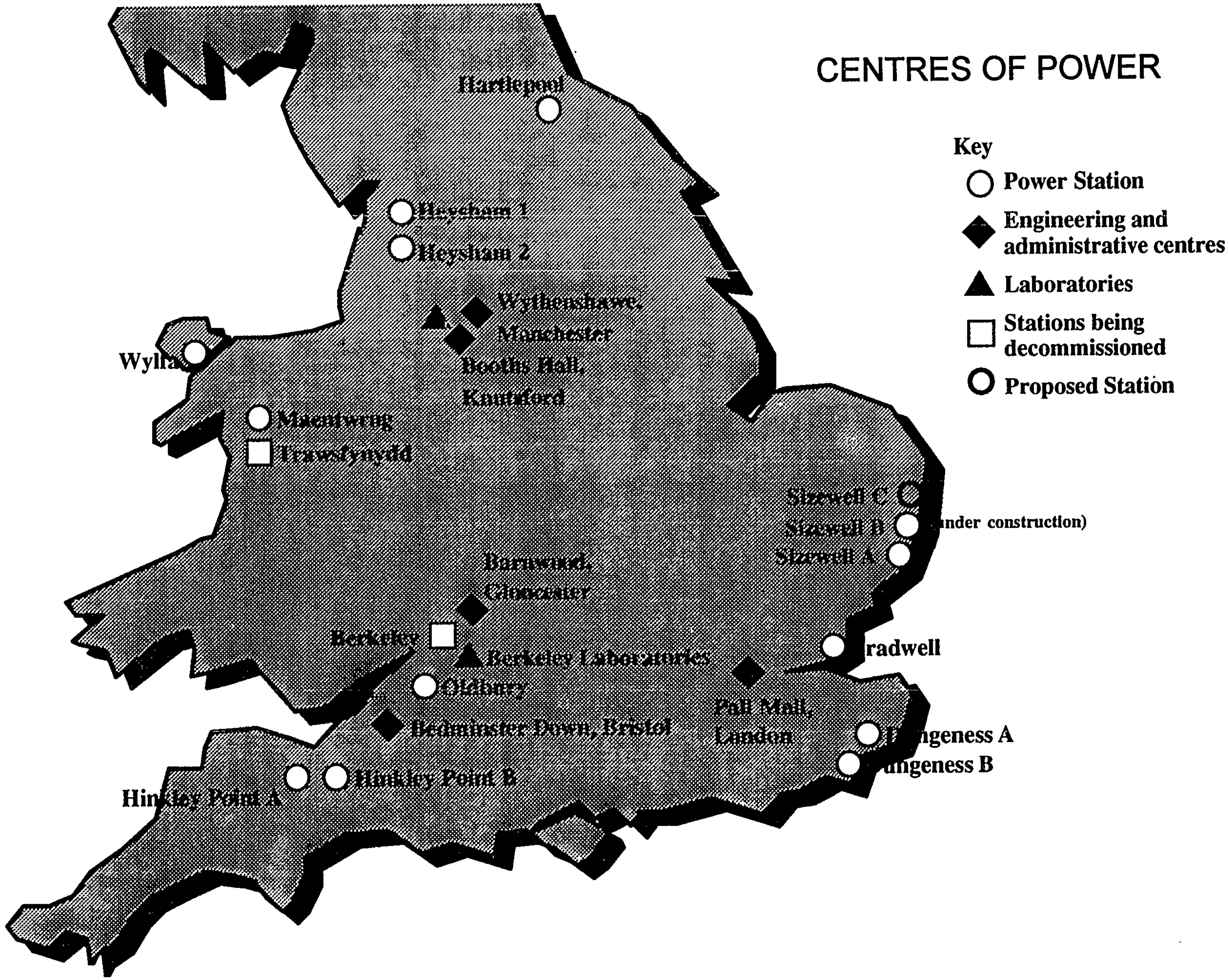
CENTRES OF POWER



SIZEWELL B DOSES

	TARGET
• Collective	240 man-Rem/yr
• Individual	1 Rem/yr

SIZEWELL B IMPROVEMENTS FOR OPERATOR PROTECTION

Approach:

UTILISE OPERATIONAL EXPERIENCE

Resulting in:

1 Minimise radiation sources and source strength
2 System and Plant layout improvements
3 Use of remote equipment

SIZEWELL B DOSES

	TARGET	ACTUAL
• Collective	240 man-Rem/yr	<197 man-Rem/yr
• Individual	1 Rem/yr	0.85 Rem/yr

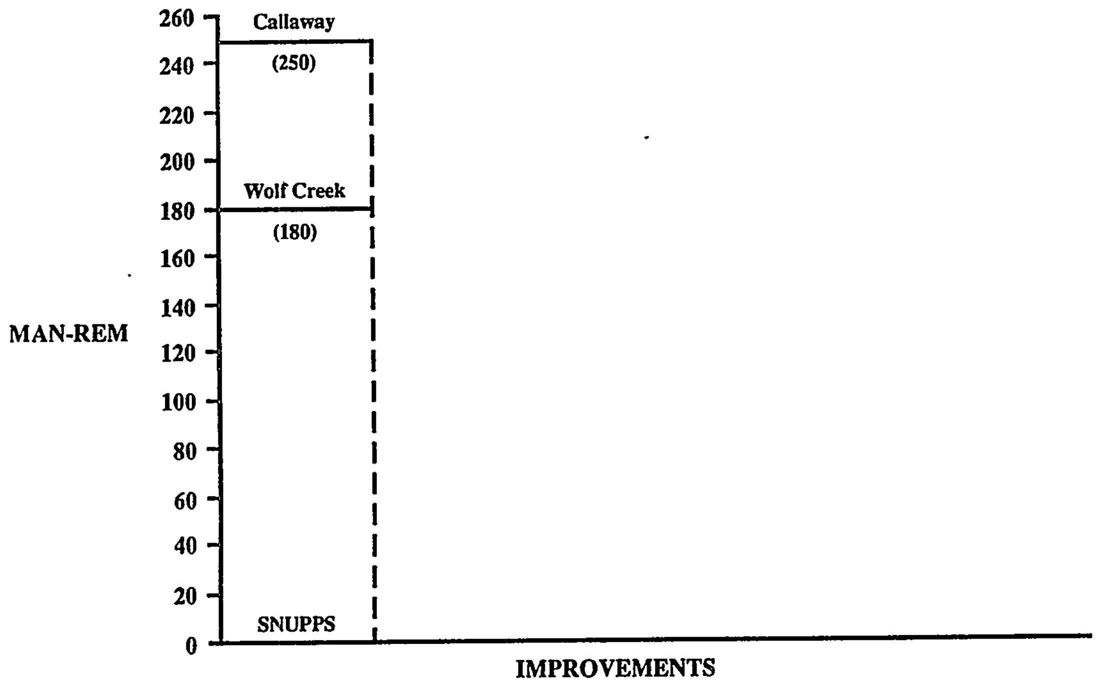
IMPROVEMENTS IN FUTURE PLANT

Future Plant a replica of Sizewell B

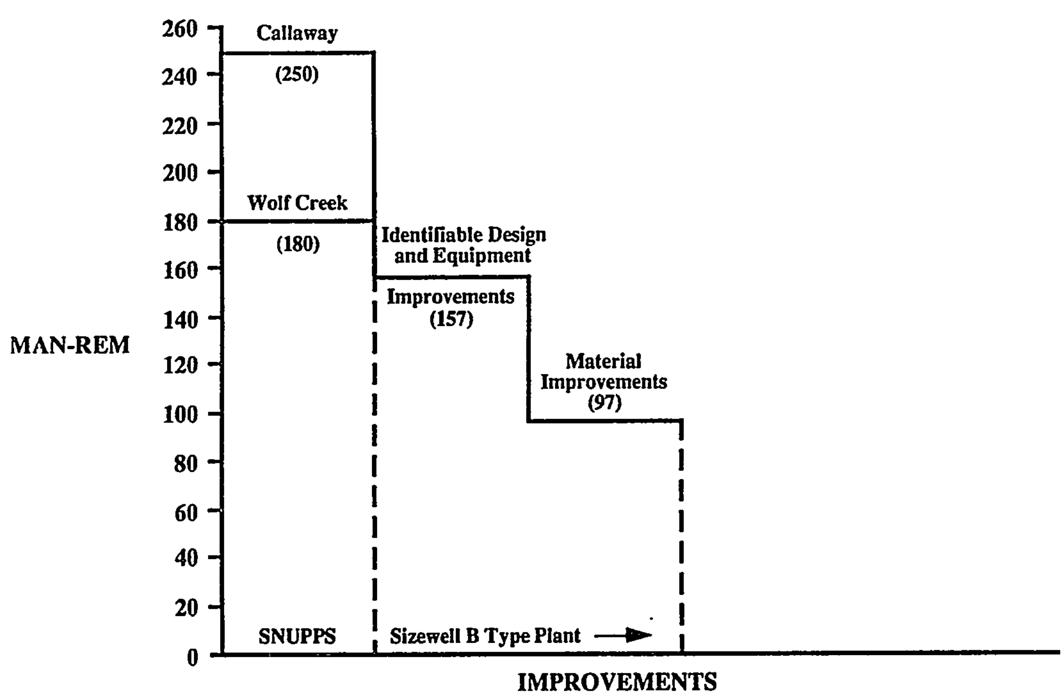
Improvements:

- 18 month fuel cycles
- Zircalloy fuel grids
- High pH chemistry
- Stellite removal

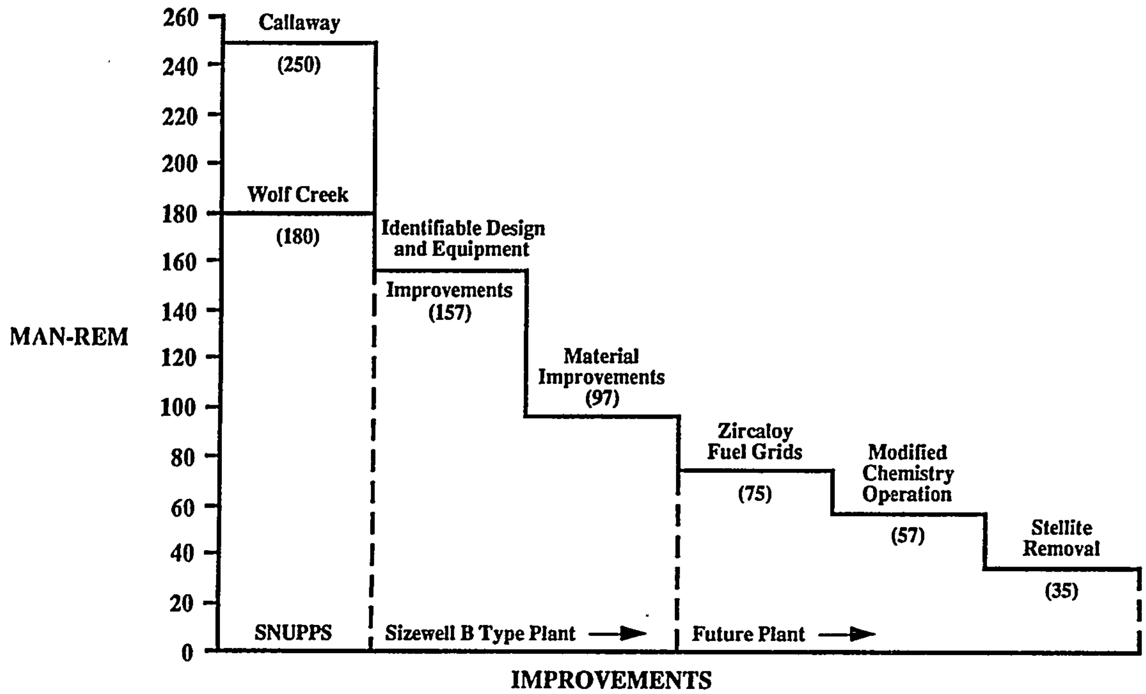
BASE OPERATING DOSE DATA



SIZEWELL B IMPROVEMENTS



FUTURE PLANT IMPROVEMENTS



CONCLUSIONS - FUTURE PWR DOSES

- Significant dose reductions to be achieved
- For SXB type plant with
  - 18 month fuel cycles
  - Zircalloy fuel grids
  - Higher pH chemistry
  - No stellite

  } 30-40 man Rem/y
- German experience: < 20 man Rem/y

IMPLEMENTATION OF ALARA AT THE DESIGN STAGE OF NUCLEAR POWER PLANTS

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INTRODUCTION

In the 1970s, Electricité de France (EdF) had limited knowledge and experience of pressurized water reactors (PWRs). Electricity generation by nuclear units was oriented towards gas-graphite reactors, even though EdF had a share in the PWR unit of CHOOZ A-1 (250 MWe, later upgraded to 320 MWe). Some facts about the origin of doses in that kind of reactor were known to the research and development (R&D) support staff of EdF, which mainly comprises the French Atomic Commission (CEA), but only a few of EdF's engineers were aware of these facts. One has to bear in mind that CHOOZ A-1 only went critical in April 1967 and was officially connected to the grid in May 1970 after some important problems had been solved. Meanwhile, the nuclear program was launched at full speed, beginning with the order for FESSENHEIM 1 in 1970, FESSENHEIM 2 and BUGEY 2 and 3 in 1971. TIHANGE 1, in which EdF had a share, went on-line in September 1975.

Also, supposing that EdF had had such knowledge and experience, it is quite evident that it would have been very difficult to modify the lay-out inside the reactor building.

Thus, in 1977, looking at results from the United States, an annual dose, D, of some 6 Sv was estimated for operating and maintaining a 900 MWe standard PWR unit.

The first signs of EdF worrying about the main source of exposure in a PWR, e.g., corrosion products (CP), can be traced back to 1975: technical studies were carried out using the CEA software "PACTOLE" to try and predict the influence of such parameters as cobalt and nickel input, primary water chemistry, on the deposition of activated corrosion products. These studies were finalized in the early 1980s, with the following conclusions:[1]

- with the available design, the cobalt input was dominated by (in decreasing order of priority) (a) the cobalt content of the steam generator's alloy, (b) the cobalt content of in-vessel materials, particularly Alloy 718 of the fuel grids, (c) the cobalt content of CVCS parts downstream filters and ion exchangers, and (d) hard-facing materials with high cobalt (stellite™) during normal wear and corrosion (and, on the contrary, a possible high impact on Co^{60} build-up in abnormal situations)[1]

- pHt (300°C) should be strictly controlled and be kept at a constant value of 7.0* (as much as possible, given the specified limits for Li concentration) and cold shutdown operations should be carefully managed to prevent CP redeposition, and a too high a residual activity in the water; this was applied as early as mid-1981[2]

- it was interesting to study the impact of load follow operation mode, the effect of "efficient" purification processes (high temperature, high flowrate), the effect of passivation, surface roughness, and decontamination (soft and hard chemicals, both singly or combined with ultra-sonic devices)

*In Reference 2, a pH of 6.9 is quoted: this is because the use of dissociation coefficients for water different from international practice.

From this, it is clear that the underlying assumption was "...a reduction of sources will lead to a reduction of dose rates and thus to a reduction of doses."

The aim of this presentation is to show that even though we did obtain, and still have some control on sources and dose rates, it has not been sufficient to satisfactorily control the doses. We will try and explain why, and what countermeasures are being applied to the existing plants.

Although it is not yet decided whether to build nuclear units with a new design ("PWR 2000"), we will describe how we think ALARA should be applied at the design stage.

CONTROL OF SOURCES AND DOSE RATES

Control of Sources

EdF has applied a methodology to control sources which includes the traditional three steps of the ALARA approach: prediction of parameter values, measurement of actual values, and analysis of actual discrepancies versus measured values. This has been described in Reference 3 and is only briefly summarized here:

- *prediction* of the active deposition of corrosion products (mainly $GBq.m^2$ of Co^{58} and Co^{60}) with the "PACTOLE" software:[1] although this software is not, and was not believed to be perfect, it has a phenomenological basis, thus introducing some logic in the results. Several of the results are "tied" together (e.g., production of isotopes is tied by the release of the base materials, a value that is calculated) and it is not possible to adjust one given value to the actual measurement (say, Co^{58} activity) without introducing a discrepancy in another (say, Co^{60}, Mn^{54}, or Fe^{59}).

- *direct measurement* of the values of sources through the walls of circuits and components using an *in situ* quantitative gamma spectrometry device (EMECC, described many times, and lately in Reference 4). This kind of technique, which is frequently applied by various utilities or companies, has the advantage on dose rate measurement to get at the origin (dose rate can be a misleading parameter because it integrates sources, lay-out, and the components' characteristics), and to bypass the numerous problems related to build-up factors.

- *analysis* of the differences between actual and predicted values is carried out using various softwares, including EdF's "TIGRE-RP" data base.[4] "PACTOLE" also is very useful because it allows sensibility studies in which tentative hypotheses to explain discrepancies can be tested.

Figures 1 and 2 show the contamination by Co^{60} of primary loops and steam generator (SG) tubes (which is nearly equal to the total out-of-vessel activity) for a few units versus cycle #. The predicted maximum values, referred to in EdF in 1983 as the source term for corrosion products, are indicated. The margins between the envelope and the actual evolution for given plants are explained (not all...) in terms of:

- lower input of cobalt: content of SG tubes (see Figure 3, impact of SG tubes with ~130 ppm Co^{59} at DAMPIERRE 1 instead of 450 ppm), introduction of zircaloy fuel grids,

- higher input of cobalt: abnormal wear of stellites™, high cobalt content of nickel plating of inconel grids,

- effect of surface roughness of SG tubes affecting the distribution of deposits (between SG and loops and channel heads areas),

- effect of primary chemistry (principally for a few units that had their first cycle in 1981 and 1982).

Figure 1

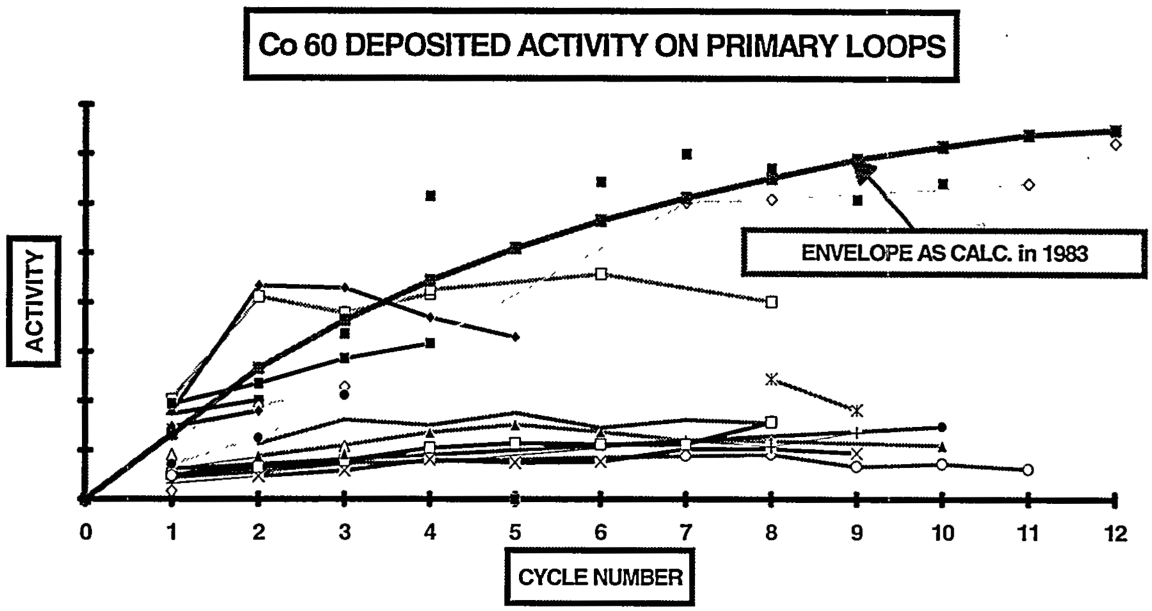


Figure 2

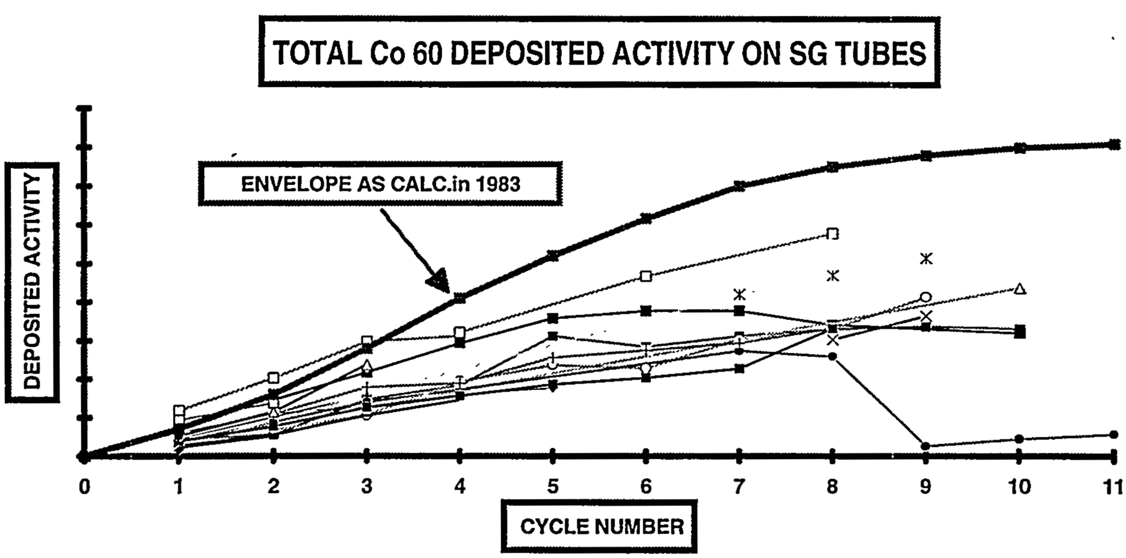


Figure 3

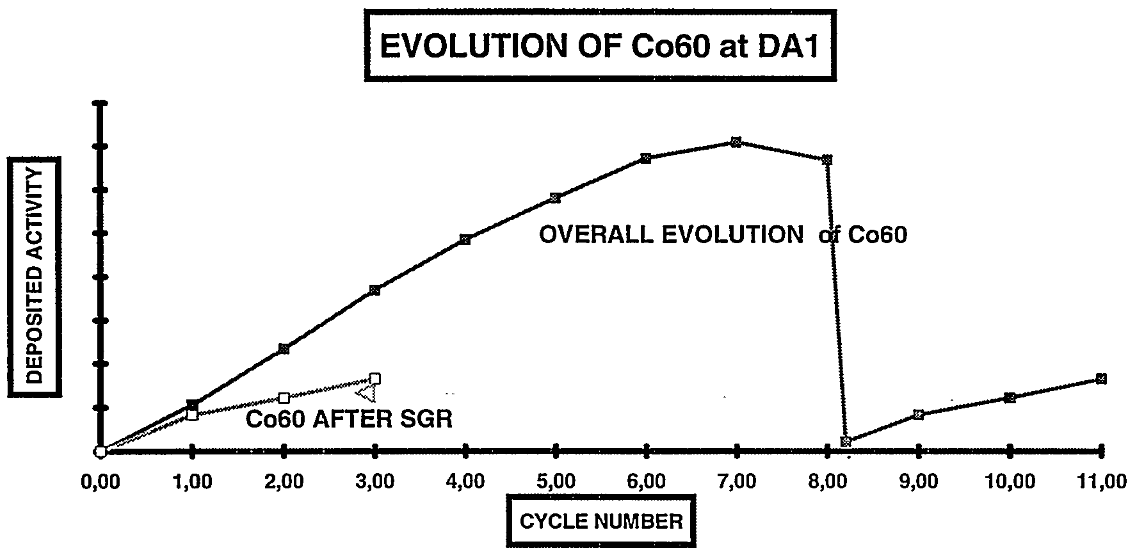
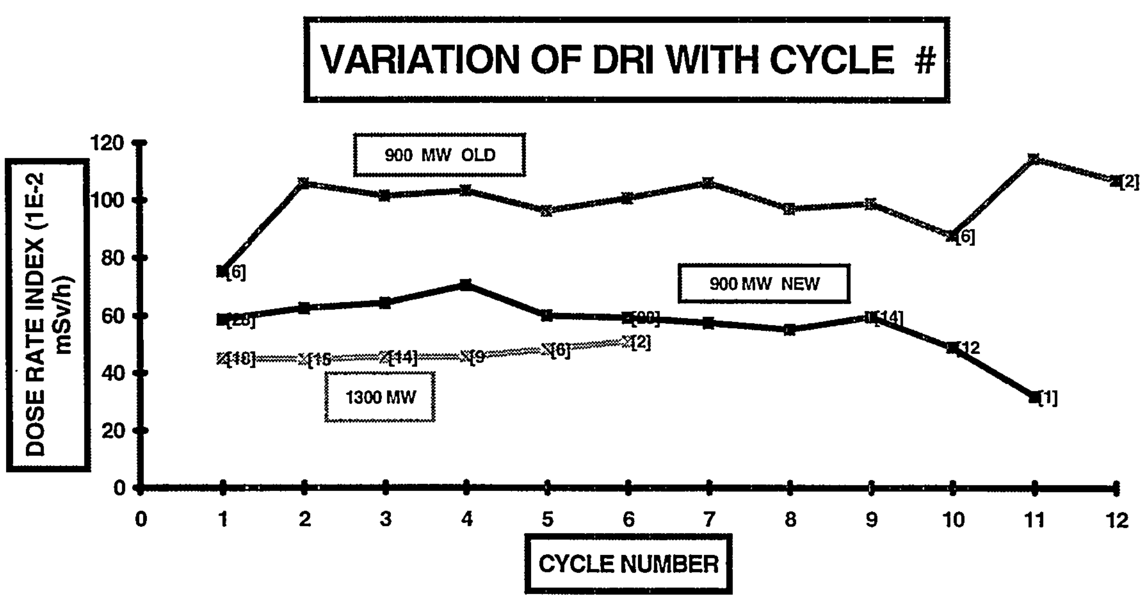


Figure 4



In fact, in most cases, the observed margin has, for its origin, a combination of the above causes (see EPRI seminars on radiation field control and BNES conferences).

Control of Dose Rates

Dose rates (DRs) are measured on all plants, at all EOC, within the frame of the "Standard Radiation Monitoring Program," which is quite similar to that of the EPRI. From these measurements, a dose rate index (DRI) is calculated. There is a good correlation between a function of Co^{58} and Co^{60} deposited activity on the loops and DRI.[3]

Figures 4 and 5 show the evolution of DRI versus cycle number and unit type.

The DRI and average DR (per unit) at the center point of the SG channel head is presented in Figure 6.

The figure shows that the average values of DRIs are moderately high for the oldest 900 MW units, and lower for the more recent 900 MW units. The DRIs for 1300 MW units are quite low. A rapid stabilization of DRIs is seen, that can be explained (at least in part) by the coordinated chemistry applied to the primary water.

EVOLUTION OF DOSES

Presentation of Results

On the basis of the standard indicator of collective dose/year/unit, Figure 7 shows that the French results compared favorably with those of other countries (with a nuclear capacity of more than 10 GWe) until 1988 with 1.77 man-Sv. Afterwards, and up to 1991 (2.44 man-Sv), the trend was an increase, whereas for other countries a decrease was observed. The increasing trend in the French results has been reversed lately with 2.36 and 2.04 man-Sv in 1992 and 1993, respectively.

Explanation of the Observed Dose Trends in France

DRI is representative of the primary circuit. The studies currently carried out in EdF do not reveal a very strong link between the activity and dose rates of primary and auxiliary systems. There are several (tentative) explanations for that, such as a higher possibility of contamination during the cooling down operation for auxiliary circuits than for primary circuit, or preferential deposition of Ag^{110M} (when present) on the cold parts of auxiliary systems.

However, as a first step, one can assume that low DRIs correspond to low DRs on auxiliary systems. The collective dose for maintenance (~85% of the annual dose) can roughly be broken in two equal parts for primary and auxiliary circuits. One should expect some kind of correlation between DRIs and doses. It can be seen from Figure 8 that the ratio of doses to DRIs largely increases with years in France. Because the DRIs are mainly stable (see section on Control of Dose Rates), the increasing parameter is the volume of work. It is mainly explained by:

- the extended visits that are mandatory at the end of cycle 1, 5, 10 etc. on a 900 MW. Experience shows that these visits require 50 to 100% more dose than for a standard visit,

- EdF's unit have suffered from the "Inconel 600 syndrome," the latest (last?) episode being cracks in the penetrations of the head of the vessel,

Figure 5

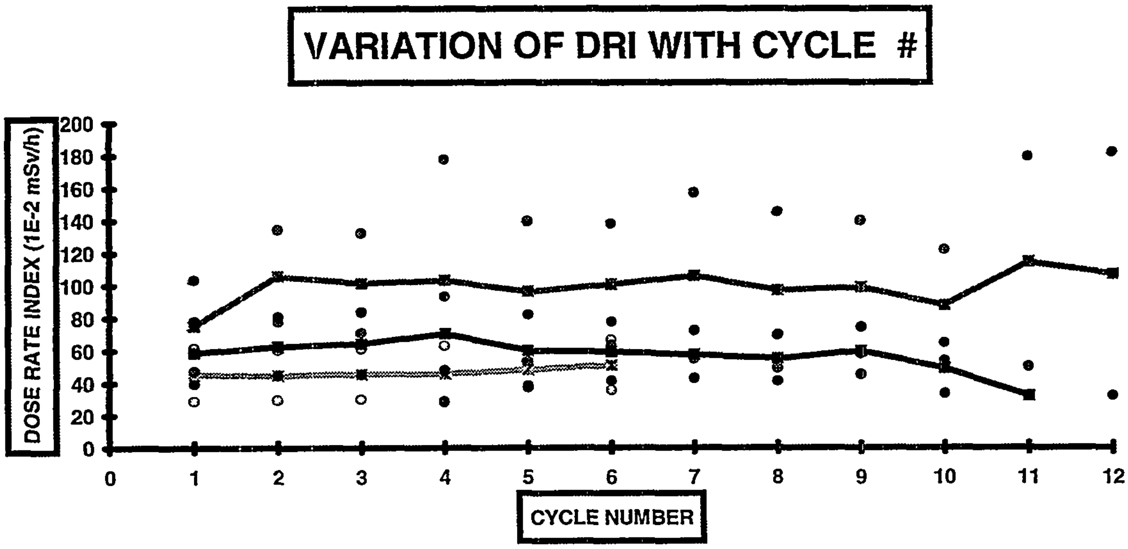


Figure 6

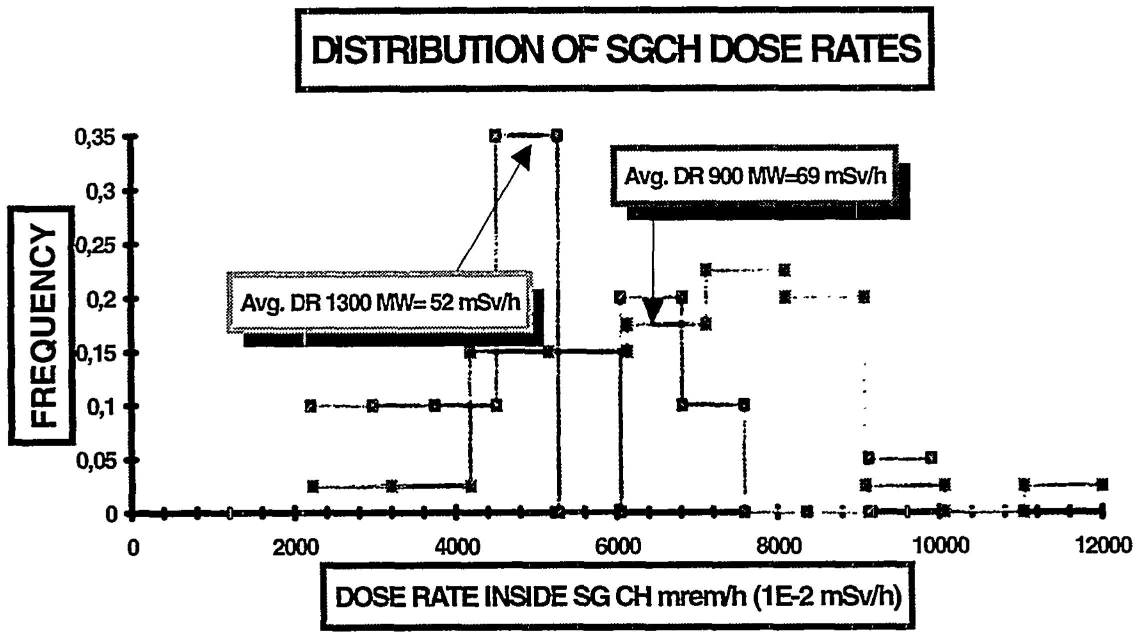


Figure 7

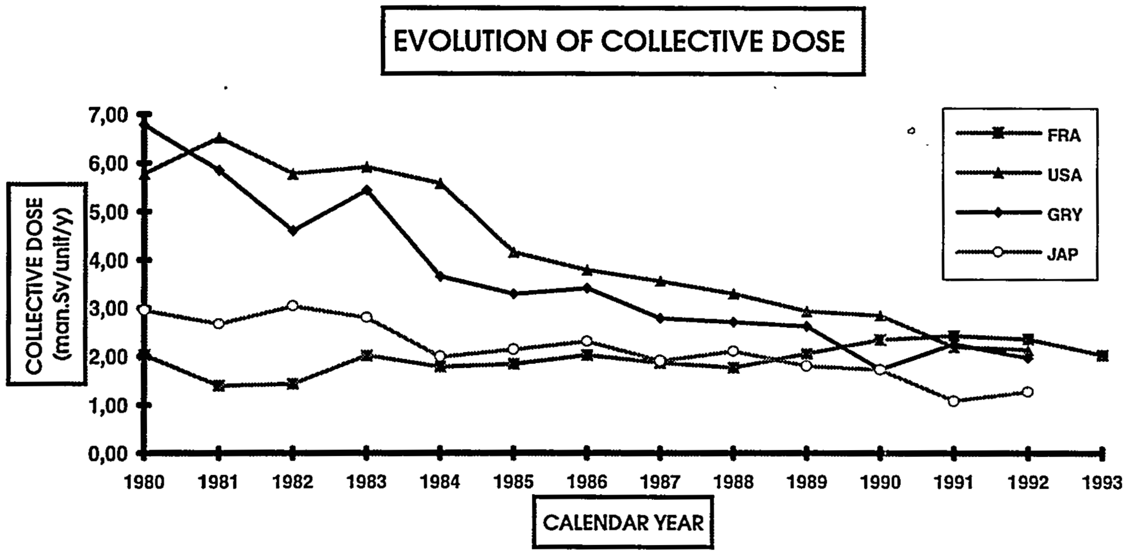
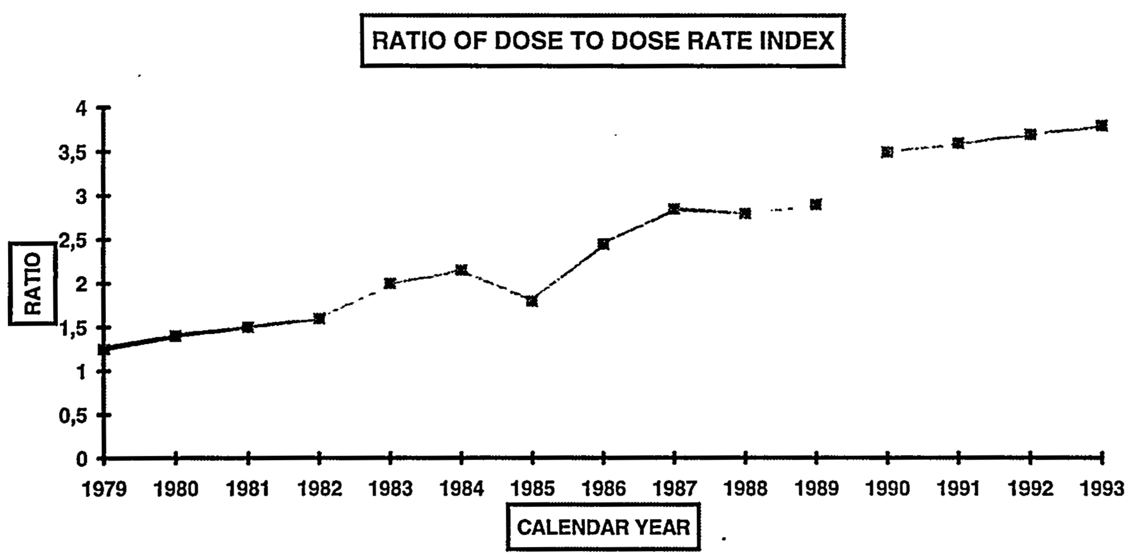


Figure 8



- units are operated presently with a cycle duration of approximately one year (which usually is recognized as penalizing for the annual dose indicator value) and the move to long cycles will take many years,

- a perfectible organization of maintenance jobs.

The decreasing trend observed these last two years is attributed to the ALARA policy that EdF decided to apply in 1990. Although one should not be too optimistic, it is noteworthy that this decrease is not an artifact of the indicator: the number of shutdowns for refueling and maintenance has been approximately the same (~45) through these years. Moreover, the dose for inspecting and repairing the vessel heads was 9% and 6% (in 1992 and 1993, respectively) of the total dose for maintenance.

A value of 1.6 man-Sv has been discussed as the goal for 1995.

INTEGRATING THE ALARA APPROACH TO NEW DESIGNS

Clearly, a new design should integrate more deeply than in the past the radiation protection *component* (rather than *constraint*). Should the radiation protection component not be taken into account in designing a nuclear power plant (NPP), one could ask in what kind of installation or circumstances it should be dealt with.

Many techniques to *minimize* the sources in PWRs (and other reactor types) are well established and agreed upon by the international community. Moreover, a proof that sources can be reduced to a very low level is given by the results of the German "konvoï" units for which an ALAP policy was applied. Indeed, the search for other or more efficient techniques is to be continued. Because these techniques are reviewed in many papers that are easily available, this paper will not add to the discussion. Rather, the aim of this paper is to emphasize that:

- analyzing EdF's operating units results in terms of doses, we learned that *keeping sources and dose rates at a moderate or low level was not sufficient to control doses*,

- by applying a complete ALARA methodology to the steam generator replacement (SGR) operation, we confirmed that *to control the dose* (including the associated costs), *it was necessary to cope with all the components of the dose.*

Lessons Learned from Replacing the Steam Generators

The steam generator replacement (SGR) operation at DAMPIERRE 1 in 1990 was, in EdF, the first maintenance operation carried out applying a ALARA methodology. Prediction of sources, dose rates (400 locations!), duration, and number of workers for any so-called "elementary task" was assessed, in which the whole operation was broken down. Several protection dispositions were compared for their impact on doses, costs, man-hours (shielding, decontamination, remote handling of tools, and automatization of tasks). Even though everything was not perfect, the results, in terms of dose, were sufficiently encouraging (2.13 man-Sv) to promote the extension of this approach to other SGRs: BUGEY 5 (BGY5) operation was performed in 1993 with a dose of 1.54 man-Sv, and GRAVELINES 1 (GRA1) was terminated in mid-April 1994 with a dose of 1.36 man-Sv (although the DRs were higher by 35% at GRA1 than at BGY5, and less protection was used; 50t instead of 70t at BGY5).

Dose for SGR (man-Sv):

U.S exp.	OBRIGHEIM	RINGHALS 2	DAMP. 1	DOEL 3	BUGEY 5	GRAV. 1
≥ 4.0 *	~ 7.0 *	2.9	2.13	1.96	1.54	1.36 **

*

With reference to N.E.I. April, 1993 issue; 2.4 at NORTH ANNA 1 from another reference.

**

Recent information to be confirmed.

A schematic representation of the formation of doses in a PWR is given in Figure 9. It is used inside EdF to keep in mind *the multiparameter aspect of the dose*. One cannot expect to keep doses ALARA by acting upon a single parameter, even though some might be more influential than others.

Moreover, the dose is described as having a three-layered constitution, D1 + D2 + D3, where:

- <u>D1 is quite impossible to reduce</u> considering a given state of the design and knowledge,
- <u>D2 can be reduced</u> (with increasing efforts when D2 + D1 tends to equal D1)
- <u>D3 that can be avoided</u> rather than reduced.

These parameters can be combined with the dose reduction vs. cost graph of CEPN (Figure 10). Although ALARA methodology can be applied to reduce any of the components D1, D2, and D3, the knowledge and tools required generally will not be identical.

Roughly speaking, pragmatism, common sense, adherence of all involved to the goals, basic knowledge, and adequate organization will be necessary in any case, but probably sufficient for avoiding D3. By contrast, an increasing level of sophistication in the knowledge and tools will be necessary to succeed in reducing D2 and D3.

Current Status of Dose Reduction in EdF

With this breakdown of the total dose, and its application to *operating units*, to "exceptional maintenance" such as *SGR*, and to *new projects*, the current status of dose reduction is the following:

- avoiding D3 is in progress for the *operating units*: if all units could achieve the results of some "leading units," the average dose would be ~1.5 man-Sv for a 900 MW unit and ~1 man-Sv for 1300 MW units. Therefore, the dose savings corresponding to avoiding D3 can be estimated to 30% of the 1993 value (0.4 out of 2. man-Sv).

A major part of D3 can be estimated as having been avoided for the *SGR operations*. Drawbacks always are to be expected, but feed-back experience is building-up; this operation will never be of the routine kind, but all efforts are made to standardize it.

At the design stage of *new projects*, the lay-out should be carefully studied to make sure that working conditions for maintenance and operation are taken into account: irreducible jobs should not generate avoidable dose (D3).

Figure 9

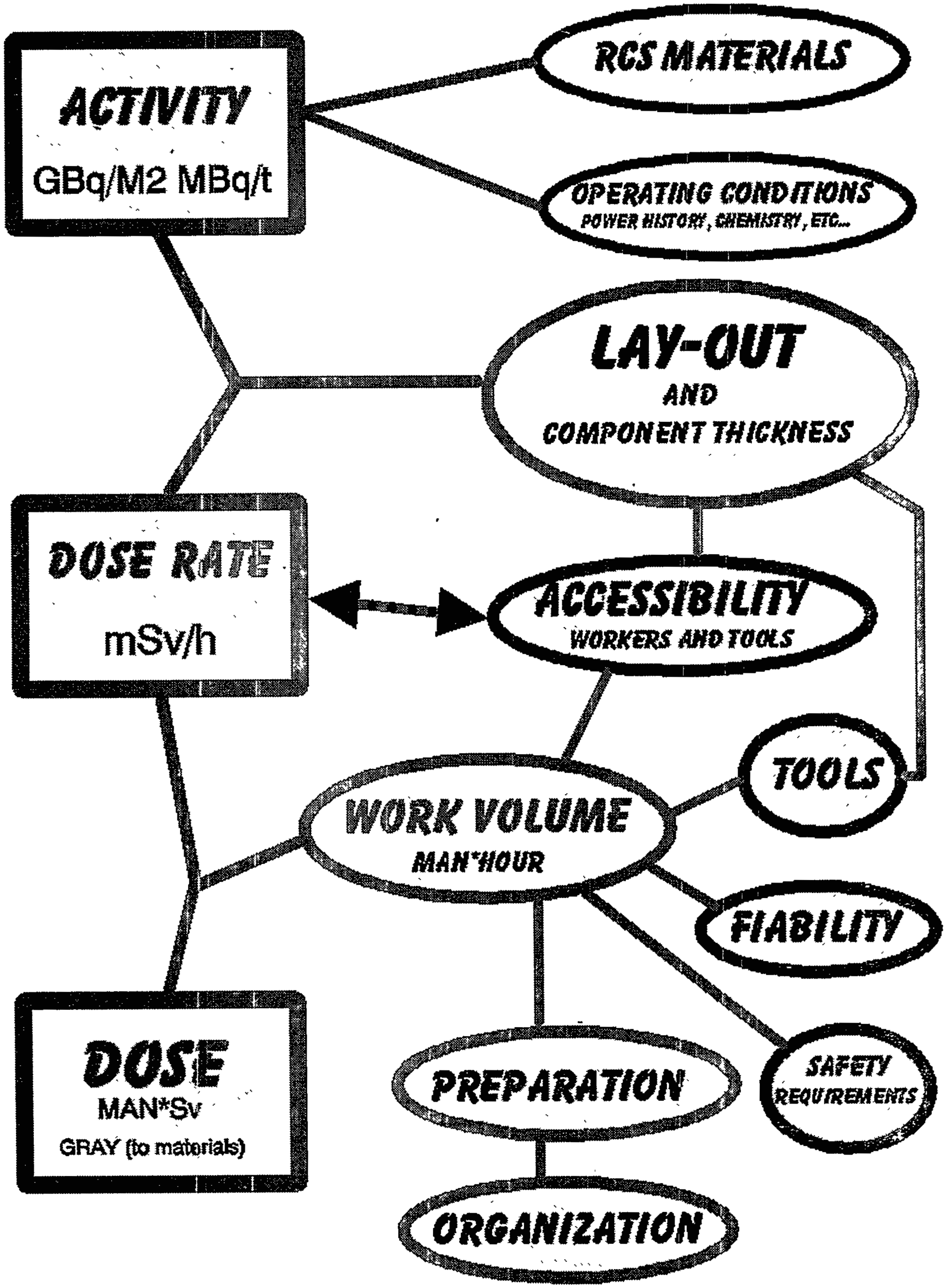
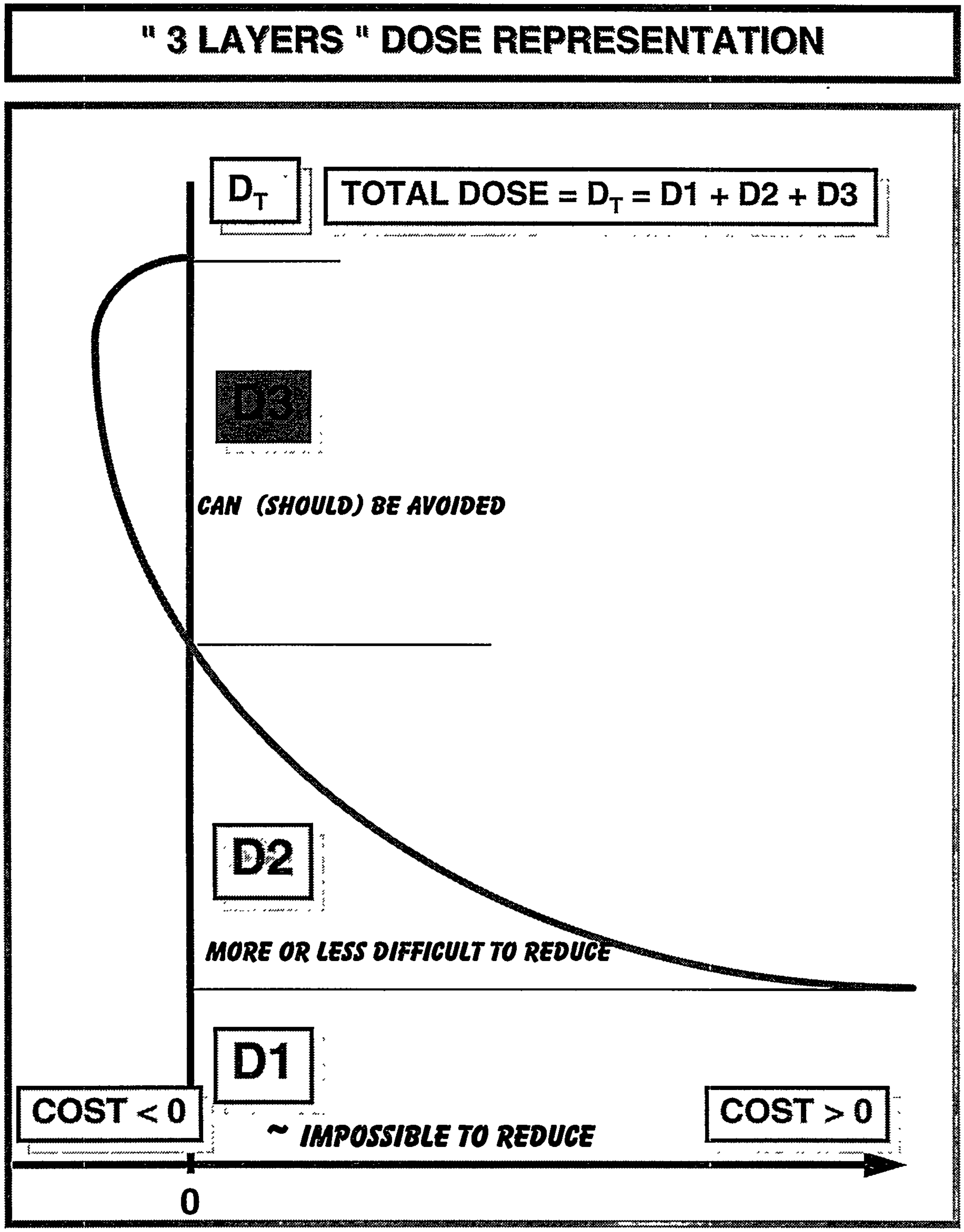

ACTIVITY
GBq/M2 MBq/t
RCS MATERIALS
OPERATING CONDITIONS
POWER HISTORY, CHEMISTRY, ETC...
LAY-OUT
AND
COMPONENT THICKNESS
DOSE RATE
mSv/h
ACCESSIBILITY
WORKERS AND TOOLS
TOOLS
WORK VOLUME
MAN*HOUR
FIABILITY
DOSE
MAN*Sv
GRAY (to materials)
PREPARATION
SAFETY
REQUIREMENTS
ORGANIZATION

Figure 10


" 3 LAYERS " DOSE REPRESENTATION
D_T
TOTAL DOSE = D_T = D1 + D2 + D3
D3
CAN (SHOULD) BE AVOIDED
D2
MORE OR LESS DIFFICULT TO REDUCE
D1
COST < 0
~ IMPOSSIBLE TO REDUCE
COST > 0
0

- reducing D2 is the field for optimization (for *new* or *operating units* and *SGRs*): further reduction of doses will require the reduction of sources, and/or dose rates, and/or man-hours that can be achieved but are not costless, or that interfere with safety or other important issues. Thus, cost-benefit analyses will have to be performed. For example:

- reducing dose rates without reducing sources will require additional shielding, either movable or permanent. With an available area fixed by the lay-out, it will be necessary to balance the decrease of dose rates against the possible increase in man-hours (less available space/time to set up the shielding).

- reducing sources for a *new unit* generally will be easier than for an *operating one*: the possibility of backfitting is, unfortunately, limited when extended modifications are necessary (one may recognize, as SIEMENS does, that a major source of cobalt 60 is the in-the-vessel hard facing material with high cobalt 59 -- except CRDMs -- and still not be able to replace these parts on "old" units). This is probably a good reason for the intensive search for "non-intrusive" means to reduce sources, such as pH control, and addition of additives to the coolant (for example, zinc for PWRs).

- to reduce man-hours (after D3 has been avoided) principally for *operating units*, one can try to demonstrate that a particular inspection has little value, or that a limiting value (for example, acceptable leakage flowrate) can be relaxed. Generally, it will be difficult to make a convincing demonstration if safety issues are part of the problem.

- reducing D1 is the field of R&D and *new plants* (sometimes with a possible application to operating units).

For designing *new units*, efforts will be made to keep the total dose (D1 + D2 + D3) ALARA. EdF adopted an approach and an organization whose acronym is "CIDEM": design and lay-out, taking into account reliability, experience, maintenance requirements, and radiation protection. A dose objective was set with a value of $D = 0.5$ man-Sv y^{-1} and an uncertainty margin of 50% ($D < 0.75$) at the present stage of the project. It is an average for the whole life of the unit for normal operation and maintenance. This value may appear too high for a unit to be connected to the grid in 2005: it is thought that it can be obtained at a reasonable cost (ALARA is to be applied rather than ALAP). This overall dose is broken down into "dose credits" for the tasks to be performed for operation and maintenance, such as the repartition used in the ISOE system, but with more details.

Tools Useful to Apply the ALARA Methodology

Adequate management and organization of a project in which the ALARA methodology is applied are necessary, and have been described in several papers (OECD, CEPN, NRPB). Adapted tools also are useful; some of them currently used in France and EdF are:

- provision of sources: CEA "PACTOLE" and "PROFIP" softwares for corrosion and fission products respectively

- provision of dose rates: "MICROSHIELD" (GROVE ENGINEERING, Inc.), "MERCURE" (CEA), "PANTHERE-RP V.0" (EdF-SEPTEN)[6]

- provision of work volume: EdF's data bases,

- provision of doses: "TIGRE-RP", "DOSINAT", data bases (EDF), "FRADOSE" data base (FRAMATOME)

- dose management: "DOSI-ANA" (CEPN, FRAMATOME, EdF)[5]

These softwares are sophisticated, but nevertheless, generally user-friendly. Efforts are continuing in that direction: PANTHERE-RP V1.0 will be available in EdF's design departments at mid-1995. A more complete package (version V2.0) comprising the integrated data base (possibly with an expert system) and computer mock-ups of the main parts of the units are currently developed.

The use of sophisticated tools is required for the accurate provisions that have to be made when trying to reduce D2.

CONCLUSION

EdF experienced the fact that what apparently was a rather satisfying control of sources and dose rates was not sufficient to control the dose for operation and maintenance of its PWR units. Even though sources and dose rates have been, and still are, maintained at a moderate to low level, after 10 years ending in 1988 with excellent results in terms of annual doses per unit or produced energy, an increasing trend was observed up to 1991. Analyses showed that this resulted mainly from a large increase in work volume for maintenance.

Meanwhile, the successful management of doses for the SGR operations (DAM1, BGY5, GRA1 units) demonstrated the potentialities of the ALARA approach. Thus, ALARA organizations have been created at different levels (plant sites, central management of all NPPs) and attempts have been made to apply this approach to normal operation and maintenance of the units.

The reverse of the increasing trend in 1992 and 1993, despite the problem of penetration cracks in the vessel head that added 6 to 9% to the normal dose, is the result of the extended ALARA approach.

SGRs operation at BUGEY 5 (1.54 man-Sv) and GRAVELINES 1 (1.36 man-Sv) show that an ALARA approach ensures some reproducibility in the results (although drawbacks should always be expected).

The total dose in the next fifteen years or so for SGRs at EdF's units (supposing that 31,900 MW units will have to ensure this) can be estimated to lie in the range 30 to 45 man-Sv. This is far less than the present annual dose for all EdF's units which is about 100 man-Sv.

Thus, a decision was taken at the end of 1993 to apply the ALARA approach to normal maintenance operations more strictly (known in EdF as the ALARA Project), as well as to new projects (e.g., units with a new design, auxiliary facilities to NPPs).

A goal of 1.6 man-Sv/unit/year (averaged on all units) in 1995 is under discussion.

For a new design, the dose objective, averaged over the lifetime of the plant, has been set at 0.5 man-Sv (with an uncertainty of 50%). Although lower or equal values already have been obtained for a few units (inside and outside France), this goal will be maintained to take into account the economical parameters (ALARA approach, as opposed to ALAP).

The methodology is a classical one, and corresponds to EdF's CIDEM organization of the project. Many tools are available to help to implement the prediction of the dose parameters (sources, dose rates, work volume), taking into account varied options in the design and the lay-out.

Finally, we point out that we believe that setting a dose objective implies a rational approach, and this is probably as important as the value of the goal itself. The radiation protection component has a pronounced transverse turn that should be taken into account. When an ALARA methodology is fully applied, there is confidence that radiation protection aspects are part of the decision process.

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Authors' Biographies

Alain Brissaud is the Head of the Radiation Protection Group of EDF's Design Department for Thermal and Nuclear Projects (SEPTEN). The group is responsible for research and development in radiation protection applied to nuclear power plants, the development of methodologies and appropriate softwares to carry out simulation of radiation protection problems. Alain Brissaud recived his engineer diploma in Nuclear Physics in 1977.

Phillipe Ridoux is a senior engineer in the Radiation Protection Group of EDF's Design Department of thermal and Nuclear Projects. He received his engineer diploma in 1980 with a specialization in atomic engineering.

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PAPER 3-7
DISCUSSION

Andersen: Are you discovering any actual facility design modifications that look cost effective for existing facilities out of this process, or is it mostly accomplished through other means?

Brissaud: Yes, but I think it is not a very new dicovery. The idea is when one is designing a plant, also one should design the maintenance. Think about the people that are going to work, because some maintenance will have to be carried out. It's better to think about it beforehand, not afterwards. Not so many new things, nothing revolutionary.

SESSION 3
DISCUSSION

Co-Chair: (Bari) As Frank Congel mentioned earlier, the seventh paper in this session will not be given today but later on in this conference. The author has not arrived at the meeting site yet, so stay tuned, it will be presented on a later date. This concludes the formal presentation in this session, and we now turn to that portion of the session in which you, the conference participants, get to query the authors on topics that go beyond the specifics of their papers. The only constraint here is that you stay as close as reasonably achievable to the topic of the session, the intersection of ALARA at Advanced Reactors. I would now like to ask the six authors to join us at the front table and we will open the panel discussion.

Khan: I have a question for Charlie Hinson. You saw that between Sizewell B and Sizewell C there has been potentially a very drastic reduction in estimated plant dose. Now, the advanced reactor designs in the U.S., I understand, are going to be standard designs. If we see any possibilities of reducing doses very drastically by some techniques, how much provision is there to incorporate that into the design after the design has been frozen?

Hinson: As I stated in my paper, one of the objectives of the current review process being used to review the "next-generation" reactor designs (in accordance with the new Standardization Rule) is to end up with a standard nuclear power plant final design which can then be incorporated into individual facility license applications. Once a design for a "next-generation" plant has been approved by the NRC (through the issuance of a Final Design Approval (FDA)), it may be several years before a utility decides to purchase one of these plants and incorporate the design into their license application. In the interim period between the issuance of the FDA and the initiation of actual plant construction, technology may have improved so much that some of the plant components may be outdated or obsolete. The FDA allows for such changes to be made to the plant design based on improvements in technology. For example, the "next-generation" reactor designs state that the cobalt content of piping and other components in direct contact with the reactor coolant be restricted to 0.05 weight percent. Several years from now, it may be cost effective for utilities to manufacture reactor coolant piping which limits cobalt content to 0.02 weight percent or less. If this is the case, then the NRC will encourage the use of the lower cobalt material. For other areas, such as plant shielding, the specifications (materials, dimensions) for components used to store and process radioactive wasates are not currently provided in the plant design. These specifications will be determined at a later time by a licensee wishing to build one of these "next-generation" design plants. Once these specificaitons are made, then the shielding design can be finalized. To summarize, there is sufficient flexibility in thefinal plant designs approved by the NRC to allow for design modifications to reflect improvements in technology. Some of these modifications could result in lower plant doses.

Co-Chair: (Congel) Let me add a little bit to what Charlie had to say. First of all, there is the hearing that has to be held before the final design approval is specified with all of the requirements of the form of the ITAC, the inspection test acceptance analysis criteria. So there's still a couple of years during the hearing process. Secondly, with maybe just a few exceptions, final materials are not stated composition wise. What is really stated are requirements for specifications to be met in terms of strength, size, capacity, seismic resistance, things of that sort. So when it comes to actual materials specificity, there could be a continual learning curve and until a plant is actually ordered and there is all the steam supply system matched to the reactor design, only then are things truly finalized. So there is quite a bit of time left, and I have to emphasize that design approval does not go down to the depth of detail that specifies "this is the composition you have to have in material." That's not the case. Just the requirements it has to meet in terms of what I just said is what is truly specified.

Robinson: All the ALARA that we have been hearing about today has all been concentrated on occupational exposures during normal operations. I was just wondering if any of the people talking about the designs have thought about one of the Achilles's heels of the nuclear industry and that is on the decommissioning side. How have they thought about minimizing doses during that phase of work.

Hinson: The design of the "next-generation" reactors does incorporate features to facilitate decommissioning. Several of the design features that are incorporated into the plant design to minimize doses during operations, such as using modular components and shielding, etc., are the same features that would be used when decommissioning a plant to lower personnel doses. The use of modular components, simplification of design, plant shielding, improved plant accessibility, and the way the plant is designed so that a majority of the components can be removed from the plant without cutting the components into sections are some of the same features that would facilitate decommissioning of the facility.

Zodiates: I would like to add something else. The dose during the decommissioning comes from basically two sources. First is activation of components of the primary circuits in particular and the concrete around the primary vessel, and second, from contamination of the structures which need to be decommissioned. In terms of activation, this can be reduced by reducing certain impurities in the reactor pressure vessel (RPV) and primary circuit materials but there are limits how far that can go because the RPV is made of stainless steel. In terms of contamination, by reducing radiation sources which help operator doses, by reducing the spread of contamination, which again reduces operator doses, you achieve a reduction in decommissioning exposure. In addition, the provision of decontaminable surface finishes in many areas, which again will include operator doses during operation easily enable decontamination of the surface for final decommissioning. So, the operator dose during operation and the dose during decommissioning go very much hand in hand. By reducing one, you achieve reduction in the other.

Meinhold: I was just intrigued by the suggestion that we could improve things by people going to work. Turning that around, I just wondered if you thought about the latest ICRP recommendations which say that we should include all the radiations received while at work, meaning that the radon and the external radiation from the concrete that you brought in there to give all this protection is going to actually irradiate them even greater with the natural background. I just wondered if you thought about that in some of your advanced designs.

Zodiates: My comment was made with tongue in cheek, but there is a bit of truth in it. In terms of external radiation, we do account for it because our film badges do respond to all radiations in the environment. In terms of radon exposure, we are probably much safer in our stations because of all the heating, ventilation, and air conditioning (HVAC) we provide to keep airborne contamination low. I don't have numbers in terms of radon exposure, but I do know we have a lot of ventilation in our stations which would minimize the exposure to radon.

Haynes: Maybe I could just comment on that, too. I'm not really answering Charlie Meinhold's question, but certainly in our environment where we put a lot of emphasis on self-protection and a lot of emphasis on training, we go to fairly extreme lengths to at least put natural background radiation doses into perspective with occupational exposures, and there's no question in the newer plants we're getting to the point where the two are very comparable in any given year. It's important that people understand that so that we don't bend the ALARA equation too far in the wrong way.

Lee: I'm from the Korea Institute of Nuclear Safety. I have a question for Mr. Crom for System 80+. Do you have any definite figures or estimation for interlock system when you have the in-core monitoring was pulled was pulled out from Duke Power Engineering experience?

Crom: No, but the Duke Power experience is based on a different type of configuration than the Combustion Engineering (CE) design. What you are asking is what the actual dose in that particular in-core is? We have not gone through a detailed calculation of what it is for the System 80+ design to date. This will be done during the very detailed design through the design acceptance criteria.

Lee: I don't know whether I can give you this in figures, but you have technically no open items, but I know that you have nine confirmatory items. Does that include the in-core monitoring system or not?

Crom: No. The confirmatory items are mostly procedural in nature. It is basically incorporating agreed-upon commitments with NRC into the final SAR to insure that all the commitments are included. Resolving confirmatory items is basically just an editorial exercise of the final SAR plus an integratived review by the NRC of the SAR and ITAAC tier-one document. The NRC did a separate review between the tier-one and the SAR material to make sure that they matched up.

Westbrook: At Oak Ridge National Laboratory we are working on the advanced neutron source reactor design, which is kind of a new reactor to end all new reactors. Being a research reactor it doesn't necessarily have all the basis of experience that the nuclear power plants have. However, as a radiological person who is consulting to the design team, I have some problems with them based on how to come up with the dose estimates. They will ask me, "what is an acceptable dose rate for this area," and I will say, "you tell me what work is going to be done there, you tell me how many hours a year occupancy, and so on, and we will figure how much shielding we need to correspond with some reasonable dose rate." And they say "well we will have to figure out how many man-rem are going to go to maintenance of this, maintenance of that, normal operations, inspections, and so on, and they are trying to figure this out. But they keep coming back to "we can't figure out the dose until we know the dose rate." The shielders say, "I can't tell you how much shielding until you tell me how much dose rate you want on the other side." Then they also go back, "Well, what are the source terms?" See, we go around and around with this. The nearest equivalent reactor is the Institute Laue-Longevin in Granoble, France, but that, if memory serves, is something like 20 years old. So it has not had all these improvements that you are speaking of. When you guys are doing all these estimates and dose estimates, how do you handle this "dosi-do" problem of "you tell me first, then I will tell you." Do you always start out with some set amount of source terms? You can do that with the core, but then you have to process it in some way and say "we expect so much in the steam generator, we expect so much at our hot spots, we expect so much in the condenser..." How do you handle this problem of trying to get someone to make the first estimate and then somebody else refines it? How do you handle this?

Crom: First of all, as far as source term, that's pretty straight forward for current generation plants. We determine what the source is based on one-quarter percent failed fuel and then go through the various processing through filters and demins of the CVCS system and know basically what that dose of all the components is going to be. From the standpoint of shielding, we then had our people who have done dose analysis over and over again for current plants judge what they think the dose should be in the adjacent areas. We know what it will be in the cubicle from the equipment based on that source term analysis, now what do we want it to be in the adjacent area. What we have put into the SARs are radiation zone drawings of what we want the dose to be. We then developed design acceptance criteria to do the detailed shielding once piping is routed such that we then can do the shielding calculations. Then we have to maintain those within the limits we show in the radiation zone drawings in the SAR.

Lau: I'll speak to that briefly. My experience has been that we can establish desired access times and then it becomes somewhat of an iterative process to determine whether or not it is practical to meet those or whether or not you need to reallocate the space or redesign the equipment locations so that you can actually meet the shielding requirements without eating up all the space available for the shield. So it is somewhat iterative, but you should be able to establish at least a desired access time based on known maintenance, and in your case perhaps experiment loadings and things of that sort that have to take place in the reactor area.

Co-Chair: (Bari) Would anyone else like to comment on that? I have a question. A few of the authors compared their results with the EPRI limit. Can you tell me what the uncertainty is in your calculated estimate?

Crom: Maybe I'll take first crack at that because I asked Charlie why I didn't get credit for 30-40 man-rem because we thought we had the same improvements. We think we were conservative in our particular estimate. We think that if we do a detailed time and motion study with source terms per some of the NUREGs in the detailed design, we will have lower estimates.

Zodiates: In Sizewell-C, we carried out two dose assessments. One assessment was based on dose rate measurements from equivalent plants, and then we calculated the dose by defining what work needs to be done, how long it takes to do the job, how many people, etc. Starting from the dose rate measurements you have at least a 10% uncertainty in defining and measuring a dose rate. So my judgement is that our dose estimates are of that same order of uncertainty.

Hinson: For the most part, the dose assessments contained in the applications for the "next-generation" reactor designs do not contain the level of detail specified for a dose assessment in Regulatory Guide 8.19. this is because, at this stage of the plant design, the exact piping layout and the specifications (such as component size, shape, placement, and material composition) for the components containing the radioactive source terms are not known. Without this information, the exact amount of shielding used cannot be determined. In addition, the determination of the number of personnel working at the plant is the responsibility of the license applicant. Without accurate knowledge of plant shielding or the number of plant personnel, the plant designer cannot perform an adequate dose assessment using time/motion studies (as is recommended in Regulatory

Guide 8.19). Therefore, there is a possiblity that once this missing information is known, the plant collective dose estimates calculated by the individual plant licensee may be lower than the currently provided dose estimates.

Lau: I started to say that most of us are reluctant to hang a number on the degree of conservatism, or whatever you want to call it, that might be in those estimates, but at the same time it is going to vary based on our experience level. If it is a job that you have a lot of experience doing and you know exactly how things are going to be in regard to the dose rates and also in regard to personnel and their time to do the job, that particular estimate may be quite accurate. Again, I don't know if I want to put a number on that, but on the other hand, newer jobs that you do not have experience with are going to have less accurate estimates. When you add them all up you have a cumulative mixture of accuracy. We could all make a guess, and that's about as much as you can do at the moment.

Baum: I recall that during the design of the Sizewell B facility there were many cost-benefit analyses done on some of the major engineering modifications that were being considered, and I wonder if similar studies are being done on the other plants, and if so, what are the criteria being used to judge the cost-effectiveness and where you should draw the line. Maybe you can call \$/Sv values, or how is this being decided, how low one should drive the dose number.

Haynes: I'll just comment on that in a very general way. It's a tough question to answer and certainly one that we struggled with. At the Darlington plant, for example, we've come under some criticism, I would say, in terms of reducing dose rates too far. We've spent too much money. The nearest plant to it in terms of design is Bruce B, which started up in the mid-80s is a four-unit plant, currently operates four units typically for about 150 man-rem/yr. Application of the dose-reduction measures that we used on Darlington throughout the 1980s and using the process I described resulted in the initial dose estimates coming down by roughly a factor of three after the iterative process of applying various dose-reduction measures. And you really question whether that is good value for money in the end, given the other problems that we have to deal with, not only on the nuclear side, but on the other side, of our company's business. I don't know precisely the answer to your question, but it certainly is worth asking and raises the whole question of cost-effectiveness of dose reduction.

Lau: I might add to that for the AP600. We have performed some cost-benefit analyses basically by using the industry-accepted cost per man-rem that is in existence today, of around \$10,000-\$12,000 per man-rem. Using those numbers, we have generally been able to show whether a particular design or requirement such as the amount of cobalt impurity, or whatever it might be, has a practical limit. This methodology can be applied to more situations than we have evaluated so far, and I think it will be as we go through our final ORE estimates.

Co-Chair: (Bari) Would any other panelist like to tackle that?

Crom: I am pretty much in agreement. We have done some cost-benefit with similar type analysis, but again, when you start getting down in less than 100 man-rem, you start questioning your cost-benefit analysis. Most of the analysis we have done is for off-site, where severe accident doses is the concern. Major design features for severe accident analysis were evaluated on a cost-benefit basis.

Co-Chair: (Bari) Are there any other questions from the audience?

Ferguson: This is a question not only to the panel but to anyone in the audience who would like to answer. Listening to Mr. Tom Crom in his discussion on System 80+, and Mr. Fred Lau in his discussion on the AP600, it appears that their advanced light-water reactors for U.S. plants are addressing new source terms in terms of post-accident access requirements. Is the nuclear industry in Europe shifting toward a similar type of new source term approach, or what is the status over there for their advanced light-water reactors?

Lau: I will start off with what I can recall on that subject. With regard to using the new source term definitions in the NUREG, the existing plants have not started to use it, or at least are not headed in that direction at a rapid rate, partly because if you are going to change, you have to change, as I understand it, all of the accident source terms in the NUREG, not just one item. It would be nice to cafeteria shop through the list of things that might help your plant if you go to the new NUREG, but if you really are going to use all of the requirements that are there, it becomes a pretty extensive proposition and probably more than a lot of utilities at this time would want to get into without at least having some idea of what that is going to cost them.

Crom: I think the System 80+ experience is a little bit different because the new source term we are using does not use any new removal mechanisms. I believe the AP600 is using different removal mechanisms for the passive plants. Current plants would most likely see an improvement, if they utilize new source terms, in elimination of technical specification LCOs on the carbon filters. That was the greatest benefit as far as System 80+. Sreela is more familiar on this than I am. Eliminating technical specification limits on the Reg. Guide 1.52 test eliminates limiting conditions for operation should the test fail. This would be the biggest benefit that current plants could see, and they could probably eliminate the carbon filters all together. In System 80+ we still have the carbon filters because we have a stringent atmospheric dispersion factor in determining normal 10 CFR 20 and Appendix I limits. For this reason, we are still required to have non-safety carbon filters in the ventilation systems. But we were able to eliminate all the technical specification limits on all carbon filters except for the control room ventilation filters.

Ferguson: I think that I may not have expressed myself properly. The question I had was not the advantages the current nuclear plants would have if they switched to the new source term. There are many advantages. What I was really trying to find out is what the European nuclear industry, in the advanced light-water reactors, whether they are going to switch to the new source terms or something similar to that or whether they would stay with what their current regulatory requirements are based on. It was more of a questions on the European nuclear industry.

Zodiates: I think that is more of a question, not to the utilities, but to the regulatory board, because they define the rules of the game and the utilities operate by the rules.

Ferguson: And there is currently no such focus in the European Regulatory Boards.

Zodiates: Well, we have a few presenters here. Maybe they should give us their view.

Chair: Would anybody out here like to handle that?

Crom: Let me take a little stab at that because I was involved in a study for British Nuclear Fuels and I believe that the old TID source term was a lot more conservative than what is currently used in the U.K. In fact, I think the UK currently uses source terms closer to our NUREGs. More realistic severe accident source terms compared to what the U.S. has; at least that's what my experience was in dealing with British Nuclear Fuels.

Mirda: I'm from the Industrial Hygiene section of Consolidated Edison. My question is geared toward ergonomics. In the design of these new generation power stations, is ergonomics being incorporated into the new designs, and is the maintenance in a lot of these systems being looked at with maybe auxiliary type systems and equipment to expedite work on some of these systems. In our stations we see a major portion of doing some of this maintenance work involves just getting to a valve or setting up an area.

Crom: Let me take the first shot at that, and the answer is yes. We have spent a significant amount of time looking at the maintenance and the access. One of the things that we have recently done is to look at the staffing level. For the nuclear industry to continue, we are going to have to reduce the operation and maintenance costs from where they are in current plants. One of the main things that the industry is trying to do is to reduce the staff. The current 1300 or 1100 MW may have staffs of 1,100 people per unit. The estimate that we have done for System 80+ is somewhere around 750. A lot of the reduction is in maintenance personnel. Improved access and design of systems will require less maintenance and less maintenance staff.

Lau: As far as the AP600 goes, I know that we have looked at what is required for such operations as removing a reactor coolant pump. As I indicated, we have a cart for pump transfer, many different ways in which you can perform a job with less people, robotics -- all of these things go in line with your question. As far as the number of people for plant operation, the AP600 sort of starts as a base by looking at the Point Beach Plant, and here you are talking, I believe, 250 people as an operating staff and probably AP600 should be able to function with that number or less, although I've not followed the progress in that particular area recently.

Haynes: Let me comment on that briefly. Certainly, in the design of our Darlington plant, I would say one of the best things that we did was to allow more space for maintenance in areas where that is required. It is particularly important in most of our work areas because of the requirement for tritium protection, and therefore, you are wearing supplied plastic suits and we also always have to allow for air supplies. There is no question that we did that. We we did not do it very well at all in the design of our tritium-removal plant, and we got stuck with a fixed design and we are paying for it now in terms of ease of maintenance.

Khan: My question is to Rolf Riess. I wonder if in the newer Convoy designs, where you have these very low doses, were there any other measures taken, things like compartmentalization of components, bigger laydown areas, all kinds of other measures that the other people have talked about, or are those designs essentially identical with the older designs except for the removal of cobalt from the internals?

Riess: I would like to answer this question by simply showing a slide. This slide shows the exposure as a function of the calendar years, again, the three groups that I discussed during my presentation. If you consider the changes that were made from the first to the second group, there is no difference in the materials concept. That means, regarding the design of the unit, access to the unit, the options for maintenance, all these aspects have been introduced into the second generation and, of course, into the following generation of plants. There is a similar scale jump from this so-called first generation of plants to the second one in both design and shielding. And the final step is then to implenent or to reduce the sources of cobalt-60 and cobalt-58.

Na: This question is addressed to Mr. Hinson, NRC. I would like to know the NRC's position for whether it is an evolutionary or passive-type reactor, will you try to implement what we call the TEDE, the total effective dose equivalent concept or not?

Hinson: Yes, the staff will definitely implement the TEDE concept described in the new part 20 in all future passive and advanced reactor designs.

Na: My next question is to the two vendors of the AP600 and System 80+. Do you have any definite schedule to meet those NRC requirements?

Crom: As far as the System 80+ is concerned, we did change from the old 10 CFR 20 to the new 10 CFR 20, not only for shielding and airborne concentrations inside the plant, but also for a fluid analysis that we did for liquid and gas releases. The new 10 CFR 20 requirements are also in our radiation protection design acceptance criteria (DAC). We have to meet those particular limits in the new 10 CFR 20 in the detailed design.

Lau: I believe the same answer would essentially apply to the AP600. All of the access criteria for the various locations throughout the plant where you would do maintenance and other operations have access times and commensurate compatible dose rate requirements that meet the new 10CFR20.

Egner: Rolf Riess, you have been extremely successful with your Convoy plants. You could show figures down to, say, 20 man-rem per year, and you still have plans for future improvements. Could you really motivate further work from cost-benefit point of view, and why do we stop? Soon we are talking about a fraction of a man-rem per year for 1000 megawatt reactor.

Riess: The question was already raised during the presentation, what further improvements do you have in mind. Of course, you are right. If you start discussions with utilities what is the value of reducing, let's say, 20 man-rems per year to 19 man-rems per year in plant, so you will never end up with a cost-benefit or a benefit on your side as a utility. But the philosophy in German is that if you can introduce new technology which helps you to keep radiation fields down, it should be introduced. I mentioned a few things like reducing the Inconel surface in the system, consideration of implementing trace element injection into the primary system. This comes specifically from the older plants. Again, if you recall the slide that was just shown, we have these old plants which have high radiation fields and one of the simple considerations is just backfit the old plants with the features of the new plants and you should come down. But it is not so simple. It would take another presentation to explain why we couldn't or can't make these changes in these operating plants. Numerous political aspects are playing a major role in this regard. So we are looking for new solutions and they will be implemented. I wasn't involved in answering all the questions when reg guides came up saying you have to fulfill this. This is due to the fact that we have in Germany a different kind of philosophy. In the past, I can't tell you about the future, but in the past, if there was a technical problem, the German philosophy was to find the best technical solution and implement it and there were no major cost-benefit studies made on these issues. The best technical solution was implemented. And it was not ALARA principle, it was a principle which we called As Low As Possible, ALAP. The best solution to reduce radiation fields was implemented. So coming back to your basic question, do you see any chances for improvement. Yes, I see. But if a cost-benefit analysis is required, then you can state it would be extremely expensive, at least for the recent plants, to implement these features.

Chair: Are there any other questions, statements, by the workshop participants at this point?

Hinson: I would like to exand on an answer to a quesiton that I was asked earlier concerning whether the "next-generation" designs addressed the issue of temporary shielding use. The level of detail in the "next-generation" reactor designs which are approved by the NRC (through the issuance of FDAs) are lacking in several areas. It is the responsibility of the individual licensee wishing to incorporate one of these "next-generation" designs into its license application to provide certain details such as operating procedures, organizational structure, site-specific details, etc. Although the "next-generation" plant designs all state that temporary shielding will be used when needed, it is up to the individual licensee to specify how much temporary shielding will be available, what types of temporary shielding will be used, and the procedures and criteria for when and where this temporary shielding will be used.

Rescek: I asked a question during your presentation on the temporary lead shielding. I guess the question I am really looking at applies to the regulatory process. I understand that there are concerns over hanging lead on safety-related systems, and as a part of the process of looking at these new designs are the regulators going to be looking at how much lead and to what extent lead may be considered in the future when these plants are built that might be hung on these safety systems so that we don't get into the issue now of dealing with the regulators at our regions about "can you hang lead on this system or not, and where is your justification." I'm looking for a little bit of more proactive relationship between the designers and the NRC on addressing this issue.

Crom: Let me address that a little bit. As Charlie said, a lot of the piping analysis is not complete as far as design certification. The issue you discussed is whether it is considered as a piping loads and the pipe stress analysis. The answer is yes, it will be in the detailed design. That is one of the loading requirements in our piping analysis criteria contained in the SAR. I believe it's also an EPRI URD requirement that you consider the loading of lead shielding. We plan not to use a lot of lead shielding; however, in situations where you may be doing a unique maintenance situation, it may not have been considered.

Rescek: Yes, in fact, there is a paper Wednesday morning with Sargent & Lundy and Commonwealth Edison co-authored about PC-based programs to calculate the loading that you can put on some of these lines from a seismic standpoint. But we spend lots of money every year doing these types of analysis, and, in fact, the industry as a whole probably spends millions of dollars a year on these analyses to hang lead. If there is something that can be done on the front end, either put permanent shielding in these locations to minimize the use of temporary lead shielding, or to make it easier to pre-approve some of the areas where you think you are most likely to have the need for temporary shielding, that being done up front could save the industry lots of dollars down the road that we are spending today hanging lead on the existing plants.

Crom: I definitely agree from Duke Power's experience.

Riess: I would like to make a comment again on the previous question directed to me, namely, why do I look into further improvements to keep down radiation levels. Again, we take the position and the philosophy that you have to prepare for the future, and there are a few clouds on the horizon as we see it. I will give you an example. The economics in our country and I think worldwide for the nuclear stations is driving the operators to go to longer cycles, to have higher thermal efficiency. Immediately one starts to consider higher void fractions in the core, start boiling. This is a serious consideration. If that is done all of a sudden plant chemistry will be taken to the limits because you will concentrate lithium, thereby increasing fuel corrosion. This, in turn, requires reduction of lithium again, and that increases radiation fields, bringing us back to the old cycle. So you have to be prepared, and you have to have new answers for the questins showing on the horizon.

Co-Chair: (Bari) Would the panelists like to query each other? I know you've had a long, exciting first day of the workshop. I congratulate you on your stamina and attentiveness during this long day. I would like to thank my co-chair, Frank Congel, for participating. Finally, I would like to congratulate and thank all the panelists/authors for participating through this session. This session is closed and have a great evening.

Baum: Before you leave, I would like to close the technical meeting for today by saying that I am really impressed with the things that we have learned today, with the progress that has been made in the past five years since our last workshop, and I'd like to thank all the speakers for their very excellent and informative presentations.

BANQUET PRESENTATIONS

BANQUET PRESENTATIONS

Preamble by Tasneem Khan:

In the days before football was invented, the great spectator sport in ancient Rome used to be to throw Christians and slaves to the lions. On one such occasion a thin, skinny slave was brought forth before the Emperor and the assembled throng, as a sort of first feature, before the start of the real fun and games. These would star renowned gladiators, the Olympic ice skater Tonya Harding, hundreds of martyrs and many big cats. A ferocious lion was brought forth in a cage and the cage door was opened. There was a hush in the crowed as everyone waited to see what would happen.

The lion growled, slowly came out of the cage and eyed his prey, to see whether, in this man of skin and bones, there was any portion succulent enough to get his teeth into. The slave, from natural habit, looked at the lion diffidently, as if to say, "Can I be of service to you, kind sir?"

But then the slave thought of something, moved over to the lion. and whispered in his ear. The lion was rather disconcerted. He looked up somewhat abashed at this crowd of 20,000 unruly Romans, looked at the Emperor's box, and then meekly backed away into his cage.

Everyone was completely taken aback. The crowd was soon in an uproar. Some were laughing, some were yelling, some were applauding the slave, and some were debating whether he was a hero or a sorcerer.

The Emperor asked that the slave be brought forth. He took off a valuable ring from his finger and handed it to the slave and then asked him what it was he had whispered in the lion's ear. "Caesar, I merely reminded the lion, If you eat me for dinner, you will have to give an after dinner speech in front of all these people." Ladies and gentlemen, I identify with the lion!

Fortunately the task of giving the after-dinner speech belongs to someone far more eminent than myself. It is my pleasant task simply to start the proceedings, as a sort of first feature, before the main event, by first introducing the people directly responsible for organizing this workshop and then those who have been closely involved with the work of the ALARA Center. I will request the members of the ALARA Center staff to stand up and be recognized as I present them. In the interest of economy, I will request you to hold your applause till the end.

First, I present two young men, Mr. Clifford Yu and Mr. James Xie, who have recently gotten their engineering degrees. These gentlemen have not only helped in organizing this workshop but have done some superb work in publishing two issues of our newsletter, producing one NUREG report, helping in developing and maintaining our on-line services, and a host of other tasks.

Next, Mr. Justo Estrada who came to us on a minority student program. His work at the ALARA Center was so prolific that he was selected by the National Science Foundation (one of three students) to describe to the public the meaning of the word ALARA and his work with the ALARA Center. He has volunteered his services for this workshop.

Maria Beckman is the ALARA Center secretary. She is also our secret weapon. Like a great general, where ever she is needed, she is there "the fastest with the mostest." She received the Spotlight Award from Brookhaven National Lab for her work for the ALARA Center.

This team and I have arrived at a clear understanding of responsibilities. Whatever you like about this workshop, you can credit to me; whatever goes wrong, you can blame them. More seriously, thank you all for the many, many things you have done for this workshop and for the ALARA Center.

I will now introduce the people who are directly responsible for the ALARA Center as a whole.

First, Bruce Dionne. We were fortunate in snatching Bruce back from the nuclear power industry. His principal work at the Center is for the Department of Energy, but he still finds time to help with NRC work when called upon to do so. Bruce's tenacity in pursuing an objective is well known throughout the division. He is a very great asset to the ALARA Center.

Darryl Kaurin's principal interests are projects on hot particles, and on tracing the origins of enriched uranium. He is sometimes required to volunteer his services to the ALARA Center. He does so each time with brilliance. He comes from the West where people are known for their hospitality and big heart. In keeping with that tradition, recently he invited our whole division to his wedding -- two thousand miles away in Montana. He was kind enough to provide us complete directions, including a map on how to get there.

Our NRC project manager is Alan Roecklein. After working with Alan for nearly 10 years, we have at last succeeded in cracking the code. We have found out the big secret. At the expense of getting in trouble with our security, I will today divulge that classified information to you -- the secret of his success as a project manager. He gives people sufficient freedom of action to get their creative juices flowing. He then reinforces their ideas with excellent advice and guidance. He carefully monitors their work for results. Workers, not only in the U.S., but around the world, sometimes from as far away places as Taiwan and Romania, thank you, Alan, for all that you have done for the betterment of their health.

John Connolly is our Department of Energy project manager. He has his own magic formula for success, written in the slightly different DOE code. We have not been able to crack that code yet, but John, we are working on it. The results of your efforts speak for themselves.

I now come to our mentor and Guru, John Baum. In his relentless pursuit of perfection, he messes up the drafts of our documents with copious comments. His frugality in running his division, is proverbial. When we asked BNL management to help us move some computers for this conference, they said, "You are from John Baum's division, so you will probably want a freebie?" He counsels us with sage advice. He comforts us in moments of crisis. He holds our hand when we most need it. In fact sometimes he has to hold *back* our hand to keep us from pressing the panic button. Thank you, John, from all of us.

Finally, I would like to show you how, in some strange way, the composer Wolfgang Amadeus Mozart is responsible for the health of radiation workers here in the United States. One evening, several years ago, Charlie Meinhold, now President of the NCRP and then our chief, asked me to stay back after work to write a proposal with him for the U.S. Department of Energy. When I joined Charlie in his office, the first questions he asked me were whether I liked classical music and who my favorite composer was. I thought this a strange beginning to what I expected to be an evening loaded with technology. However, I replied, "Yes, I do like classical music, and my favorite composer is Mozart." Charlie took out a remote-control device from his desk and asked me casually, "How about Concerto number 17 for Piano?"

Soon, in the quiet of the evening, two scientists were working on the proposal with Mozart's Concerto playing softly in the background. The music not only helped us to concentrate better but also to generate some very good ideas. The proposal resulted in the creation of the DOE ALARA Center. Thus (although the main credit goes to Rick Jones for accepting our proposal), we do hold Wolfgang Amadeus partly responsible for the health of a large number of radiation workers here in the United States.

Thank you Charlie and thank you all who have made the ALARA Center possible.

The ALARA Center operates in Brookhaven under the aegis of the Department of Advanced Technology. It is now my pleasant task to introduce our Department Chairman, Dr. Romney Duffey, who has agreed to say a few words and then introduce our after-dinner speaker. Dr. Duffey comes to us from the U.K. where he worked for 10 years at the famous Berkeley Nuclear Laboratories of the Central Electricity Generating Board. He came to the United States in 1977 and was a Senior Program Manager at the Electric Power Research Institute, directing programs on power plant safety and performance. In 1987, he joined the Idaho National Engineering Laboratory, was Deputy Department Manager and directed programs on Reactor Technology and Waste Management. In 1991, he became Chairman of our Department of Advanced Technology. It is perhaps the largest department in BNL, and its divisions carry out a multifaceted program which covers many areas such as safety, risk analysis, advanced engineering concepts, nuclear safeguards, waste management, structural analysis, and radiological science. Ladies and gentlemen, Dr. Romney Duffey.

Introduction of After-Dinner Speaker by Romney Duffey

Thank you, Tas. On behalf of everybody here, I'd like to thank you for this tremendous meeting. I came here today not knowing quite what to expect, and it is one of the best meetings I have been to in many times. It is so focused; it is so well done. We owe you a tremendous vote of thanks for this work. In fact, we have decided to make him a Group Leader so that he can do even more work for us in the future.

I am going to make my remarks as short as reasonably achievable. I remember my first experience with radiological protection was when I was a child and was looking at my feet through one of these machines where you can see whether your shoes fitted or not. Every time my mother went anywhere near the stores, I always rushed into the shoe store and wiggled my toes under there. Things have changed, haven't they?

There is little to say about Charlie because it has almost all been said, except that he said to me, "Only say the things about the NCRP and the ICRP." And I said, "No, no, there's more to say than that." Charlie is a remarkable person, not only as a Senior Scientist and Deputy Division Head at Brookhaven, where it is my good fortune to commute with him on occasion to Washington and back, but he also does all of these other remarkable things. As President of the National Council on Radiation Protection and Measurement. That's a job he does part time. He insists on doing it part time because he still wants to be associated with Brookhaven. He is Vice-Chairman of the International Commission on Radiological Protection also. I guess that's another part-time job. Also, he is President of the National Radiological Protection Association. I assume that is also a part-time job. And since he can't do everything, he admits to being a past president of the Health Physics Society. This is a remarkable record for someone who literally has been working as a scientist at Brookhaven since 1957, and he has been someone to whom we all turn for advice and remarks on radiological protection. With all of these contributions, I know he is exactly the right person to talk to us tonight as he speaks about the history of radiological standards and radiation protection standards. It is with great pleasure that I introduce Charlie Meinhold.

THE HISTORY OF RADIATION PROTECTION

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Man-made radiation was born in physics, its childhood was in medicine, its teenage years in weapons development, and its adulthood at the confluence of all of the peaceful uses of atomic and nuclear energy, where it is today.

Just prior to the beginning of this century, gas-discharge physics was the darling of every experimental physicist. Every high school science and college teacher had a gas-discharge tube, and many of the scientists working with this equipment suggested that there might be radiation associated with the discharge. They knew they could make interesting things happen inside the tube. They weren't quite sure what would happen outside. Lenhart did bring electrons outside the tube through a thin window, but it was Roentgen who decided that some of the radiation could be penetrating the glass. You may know the history. He was adjusting the high voltage on his gas-discharge tube with a fluorescent screen in his hand. As he was adjusting the voltage, he saw the screen fluoresce. He realized that he was observing the results of a penetrating ray, which he called the X ray. He published this observation in less than 30 days (January 1986), and within 30 days of that publication there was the first reported radiation skin burn. So radiation came in with a flurry. Everyone who owned a gas-discharge tube learned that if they got the energy of those rays up high enough, they could make X rays.

By and large, it was the medical community that recognized the enormous potential of the X ray. It was interesting that medicine, at that time, was going through the throes of electrotherapy. Although this practice was being discouraged by the medical community as a whole, the practitioners were still there, and X rays became a marvelous new field for them. The next few years became known as the era of "bullets, bones, and kidney stones." The physicians realized from the beginning that there were potential hazards from radiation exposure. There were ulcers that didn't heal, and there were frequent reports of skin burns, both among the patients and the physicians. It was a long time, however, before anyone thought much about what was causing some of the effects since they had been seen in patients treated in electrotherapy. About 1915, only 15 years after the introduction of the X ray, a physicist named Russ, in England, suggested a set of rules to the British Radiological Society, which they actually printed up on a card, giving advice on avoiding unnecessary exposures. These rules were not very definitive, but at least the Society and, in particular, the authors, understood that there was a problem. However, not much action was taken.

You will note that this advice came to the British Radiological Society. As indicated above, the medical community had adopted this technology, and once a medical association takes ownership of a modality of this kind, they claim exclusivity. In the United States, and pretty much in England and in France, a physicist couldn't publish an article unless he had a physician sponsoring the paper. As a result, most of the literature was related to clinical effects and to clinical use. The situation was different in Germany, where physics and medicine grew up together, and the medical community embraced the physics community. This was primarily because medicine was more heavily regulated in Germany than it had been in these other countries.

Protection advice wasn't heavily organized until, in 1921, that same set of recommendations that had been brought before the British Society in 1915 were now adopted by the British Society as rules that every physician should use. This change occurred because of the development of the hot cathode tube by Coolidge, an engineer at General Electric. This tube was able to produce much higher currents and much higher energies. Many of the radiologists now recognized the tremendous hazard that it posed for them and their patients. In addition, World War I had just taken place, and hundreds of X-ray machines went into the battlefield, mostly with the Coolidge tube. In addition, there were many reports in the public press about anemia, i.e., people ill from blood disease, after the war.

An interesting thing happened at this time that changed the course of radiation measurements. These battlefield machines had to meet military specifications. The Army and Quartermaster Corps was just as difficult then as they are now, which meant that when you went to sell an X-ray machine to the Army, it had to meet a standard, and the National Bureau of Standards provided exactly that. As a result, the physicists involved became a lot more interested in measurement and quantification than had the physicians who had depended upon how red the skin got and whether or not they obtained a good image. Radium had also been discovered shortly after the X ray. The only way you could specify the quantity of radium was through measurement, and at $100 per gram that was very important, so a lot of people were involved in improving the measurement capability in order to make sure that they weren't being cheated. Commerce had its way. Finally, at last, there was measurement of activity.

In 1922, a quantum change occurred. Mutschuler, in the United States, and Sievert, in Sweden, were worrying about the problem of radiation protection. Mutschuler visited a number of well-run clinics in New York City and found that they could operate well without anyone being exposed to more than .01 of an erythema dose in 30 days. This was very important because it provided the first "limit." At the same time, and operating independently, Sievert arrived at a recommendation of .1 erythema dose in a year. Remarkably, they ended up with the same number.

In 1925, the International Commission on Radiation Units (ICRU) was formed as an advisory committee to the International Congress of Radiology at its meeting in Stockholm. Even at the time of formation, the International Society recognized the need to define an exposure quantity. In 1928, the ICRU settled on the amount of ionization in a given quantity of air as the standard, and the Roentgen was defined.

Shortly thereafter, both the International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection and Measurements (NCRP) made recommendations dealing with exposure levels. The ICRP recommended no more than .2 R/day. It turned out that this is a reasonable measure of about .01 of the erythema dose in thirty days, so that essentially what they had done was to adopt, in a way that could be measured, what Mutschuller and Sievert had recommended three or four years earlier. This means that the first recommendation on dose, although quantifiable, was based on skin reddening. Two years later, in 1934, the NCRP recommended .1 R/day. The ICRP recommendations applied to measurements made at the surface of the body, while the NCRP recommendations applied to measurements made free in air. Measurements made at the surface of the body with the soft X rays would indeed be just about twice what they would be free in air. In fact, the NCRP and the ICRP recommendations were virtually the same.

In the middle 1920s, there were a number of young women working as radium dial painters in New Jersey who tipped their brushes between their lips -- the famous radium dial cases. The physician at that facility came to New York University (NYU) and asked their toxicological group to visit the factory to see if they could help them to understand why there was such serious medical problems such as necrosis of the jaw with these women. The toxicologist who went from NYU, Glump, was very confused because he expected to find red phosphorus, which was a known industrial poison. All he could find was radium. He wrote to Madam Curie and asked if it was possible that radium was doing this. NYU has in its archives a letter from Madam Curie claiming that this fellow was certainly a charlatan, and that radium was for the good of mankind and should not ever be considered to be evil. I don't tell this story to defame Madam Curie. In fact, I will tell another story about her. Madam Curie was a Nobel Laureate by the time of the first World War. In spite of this, she put aside her radium, left her laboratory, and took an X-ray machine into the field to help in the medical service of the soldiers in the war. Half way through the war she came back and taught people how to run X-ray machines. She essentially gave up all her research over the whole wartime period in order to be of service to the soldiers who were wounded.

Eisenbud has made the point, and I will reiterate it here, that it was remarkably fortuitous that, by the late 1930s, the community had at its disposal two recommendations. They had a .1 R/day and they also had a number from the radium dial workers. Robley Evans, from the Massachusetts Institute of Technology, had established that if an individual had no more than a microgram of radium in the bone, it was unlikely to cause damage. The NCRP settled on a recommendation limiting the intake of radionuclides which went to the bone to .1 μgm. Without these numbers, it is hard to imagine what might have happened during the Manhattan Project.

During the war there was a great deal of research in radiation biology going on in places like Oak Ridge, the University of Rochester, Berkeley, the University of Washington, etc., essentially all over the country, to try to get information on the effects of ionizing radiation. Perhaps the most influential radiation protection recommendation at that time was being made by a committee at the Tripartite Conference Meetings. Canada, the United States, and Great Britain set up a framework of radiation protection which they brought to the ICRP and the NCRP in the late 1940s. By the middle 1950s, both the NCRP and ICRP produced new sets of dose limits derived from all the data obtained during World War II.

They recommended 600 mrem per week for the skin, and 300 mrem per week for other organs. I was fascinated to realize that .1 R/day is .6 R/week, which is 600 mrem per week. Essentially, the 600 mrem per week for the skin goes all the way back to .01 of the erythema dose of 1928. The 300 mrem per week limit is more interesting. If you irradiate the whole body with 150 kV X rays, the dose at a depth of 5 cm is just about half of that at the surface. If you were protected by a limit of .1 R/day with soft X rays, the dose to your tissues at 5 cm would be .05 R/day. Now, if we are going to irradiate you with high-energy gamma rays, we should have the same limit for the skin, 600 mrem (.1 R/day) and half of that value for dose at depth 300 mrem (.05 R/day). Again, all based on .01 of the erythema dose per month.

Starting in about 1955, we entered a new era characterized by weapons testing and the public response. There were people all over the world concerned with what was happening. Specifically, there were two individuals who led the scientific community in expressing concern, Mueller, a geneticist, who had been speaking about the linearity of genetic effects even during the late 1930s, and Linus Pauling, who worried about internal dosimetry. As a result of the public concern, the National Academy of Sciences in the United States and the Medical Research Council in the United Kingdom were asked to review the data. Both of these Committees came up with about the same answer. They focused their attention on genetics. They said that it was unlikely that all of man's suffering and pain from genetic abnormalities came from natural radiation background, but that some of it did. Such a consideration bracketed the genetic risk since they knew the natural radiation background levels and the natural incidence of genetic effects. Based on this analysis, both committees came up with an estimate that suggested individuals should not receive more than 50 rem to age 30 and another 50 rem to age 40. I might add that I was able to talk with Eugene Cronkite about this many years ago. Dr. Cronkite was Chairman of the Somatic Committee of the National Academy of Sciences panel at the time of the preparation of the 1956 recommendations. I asked him if the recommendations on exposure limitation came from considerations of the radiologists who had been shown to have an excess incidence of leukemia. He answered that the dosimetry was so uncertain that they could not estimate the dose nor the risk per unit dose associated with leukemia among the radiologists. He noted that what they did decide was that they would accept the genetic panel recommendations, and the Academy recommendations were therefore based almost entirely on the genetic estimates based on a linear extrapolation. Shortly thereafter, Dr. Russell at Oak Ridge showed that there probably was a dose-effect relationship for genetic effects, but it wasn't taken into account.

NCRP and ICRP had to decide the way in which they would recommend that the worker be protected under these new recommendations. As we know, the answer was (age - 18) x 5, which the Nuclear Regulatory Commission discarded just three months ago. The whole body limit was 3 rem/quarter and (age - 18) x 5 and 15 rem/year for individual organs. By the way, 300 mrem/wk for 50 weeks results in 15 rem/year. Again, the critical organ number of 15 rem finds its way back to .01 of the erythema dose in 30 days.

One of the things that I used to say when I talked about this subject in 1977 was that we didn't have a very strong scientific basis for our dose limits. However, by 1977 this situation changed dramatically. This was a result of information that came, not in 1977, but from the period 1960-77 based primarily on data that was becoming available from the Japanese survivors who had been under study from the time of the bomb.

I would like to stop here for a moment because people ought to understand the enormous contribution those survivors and the government of Japan have made in this follow up. I should add that funding for that is now is question by the Department of Energy, and it is incumbent on us all to see if we can help to maintain it.

The United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) and the National Academy of Sciences, here in the United States, review the data that comes from Japan. They noted that in 1960 the incidence of solid cancer in the Japanese survivors was slightly greater than might have been expected in that population if it had been unirradiated, but excess leukemia was clearly evident. In 1960, they estimated that other cancers are about equal to leukemia. In about 1962, they estimated that other cancers were about two times leukemia, and by 1965 they were suggesting a slightly higher ratio. By the early 1970s, the UNSCEAR suggested that the ratio of solid tumors to leukemia was five or more. What was happening was that the leukemia incidence has a wave effect. The increased incidence starts to show up three of four years after exposure, stays elevated, and drops off. But the solid cancers don't follow that pattern. What we now believe from the data that we have seen in the most recent reviews is that attributable solid cancer occurs at the same ages that you would normally get it if it weren't due to radiation. Therefore, the excess incidence of solid cancer is going to keep increasing. In 1977, UNSCEAR estimated the risk for individual tumors. This was expanded on by the ICRP in its publication 26 in 1977 -- the first scientific approach, I believe, in radiation protection recommendations. The ICRP estimated the total risk to be about 1×10^{-4}/rem. They then compared radiation risks with the risk in safe industries. In safe industries at that time, one person in ten thousand died each year (1×10^{-4}/year) and the ICRP suggested that the radiation workers ought to have at least that level of protection. The ICRP then set a limit of 5 rem/year on the basis that most people who were protected by a limit of 5 rem/year aren't likely to exceed 1 rem/year, and, therefore, the average risk will be the same as that for safe industries. Of particular importance was the concept of a risk-based system. Following this logic, the ICRP provided cancer risk estimates of the more sensitive tissues and suggested that the annual limit on intake (ALI) be based on the specific risk of each tissue. The NCRP caught up with the ICRP recommendations in 1987 (as did the Environmental Protection Agency) and issued its Publication 91.

The recommendations of the ICRP (Publication 60, 1990) are based on further changes. In 1986, a later set of data from Japan became available which suggested two things. First, there is evidence of increased risks based on new dosimetry in Japan and some additional solid cancers. This new data also gave further evidence that cancer from exposure to radiation follows a multiplicative projection model, i.e., attributable cancers will occur at the age they would if there were no exposure, so it isn't until people get to be in their mid-seventies that these cancers are likely to occur. ICRP and NCRP have adopted this new risk projection model. Having such a model is needed to estimate what is going to happen to the Japanese over the next 20 years or so, because, in fact, only about half of them have died up until this point. It is very clear from the Japanese data that exposure to radiation at high dose rates results in excess cancer. You will note I said "high dose rate" since the doses that show these excess cancers are about 100 rem, but 100-200 rem is on the order of the lifetime exposure we might expect for the most highly exposed radiation workers. Therefore, we are talking about an extrapolation from high dose rates to low dose rates, and we must ask the question whether there is time for recovery and repair which might alter our estimate of risks at lower dose rates. ICRP's Task Group on Risk, chaired by Dr. Arthur Upton, suggested you might be able to reduce estimates from very high doses (dose rates) by about a factor of two to get the best estimate in the risk at low doses (low dose rates). The NCRP Committee on Risk, chaired by Michael Fry, suggested the risk at high doses (dose rates) could be reduced by a factor of two to three. What all this means is that we now are on a very firm basis in stating that there is excess cancer in the Japanese. We still have concern about whether we are overestimating the risk by a factor of two or three, or underestimating it by about the same factor. But at least this gives us confidence that we have a fairly firm understanding of the risks that people face. As we apply these risk estimates to deriving dose limits, the ICRP and the NCRP realize the risk estimates had increased by about a factor of four since 1977, when ICRP Publication 26 was published. Since the annual limit was 5 rem in 1977, you might logically divide by four and obtain a new limit of 1 rem/yr. The ICRP did note, however, that the new projection model also changed the most likely age of death from an attributable cancer. That changed from an expectation of death in the middle sixties to expectation of death in the late seventies. In addition, the ICRP felt it was important to base the limit on the risk to the most highly exposed individuals (for whom the limit is needed). Rather than using the safe worker criteria, the Commission felt that it was more appropriate to base their limits on a comparison with an individual worker at the upper end of safe industry risks. This turned out to be about 10^{-3}/year.

This approach is tolerable for the rare individual operating at the dose limit, but it is totally unacceptable to use for any kind of average exposure for individuals who are working in the industry. It is for this reason that ALARA is the essential element in keeping the average exposure far below the dose limit. The dose limits themselves are entirely unsatisfactory as a basis for designing a protection system. The ball is entirely in your court since ALARA is essential in protecting the worker, the public, and the environment.

SESSION 4

PATHWAYS TO ALARA

Co-chairs:

James E. Wigginton
Mary P. Measures

ELECTRICITE DE FRANCE's ALARA POLICY

4-1

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ABSTRACT

In 1992, Electricité de France -EDF decided to improve the degree to which radiological protection is incorporated in overall management of the utility and set itself the objective of ensuring the same level of protection for workers from contractors as for those from EDF. This decision was taken in a context marked by a deterioration in exposure figures for French plants and by the new recommendations issued by the ICRP. This document describes the policy adopted by EDF at both corporate and plant level to meet these objectives, by :

- setting up management systems which were responsive but not cumbersome,
- a broad policy of motivation,
- the development and use of suitable tools.

The document then describes some quite positive results of EDF's ALARA policy, giving concrete examples and analysing the changes in global indicators.

INTRODUCTION

When one thinks of electrical utilities, France is often viewed in terms of nuclear power, with EDF as the main player.

Electricité de France is one of the largest utilities in the world. In 1993, it had a turnover of almost $32 billion, clearing a profit of $520 million. Its workforce consists of approximately 118,000 employees who perform a very wide range of activities, including the design, construction, operation and maintenance of facilities involved in the generation, the transmission and distribution of electricity. These characteristics are quite unique among utilities, making Electricité de France a leader in its field.

Electricité de France is the only utility in France, generating, transmitting and distributing electricity to 29 million customers in 25 million homes and to 600 large industrial clients. In 1993, Electricité de France generated 424 billion kWh. In 1993, exports to neighbouring countries totalled around 61.7 billion kWh, representing around $2.4 million.

83 % of electricity generated in France in 1993 came from nuclear power plants, 14 % from hydroelectric facilities and 3 % from fossil-fired plants. Electricité de France operates 56 nuclear units with a total installed capacity of 58,880 MW.

An ALARA approach was first applied at EDF for replacement of the steam generators at Dampierre 1, both during the preparatory stage in 1988/89 and during actual operations in 1990.

Towards the end of the eighties, the ALARA concept was not a clear part of the radiological protection culture in France. This principle only appeared in French legislation for nuclear facilities en 1988[1]. Moreover, this period saw a steady deterioration in dosimetric results for each unit and for each GWh generated at French plants, both in terms of the absolute value, and in comparison with results in other countries (See Figure 1).

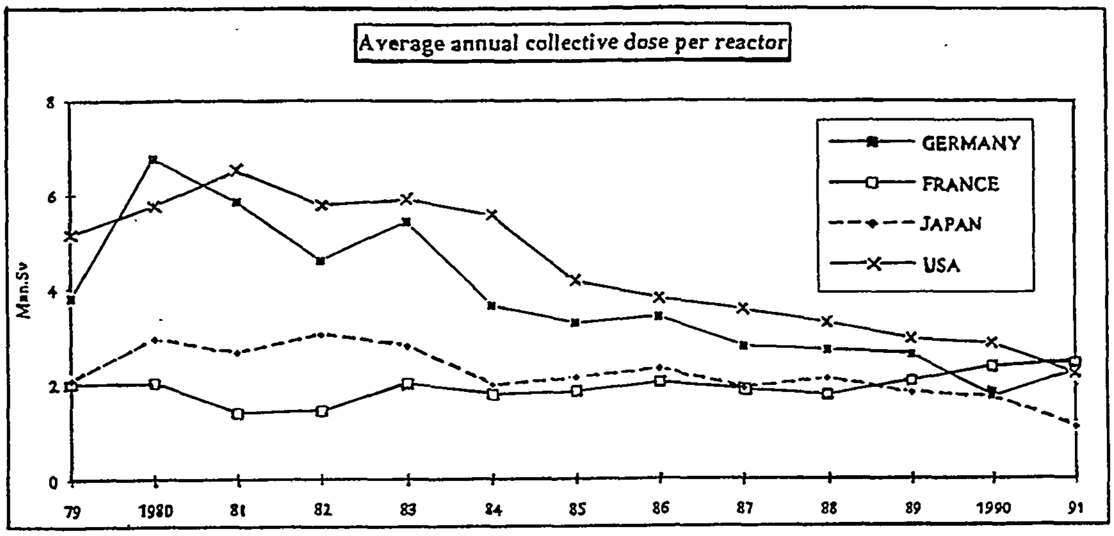

Figure 1 : Average annual collective dose per reactor (Germany, France, Japan, USA)

The results achieved for steam generator replacement at Dampierre Power Plant -2.13 man.Sv over 70 days (see Figure 2)- set a new world record at that time[2], and proved to EDF that application of the ALARA approach permitted an effective transition from the "*a posteriori* dose-limits-respect type of radiological protection" to *a priori* management of individual and collective exposures.

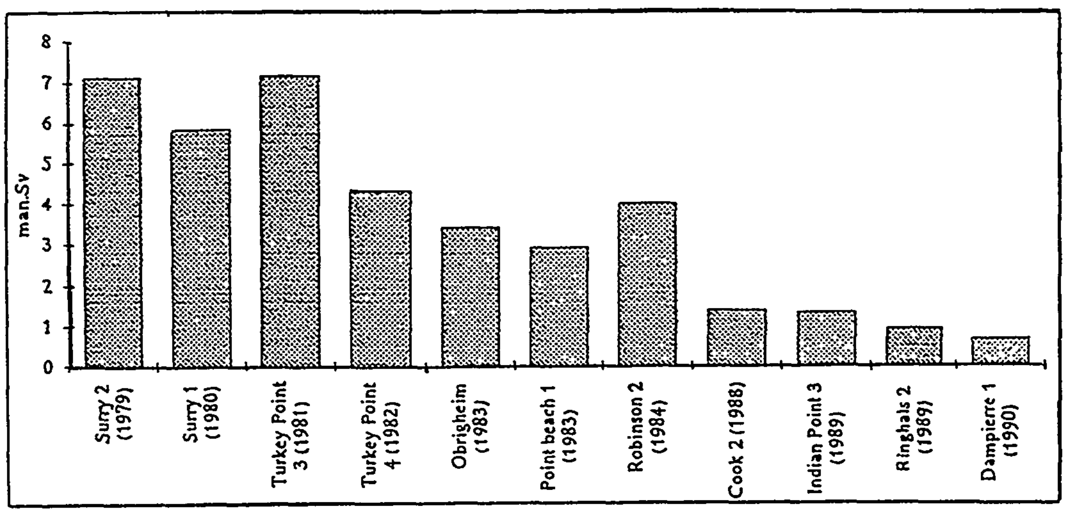

Figure 2 : Steam generator replacement total collective dose per steam generator (in chronological order)

So it was that EDF, with the aim of applying this approach to all operations carried out on the entire population of French plants, set itself an ambitious dual objective in 1992 for reducing exposure, thereby anticipating the changes in individual dose limit :

a) improve the degree to which Radiological protection is incorporated in overall management of the utility and decrease the average annual collective dose from 2.43 man.Sv per unit in 1991 to 1.6 man.Sv by 1995 ;

b) provide the same level of protection for workers from contractors as for EDF workers, and as a matter of priority reduce the exposure of those groups of workers with the highest individual dose levels.

In order to meet these objectives, EDF has set up special ALARA groups and committees, is trying to infuse an ALARA culture throughout the utility, has adopted a policy of motivating all players concerned by ionising radiation and is busy developing and using ALARA tools.

DEFINITE COMMITMENT FROM THE MANAGEMENT

EDF management has on many occasions clearly expressed the above objectives and reiterated that one of its priorities was to reduce exposure. This commitment was recently expounded in a forty page document[3], a sort of mission statement setting out the main thrusts and objectives of the utility's radiological protection policy. An action plan has been outlined for each objective to cover the period up to the year 2000. This document, with a foreword by the Deputy Managing Director of EDF, was distributed throughout the utility and outside.

A SIMPLE AND RESPONSIVE MANAGEMENT SYSTEM

EDF operates PWRs in 17 plants accross France. The ALARA committees and groups set up, which remain in essence responsive without being cumbersome, are quite naturally divided between corporate and plant level (see Figure 3).

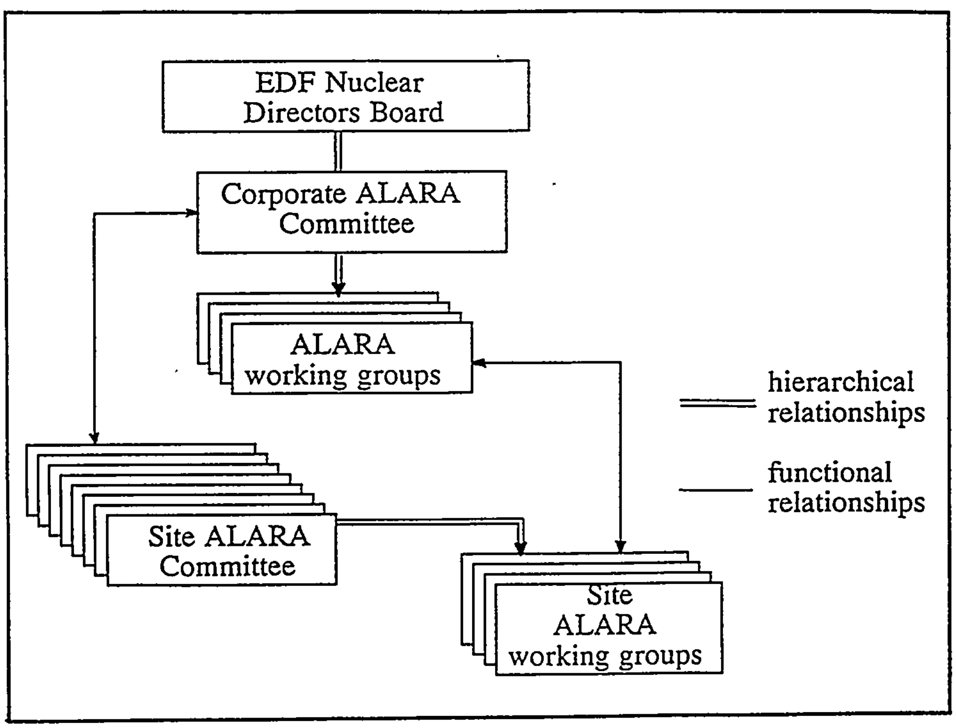

Figure 3 : ALARA management system at EDF

1. CORPORATE LEVEL

a) A Corporate ALARA Committee

This Committee was set up in December 1991 ; it is chaired by a member of the steering Committee of the EDF Nuclear Generating Division and is supported by the EDF corporate Industrial Safety and Radiological Protection Department. It has two types of members :

- standing members responsible for corporate departments (e.g. radiological protection, tool development, design of future facilities) ;

- representatives from the management of the nuclear power plants ; the plants are divided into three groups which take it in turns to be represented on this committee, swapping every eighteen months.

This Committee sets out the major thrust of EDF ALARA policy, ratifies target doses each year at corporate level for each type of reactor with the aim of meeting the objective of an average of 1.6 man.Sv per reactor by 1995, fosters inter-plant emulation to attain this goal, monitors result indicators and promotes experience feedback between plants.

b) ALARA Working Groups

Such groups can be set up when new problems arise at corporate level. In 1992, for example, shortly after the discovery of cracks in pressure vessel heads and in view of the importance of the doses incurred when inspecting, preventing or repairing these cracks, a corporate ALARA group was set up, at the request of the Corporate ALARA Committee, to implement an ALARA programme to remedy the problem. This group included not only representatives from the sites and from EDF head office, but also from the major contractors involved (Framatome, Jeumont Schneider, etc.). A group of the same sort has been in operation since 1989 for steam generator replacements (Dampierre, then Bugey and Gravelines Power Plants).

2. PLANT LEVEL

The local systems set up on the initiative of the plants mainly take the form of systems for co-ordinating the various professions at the plants. They also act as a relay : for relaying corporate policy to the plants and contractors, for relaying information between the plants themselves to promote experience feedback, for relaying experience gained in the field and for relaying plant suggestions to the corporate fora.

Practically all plants set up a site ALARA Committee between 1992 and 1993. This Committee is chaired by the site Director or Deputy Director and has executive powers. It provides a forum for dialogue. It groups together representatives from all the departments (maintenance, scheduling, chemistry, operation, general services department, occupational medicine, etc. not to mention radiological protection) and representatives from the contractors.

On the basis of the plant's own dosimetric objective, these Committees select those high-dose jobs to be most closely monitored and analysed, and decide what resources to use. The majority of the Committees are assisted by multi-disciplinary or cross-disciplinary groups responsible for suggesting cost-effective actions to reduce exposure, by acting both on the dose rates (conditions for implementing outage, oxygenation, development of biological shielding etc.) and on the exposure time (organising scaffolding and heat lagging work, training maintenance workers, development of special tools, etc.).

WIDE-RANGING POLICY TO MOTIVATE ALL PLAYERS

1. HOW TO CONVINCE THE DECISION-MAKERS

Several plants and corporate departments, with the help of experts, organised a day for managers to outline and discuss the ALARA principle and approach and how to implement them at EDF. The same sort of day was organised at corporate level in order to make the heads of external corporate contractors aware of this and to discuss the repercussions of EDF ALARA policy on relationships between the Operator and its Contractors.

The first step in implementing this policy was that of setting dosimetry objectives for each reactor, depending on its specific characteristics (type of reactors, existence of hot spots, etc.) and the work to be carried out (partial inspection, ten-yearly inspection, etc.). These objectives, which must at all times be consistent with the policy (target doses) laid down by the Corporate ALARA Committee, are negotiated between the plants and corporate headquarters before being incorporated into annuel management contracts and three-year plans ; the managers of the plants make a commitment to EDF corporate management to meet the objectives in these plans.

2. AN ALARA TRAINING POLICY

EDF then undertook a large-scale programme to train all players and make them aware of ALARA issues ; this was seen as a first step along the road towards changing the culture of all workers in the nuclear generating sector.

a) Targeting Workers Involved in Unit Outages

Exposure during outages accounts for 80 % of the annual dose, and so the majority of plants held two-day training sessions organised by experts from off-site for the unit outage management and preparation teams (outage manager and representatives from the various departments involved) : this training combines theoretical teaching of the basics of optimising radiological protection and practical studies of experience at other plants.

b) Training Instructors

About forty EDF workers (radiological protection workers, maintenance technicians, and design engineers) took part in one-week training courses on how to become ALARA counsellors and instructors at the plant or in corporate departments.

A new training initiative was set out in order to comply with the expectations of management staff from the contractors. This was intended to improve how people worked together, and to train staff to work better together to achieve the objectives of the population of nuclear power plants in respect of maintenance, and to become players in the ALARA approach.

c) Incorporating ALARA into Job Training

All these initiatives, no matter how effective they may be, are only a drop in the ocean when one considers the large number of people involved (tens of thousands of EDF and contractor personnel). In 1992, an ALARA module was therefore incorporated into the radiological protection authorizations for working in a controlled area, and modules of the same type are to be systematically included in job training, also for those persons who do not work in a controlled area, but whose work covers occupational exposure, in particular plant procedure planners or persons in charge of reactor operation.

3. INFORMATION AND AWARENESS

Each plant will develop its own policy in this area, and shall call on the imagination and creativity of its workers : display of objectives, ALARA information days at EDF and at the plant, ALARA posters, videos, articles in site or outage newsletters, competitions between teams, reception of new contractor teams, regular worksite inspections, radiological protection and ALARA items in the Hygiene, Industrial Safety and Working Conditions Committee or unit outage meetings, etc.

Motivating the players seems to be an important factor, both in the preparation stage and during operations themselves, in reducing the large number of anomalies which account for up to 30 % of the dose in France[4]. Motivation should be a constant concern, because experience feedback from certain plants has shown that as soon as the pressure is released, results worsen.

4. SPECIAL MOTIVATION OF CONTRACTORS IN THE CONTEXT OF A CONTRACTUAL PARTNERSHIP

There are no plans to implement an ALARA policy without the active participation of the contractors, since it is their workers who will perform the majority of inspection and maintenance operations and who account for over 80 % of the collective exposure, and since these contractors are developing many processes and tools.

EDF is therefore striving, in this area as in many others, to develop a contractual partnership policy by :

- incorporating ALARA exposure reduction into the specifications for the work to be carried out,

- studying the reasonable cost of contractor proposals in respect of work organisation, process modification or development of tools,

- incorporating dosimetric objectives into orders, without awarding financial rewards for achieving these objectives, but using the effective committed dose as a criterion for selecting the contractor during subsequent operations,

- demanding analysis of experience feedback on radiological protection on closing worksites,

- promoting the implementation of internal ALARA programmes for contractors, etc.

DEVELOPING AND USING SUITABLE RESOURCES

1. IMPROVING OPERATIONAL MONITORING OF COLLECTIVE AND INDIVIDUAL DOSES TO EDF AND CONTRACTOR WORKERS THROUGH AN INFORMATION SYSTEM

In order to improve operational management of exposure, EDF is gradually equipping its sites with a real-time computer management system (Real Time Dosimetry - known by its French acronym DTR), for the doses incurred each time a worker enters a controlled area, or each time he enters a sub-area.

Furthermore, in order to provide its workers and workers from contractors with the same degree of protection, EDF has set up a computer link between the various plants (the DOSINAT system) ; this allows contractor employees to be monitored by name when they move to another plant, thereby ensuring, both by questioning employers and by investigating the working conditions in the plants, that radiological protection be applied as strictly as possible with regard to dose limits. This computer application was licensed by the French state-run Data-Processing and Civil Liberties Commission (*Commission Nationale de l'Etat Français "Informatique et Libertés"*-CNIL).

In years to come, this database will be extented to the other links in the nuclear energy chain.

Moreover, since mid-1993, a new EDF access log is required for all workers assigned to work in a radioactive area when they arrive at the plant. The first page of this logbook lists the training undergone by the worker, and the third page gives dosimetric monitoring information together with DOSINAT dosimetric results and the results of whole-body-counts.

2. PROMOTING CORPORATE AND INTERNATIONAL EXPERIENCE FEEDBACK

Rapid distribution of operating experience is the key to the success of any ALARA policy. EDF, with a population of 54 reactors of similar design, is striving to increase the effectiveness of exchanges of operating experience between plants. In addition to the training structures mentioned earlier and the databases specific to each plant, corporate working groups have made it possible to :

- select high dose jobs, assign each of these jobs to a plant, apply a common procedure for predicting all job doses and for following up data during the jobs, enter this data into the computer application DOSIANA, draft an operating experience report and provide the other plants with all of this information. This application was first used for all operations linked to the inspection and repair of pressure vessel heads. Eventually, all unit outage operations should profit from this system ;

- build up a corporate operating experience file on radiological protection "good practices" and distribute it to as many people as possible at all sites (see Mr Rocaboy's presentation on scheduling in this Workshop) ;

- create a corporate operating experience file on treating hot spots (see Mr De Guio's presentation in this Workshop) ;

- create a corporate file on the treatment of problems relating to contamination with silver-110.

In parallel with this desire to improve French operating experience feedback, EDF has been an active supporter of implementing the ISOE system [5] to promote exchanges of operating experience between operators in different countries. EDF regularly updates the system with its good practice files and new operations or problems encountered.

3. CONTRACTOR APPROVAL

Quality training is essential for mitigating risks. In 1990, the French Committee for accrediting organisations for the training and dosimetric monitoring of workers exposed to ionising radiation (French acronym CEFRI) was set up. This committee was set up with the full approval of the operators (EDF, COGEMA, French Atomic Energy Commission and the French military), the French ministry for health and the radiological protection authorities, the ministries for labour and industry . This body issues approval, subject to a positive audit, to :

- training organisations,
- temporary employment agencies supplying staff,
- contractors employing staff working in nuclear installations.

This approval covers the quality of training and the management system set up to perform dosimetric monitoring and medical surveillance for field workers. Eventually, only contractors approved by the CEFRI will be accepted into nuclear plants.

In addition, EDF is changing its contractual relationships with contractors and introducing a clause stating that approval will be withdrawn from the contractor should a contract of employment be rescinded or suspended between this contractor and one of its employees reaching or exceeding a regulatory dose limit.

4. A CORPORATE POLICY FOR AUTOMATED TOOLS

One of the most promising areas for the ALARA concept is that of implementing a corporate development policy for automated tools ; this topic will be covered by Mr Cazin in his presentation at this workshop.

INITIAL RESULTS

1. THE TREND INFLEXION IN 1992-1993 (FOR DOSES)

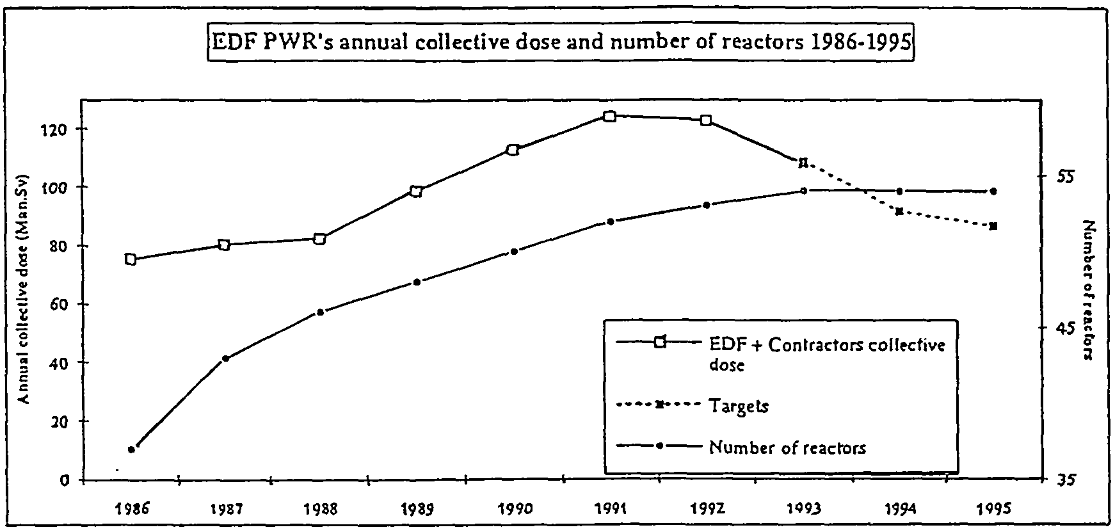

Figure 4 : EDF PWRs, annual collective dose and number of reactors (1986-1995)

Between 1983 and 1989, the average dose per unit per year ranged between 1.8 and 2 man.Sv. In 1990, this figure was 2.35 man.Sv, and in 1991 it was 2.44 man.Sv, owing to a large number of ten-yearly inspections that year. Promotion of the ALARA principle from 1992 onwards reversed this trend : from 2.36 man.Sv in 1992 to 2.04 man.Sv in 1993.

The target of 1.6 man.Sv which EDF has set itself remains very ambitious.

2. LARGE OPERATIONS OPTIMISED

a) Steam Generator Replacement

Steam generator replacement operations systematically give rise to planning work, monitoring and experience feedback in accordance with an official ALARA approach in the framework of a working group combining the EDF Engineering and Construction Division and the Operator.

The ALARA programme adopted for steam generator replacement operations is mainly centred on ;

- a study of how to optimise protection initiatives : water levels in the steam generators, biological shielding, decontamination of the ends of reactor coolant pipework ;

- strong motivation of the workers ; theoretical and practical training before steam generator replacement operations ; close relations between the ALARA team and workers during site work, making use of the many media and supports (readouts, projected dosimetry curves produced, worksite meetings and experience feedback, etc.).

The results of steam generator replacement operations at Dampierre 1 and Bugey 5 have been very satisfactory.Last results at Gravelines 1 in 1994 are still better..

STEAM GENERATOR REPLACEMENT

	projected dose (man.Sv)	dose achieved (man.Sv)
Dampierre 1 (1990)	4.5	2.13
Bugey 5 (1993)	2.6	1.55
Gravelines 1 (1994)	1.41	1.32

The work at Dampierre 1 showed, according to surveys of field workers carried out during work, the positive effect of ALARA initiatives, especially in the field of motivation.

b) The pressure Vessel Head Incident

The accumulated dose for the vessel head incident until end 1993 is as follows :

	annual dose (man.Sv)	Cumulated dose (man.Sv)
1991 dose	1.9	1.9
1192 dose	9.0	10.9
1993 dose	7.0	17.9

An analysis of the years 1991 and 1992 gave a total forecast to the end of 1992 of 16 man.Sv without the ALARA programme. The figure of 11 man.Sv achieved over the same period represents a saving of 5 man.Sv which can be attributed to applying the ALARA principle at vessel head worksites.

The operation to replace the vessel head at Bugey Power Plant was carried out at the start of 1994 at a cost of 0.2 man.Sv compared with the projected value of 0.45. This projected value was of course very imprecise because it was the first time the work had been carried out at a plant of this type. Nonetheless, the small risk incurred must be attributed to the good ALARA preparation of the plant in collaboration between the plant, the corporate departments and the various contractors.

3. REFUELLING OUTAGES

Golfech 1, a 1 300 MW reactor commissioned in February 1991, achieved dosimetry figures of 0.47 man.Sv for the year (0.45 man.Sv of which was due to the unit outage). This result is the best to date for a French reactor and shows that our reactors can aim for an average level of performance which will bring them close to the best in the world.

EDF corporate policy for the past several years has aimed at improving safety and competitiveness. This policy has produced good results and the analysis of these results shows that the most efficient units are the best in terms of safety, availability, cost and radiation protection. The feedback experience demonstrates that quality, efficiency, safety and radiation protection are closely linked together : any improvement of one of these items leads to progress for the others, being therefore a very important motivating factor.

CONCLUSION

The main objective in radiological protection is of course to protect man ; the main way of doing this is to monitor the doses received.

However, the introduction of the ALARA principle into corporate culture is changing the philosophy behind the initiative and behind behaviour ; to really protect man, we also need to implement residual risk management i.e :

- a predictive approach : anticipate exposure and the means of reducing it,

- an effective approach : reasonable use of resources given over to protection,

- an evolutive approach : one which takes account of changes in the technical, financial and social context.

For such a management system to succeed, all workers involved must be motivated. Operating experience has shown that such a consensus is easier to reach if it is based on a policy of openness with regard to the residual risks due to ionising radiation and the means of making it as low as reasonably achievable.

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PAPER 4-1
DISCUSSION

Andersen: You talked about the behavioral and cultural changes that were necessary to put this program in place. Now that you have had a few years of experience, are you getting much feedback from the work force? Is this being received favorably by the workers? Are they now motivated, or are you still having to work on that?

Rollin: Yes, of course. As I said at the beginning of my talk, people were not at all aware of ALARA. They did not ask themselves any questions. They were satisfied with the results. Then they began to see the improvements, especially after the Dampierre steam generator replacement. This presented an opportunity to spread the ALARA culture all over the EdF. We had meetings with all the persons responsible for the maintenance programs. At first, many of them were not convinced. When they saw the results, they became interested. They saw that ALARA programs were going on everywhere in the world, and they formed programs. This formation spread all over the company, and it has been interesting to see the progression of the idea. More and more people feel involved, and when they get involved, you are sure they are getting the point. It is very effective.

Burholt: Please explain the interface between the site ALARA committee and the conventional radiation protection services. Can you describe the management responsibilities?

Rollin: I mentioned the site ALARA committee and the site ALARA working group. In the site ALARA committee, the committee head is the deputy manager of the plant. You find in the ALARA committee all of the main departments of a nuclear plant. You find maintenance, chemical, operation, and, of course, the radioprotection personnel. They participate in all the jobs at the level of site ALARA committee. Then, they are better accepted by all the staff because they participate and they give advice on the job. If there is a special group set up to study a special problem, they also participate in that group. Perhaps, in France, the general management for radiation protection is different than in other countries. That means, for the moment, that the responsibility remains at the level of manager or deputy manager, and perhaps there is not enough authority for the radiation protection staff. We are considering giving the radiation protection staff more weight in decision making.

ASSESSMENT OF THE BENEFITS AND IMPACTS IN THE U.S. NUCLEAR POWER INDUSTRY OF HYPOTHESIZED LOWER OCCUPATIONAL DOSE LIMITS

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ABSTRACT

The International Commission on Radiological Protection and the National Council on Radiation Protection and Measurements have issued recommendations that would limit occupational exposure of individuals to doses lower than regulatory limits contained in the Nuclear Regulatory Commission's 10 CFR Part 20, "Standards for Protection Against Radiation." Because of this situation, there is interest in the potential benefits and impacts that would be associated with movement of the NRC regulatory limits toward the advisory bodies' recommendations.

The records of occupational worker doses in the U.S. commercial nuclear power industry show that the vast majority of these workers have doses that are significantly below the regulatory limit of 50 mSv (5 rem) per year. Some workers' doses do approach the limits, however. This is most common in the case of specially skilled workers, especially those with skills utilized in support of plant outage work. Any consideration of the potential benefits and impacts of hypothesized lower dose limits must address these workers as an important input to the overall assessment. There are also, of course, many other areas in which the benefits and impacts must be evaluated. To prepare to provide valid, constructive input on this matter, the U.S. nuclear power industry is undertaking an assessment, facilitated by the Nuclear Energy Institute (NEI), of the potential benefits and impacts at its facilities associated with hypothesized lower occupational dose limits. Some preliminary results available to date from this assessment are provided.

BACKGROUND

On May 21, 1991, the U.S. Nuclear Regulatory Commission (NRC) published the revised 10 CFR Part 20[1] that is based on the 1977 recommendations of the International Commission on Radiological Protection (ICRP)[2]. The revised Part 20 includes an occupational dose limit of 50 mSv (5 rem) per year. Implementation of the revised Part 20 was required of all NRC licensees as of January 1, 1994.

In 1991, the ICRP published new recommendations[3] that include limitation of occupational dose to 100 mSv (10 rem) in 5 years, not to exceed 50 mSv (5 rem) in a year. The National Council on Radiation Protection and Measurements (NCRP) published new recommendations[4] in 1993 that include limitation of an individual's cumulative lifetime dose to 10 mSv (1 rem) times the individual's age (in years), not to exceed 5 rem in a single year. The new recommendations would have the effect of limiting an individual's lifetime risk from occupational exposure to approximately 3×10^{-2} (NCRP) and 4×10^{-2} (ICRP).

In its issuance in 1991 of the revised Part 20, the NRC acknowledged the pending recommendations of the ICRP and NCRP that were not being addressed in the revised regulation. The NRC expressed its intent to consider these recommendations at a future date following final publication by the ICRP and NCRP. The NRC has requested that a preliminary study be made by the Brookhaven National Laboratory (BNL) to analyze the potential impacts of lower dose limits on NRC licensees. This could provide a portion of the technical bases for making future decisions on regulatory limits. The results of that study have been published in draft form for comment as NUREG/CR-6112, "Impact of Reduced Dose Limits on NRC Licensed Activities."[5]

The draft NUREG includes the results of a survey made by BNL of NRC licensees to assess potential impacts of lower occupational dose limits. The draft NUREG also includes an historical background on previous reductions in regulatory occupational dose limits, as well as a review of existing literature on recent occupational dose data and trends in various U.S. nuclear industries. As such, the draft NUREG provides the an up-to-date and comprehensive overview of the potential effects of hypothesized lower occupational dose limits

The NRC has published the draft NUREG and is soliciting further comments from interested parties regarding potential impacts of the different hypothesized lower occupational dose limits discussed in the draft NUREG.[6] In the draft NUREG, the NRC makes the assessment that a relatively small number of licensees responded to questionnaires and surveys, thereby limiting the extent to which the survey results can be assumed to be an accurate representation of the potential impacts of hypothesized lower occupational dose limits.

To support preparation of valid, constructive input to NRC on the matter of lower occupational dose limits, the U.S. nuclear power industry is undertaking an assessment, facilitated by the Nuclear Energy Institute (NEI), of the associated potential benefits and impacts at its facilities. Preliminary results available from this assessment are presented in this paper.

ISSUE

The issue being assessed by the nuclear power industry is what effects lower occupational dose limits consistent with NCRP or ICRP recommendations would produce, in terms of potential benefits and impacts that would ensue. To provide focus to the nuclear power industry's approach to the issue, it has been segmented into three questions as discussed below.

Will lower dose limits provide a substantial improvement to the protection of worker health and safety ?

Of primary importance in answering this question is to identify the population of workers that might be affected by lower dose limits and the nature of the effects on them. In particular, we are interested in the extent to which individual worker doses would be limited below the levels presently being experienced under the current regulatory dose limits and industry practices to maintain exposures as low as reasonably achievable (ALARA). We must also try to understand how continued employability of individuals may be affected by lower dose limits. We set out to determine this by reviewing available data and by surveying utilities that operate nuclear power facilities. In the survey we requested dose data and other information regarding workers whose 1993 annual dose exceeded 20 mSv (2 rem) or whose cumulative lifetime dose exceeded 0.25 Sv (25 rem) as of 1993 to characterize the affected worker population and discover trends.

The otherwise available records of occupational doses show that the vast majority of U.S. nuclear power plant workers receive doses that are significantly less than the current regulatory occupational dose limit of 50 mSv (5 rem) in a year.[7] For example, the data indicate that more than 99% of nuclear power plant workers monitored in 1990 received less than 2 rem annual dose, which is comparable to the average annual dose implied by the ICRP dose limitation recommendations. The data also indicate that average annual occupational doses have been generally declining.

Draft NUREG/CR-6112 provides data that indicate that only a small fraction of workers have received lifetime doses that exceed their age. Other available data show that few workers' lifetime doses approach or exceed the 70 rem or 100 rem values that are implied by the recommendations of the NCRP and ICRP.[8] However, the data are sparse and do not provide specifics on the types of workers involved or recent lifetime dose trends. Also, there does not appear to be published data regarding the doses of U.S. nuclear power plant workers for 5 year periods (i.e., comparable to the ICRP dose limitation recommendations).

Additional data are being sought to support a more detailed assessment of whether lower occupational dose limits, consistent with ICRP or NCRP dose limitation recommendations, would provide a substantial improvement to the protection of worker health and safety.

Will lower occupational dose limits result in an increase in collective dose ?

Lower occupational dose limits may result in the need to increase the number of workers, e.g., to address the situation of specific work groups that become constrained by the lower limits. Such situations may involve inefficiencies, such as the use of multiple workers to complete the task, that result in higher collective doses. Limiting individual doses must be considered relative to the potential for increases in collective dose. It is therefore necessary to characterize the specific work groups that may be constrained and the potential effects, including collective dose impacts, of addressing the constraints. Further dose reduction measures, consistent with the ALARA approach, would also be considered and will have an impact on collective dose.

Draft NUREG/CR-6112 concludes that "there would be minimal impact on collective doses ..." under the NCRP dose limitation model of 50 mSv (5 rem) per year and lifetime dose not to exceed age (in years), and that a "grandfather clause" allowing up to 20 mSv (2 rem) per year after exceeding the lifetime dose limit "... may be required for perhaps less than 1000 workers." However, the related survey responses tabulated in the draft NUREG appear somewhat ambiguous regarding potential increases in collective dose. None of the nuclear power industry respondents expected an increase in collective dose with an annual dose limit of 5 rem per year and lifetime dose limit of age in rem; at the same time, most of the nuclear power industry respondents expected collective dose to increase with an annual dose limit of 2 rem per year and a lifetime dose limit of age in rem, which is analogous to the use of the "grandfather clause" option. Also, available data regarding lifetime dose trends are not sufficient to determine whether a limit on lifetime dose would increase collective dose as a potentially increasing population of workers approach cumulative lifetime doses equal to their age.

To develop an answer regarding whether lower dose limits will result in larger collective doses, we surveyed to identify which work groups may be constrained by lower dose limits and asked whether the number of workers might be increased to accommodate the lower limits and whether related potential inefficiencies (e.g., due to multiple work crew changeouts) may lead to increased collective dose. We will also look at lifetime dose trends to determine, if possible, the extent of the potential population of workers that over time may approach cumulative lifetime doses equal to their age.

What degree of impacts on licensees will result from lower occupational dose limits ?

The degree of impacts resulting from lower occupational dose limits may vary substantially between nuclear power licensees due to differences in operating and maintenance history, source terms, and a number of other factors. This variation is apparent in the summary of survey responses from nuclear power licensees that are provided in draft NUREG/CR-6112. Cost estimates provided in the responses regarding potential facility modifications and changes to radiation protection programs to accommodate lower dose limits vary over several orders of magnitude. The data provide little insight into how the cost estimates were derived and what assumptions were used, which makes difficult any attempt to project the extent of potential impacts for the industry as a whole. Also, there may be other types of impacts that need to be considered, in addition to the potential for facility modifications or changes to radiation protection programs that are specified in the draft NUREG. Likewise, other available data[8,9] regarding potential impacts associated with lower dose limits are either too general or otherwise not directly applicable* to making projections of impacts on an industrywide basis.

To help answer the question of the extent of impacts on nuclear power plant licensees from lower occupational dose limits, in addition to obtaining data regarding specific work groups and tasks that may be impacted, we are surveying for additional related information, e.g., administrative dose guidelines and $/person-rem values used in ALARA cost benefit analysis, to support a more broad and detailed assessment.

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* For example, the Atomic Industrial Forum (AIF) "Study of the Effects of Reduced Occupational Radiation Exposure Limits on the Nuclear Power Industry" provides sophisticated methods for assessing impacts, but only considers various <u>quarterly</u> dose limits, and does not consider annual or lifetime dose limits.

APPROACH

NEI has formed an Ad Hoc Advisory Committee (AHAC) of radiation protection professionals from more than 20 nuclear power utilities and nuclear steam supply system (NSSS) vendors to assist in performing an in-depth assessment of the potential benefits and impacts of hypothesized lower occupational dose limits on the nuclear power industry. Two surveys have been developed and widely distributed within the nuclear power industry to obtain specific data regarding nuclear power industry worker dose trends and potential impacts. Specific data being requested in the surveys include the following:

1. Survey on Worker Dose Trends

    a. Workers whose lifetime dose exceeded 25 rem as of 1993 and
    b. Workers whose annual dose exceeded 2 rem in 1993:
        - Date of Birth
        - Work Group
        - Utility or Non-utility
        - Lifetime Dose
        - 1993 Annual Dose
        - Total Dose for 1988-92 (if available)

2. Survey on Potential Impacts

    a. Work Groups Potentially Impacted
    b. Jobs/Tasks Potentially Impacted
    c. Administrative Dose Guidelines
    d. $/Person-Rem Values for Cost-Benefit Analysis
    e. Major Dose Reduction Initiatives

The AHAC will assist NEI in assessing the survey results and developing a nuclear power industry perspective on potential benefits and impacts related to hypothesized lower occupational dose limits. Following broad review of the perspective by industry, input will be provided to the NRC in response to its request for comments on the draft NUREG.

PRELIMINARY RESULTS

To date, we have received 50% of the responses to the survey on worker dose trends. We have compiled the data in a statistical analysis database and have developed preliminary results to validate the approach taken to consider some of the questions related to worker doses. Selected preliminary data are provided below. Table 1 shows projected dose trends for nuclear power plant workers monitored in 1993. The numbers have been rounded to 2 significant figures and do not account for potentially redundant data for workers not directly employed by the utility (i.e., non-utility workers) who may have worked for and been reported by several respondents as being monitored in 1993. These data will be refined when the balance of responses have been received. Figures 1, 2, and 3 show the actual data (i.e., not projected) from the survey responses regarding the workers whose lifetime doses exceed their age sorted by work group, age, and dose range, respectively. Because the data are preliminary, specific conclusions are not yet presented. The survey on potential impacts was recently sent out to the industry with responses due back in several weeks, therefore, preliminary data from that survey are not yet available.

CONCLUSION

A survey of nuclear power industry worker dose information is warranted to assess what effects lower occupational dose limits consistent with NCRP or ICRP recommendations would produce. This is because the available data limit the detailed assessment appropriate to making decisions on whether to establish such regulatory requirements. The potential benefits and impacts important to that assessment can be defined by addressing the questions detailed above. NEI is pursuing data collection and its assessment in order to provide nuclear power industry perspectives to the NRC in response to its request for public input on this subject.

Table 1. U.S. nuclear power plant workers monitored in 1993 (projected from preliminary data)

Group	No. of Workers
Monitored - Total	180,000
Monitored - with Measurable Dose	100,000
Annual Dose > 2 rem	2,000
Lifetime Dose (rem) > Age (years)	700
• Annual Dose > 1 rem	100
• Annual Dose > 2 rem	25

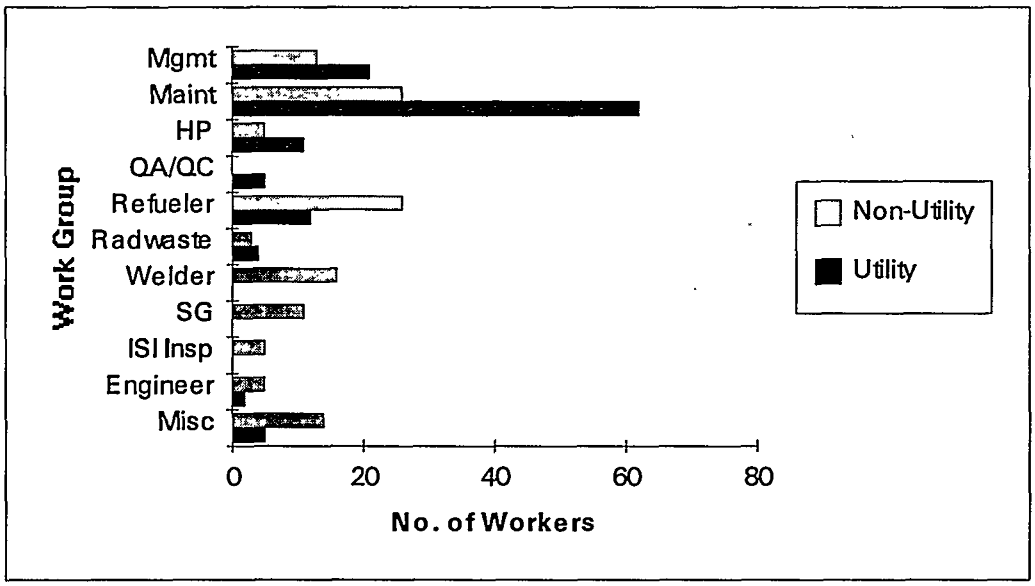

Figure 1. U. S. Nuclear power plant workers with lifetime dose > age sorted by work group

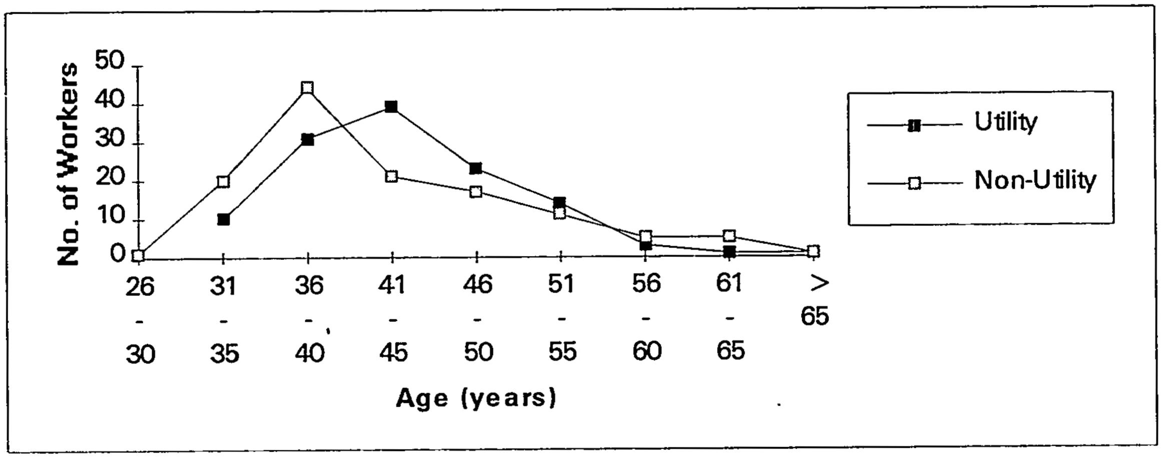

Figure 2. U.S. nuclear power plant workers with lifetime dose greater than age sorted by age group

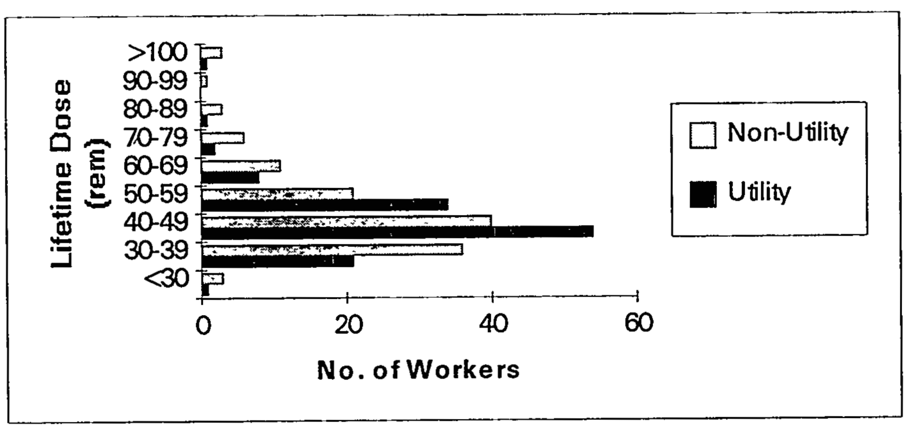

Figure 3. U. S. nuclear power plant workers with lifetime dose greater than age sorted by dose

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ALARA IN EUROPEAN NUCLEAR INSTALLATIONS

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ABSTRACT

For over a decade the Commission of the European Community has sponsored research projects on the development and practical implementation of the Optimisation principle, or as it is often referred to, ALARA. These projects have given rise to a series of successful international Optimisation training courses and have provided a significant input to the periodic European Seminars on Optimisation, the last one of which took place in April 1993. This paper reviews the approaches to Optimisation that have developed within Europe and describes the areas of work in the current project. The on-going CEC research project addresses the problem of ALARA and internal exposures, and tries to define procedures for ALARA implementation, taking account of the perception of the hazard as well as the levels of probability of exposure. The relationships between ALARA and work management, and ALARA and decommissioning of installations appear to be other fruitful research areas. Finally this paper introduces some software for using ALARA decision aiding techniques and databases containing feed back experience developed in Europe.

INTRODUCTION

Since the publication of ICRP 22 [1] and ICRP 26 [2] in 1973 and 1977 respectively, the understanding and practical implementation of the concept of Optimisation of Radiation Protection has developed considerably in Europe. This past progress can be split into three periods. The first period, lasting up to 1982 was mainly focused on theoretical aspects and an evaluation of possible quantitative decision aiding techniques, with most emphasis being placed on cost effectiveness and cost benefit analysis. The second period from 1982 to 1987 was mainly devoted to the development of a structured approach to optimisation, the ALARA Procedure, within which decision aiding techniques, if required, could be used. The period also saw many case studies being carried out in a wide variety of installations in relation to both design and operational problems, but predominantly a posteriori. The third period from 1988 onwards has seen the development of more structured approaches and "tools", which together with an a priori predictive approach are being integrated into operational radiological protection programmes. This evolution can be traced through the proceedings of the four European Seminars on Optimisation [3,4,5,6], the last of which was in Luxembourg in April 1993.

Staff from CEPN, France, and NRPB, United Kingdom, have been working on the practical implementation of ALARA for a number of years and some of the results of this work have been published in a book [7]. Much of this has been financially supported by the Commission of the European Community (CEC) within joint research projects. In 1993 two other organisations, SCK/CEN from Belgium and GRS from Germany joined the European research project on radiation protection optimisation in installations. During the last four years CEPN and NRPB have also been heavily involved in running training courses on optimisation. One of the strengths of these events has been the input provided by lecturers from utilities in the UK, France and Sweden who have experience in implementing ALARA in the nuclear industry. Another strength has been the wealth of practical experience that the participants themselves, from a range of countries and different backgrounds, have been able to bring to these courses. Many of the perspectives on optimisation of radiological protection in Europe given in this paper have developed out of these research and training programmes.

I - EUROPEAN ALARA PROGRAMMES: 1994 STATE OF THE ART

We have always taken the ALARA principle to apply to both individual and collective exposures. The principle as now stated in ICRP60[8] explicitly covered this but also emphasised the need to focus on individual exposures. To satisfy this principle it is clear that one should not just pursue control of individual doses relative to limits or targets; one has to implement an ALARA approach i.e. an a priori management of both individual and collective exposures. This means that radiation protection has to be integrated into the global management of organisations. The ALARA approach may then be characterised with some key words such as "prediction, efficiency, and equity ".

In most of our countries management of the radiological risk is now a feature of the operational and maintenance phases of nuclear installations through implementation of the so called ALARA Programme. The programme fits with the three phases of any project (see figure 1): setting dosimetric objectives during the preparation phase, following up the dose results during operation and analysing the feed back to improve the next operation.

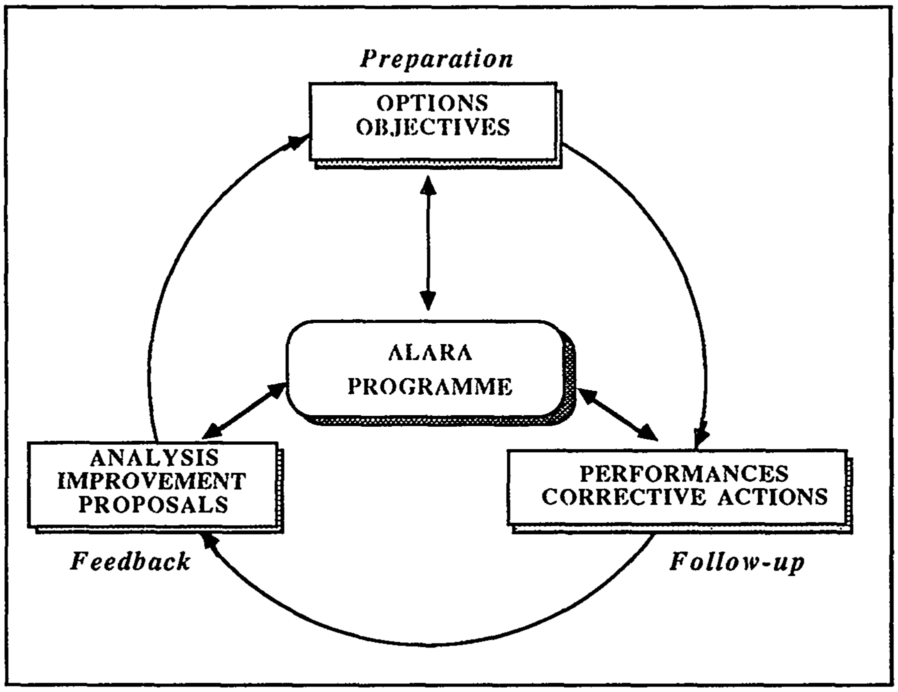

Figure 1. ALARA Programme and phases of any operation, outage or year

The main characteristics of such programmes are the commitment of all the "actors" from the regulators to the workers, the appropriateness of the decisional structures to ALARA implementation and the use of adapted "tools" such as the monetary value of the manSv (often referred to as the alpha value). The first part of this presentation does not aim to provide an exhaustive description of the European situation but will address some specific points of interest within the European context.

Regulatory Arrangements

The ALARA principle has been progressively incorporated into most European national regulations (see table 1) [9], where in most cases it appears as a top level general requirement or objective. Major exceptions are Germany and Portugal where the regulations still require the minimisation of doses i.e. as low as possible.

Table 1. ALARA and National regulations in Europe

Countries	Date and reference text	Wording of optimisation
Belgium	Royal Decree, 25/04/1987	
Ireland	Statutory Instruments 43, 1991	
Luxembourg	Grand-ducal Regulation, 29/10/1990	"...as low as reasonably possible"
Netherlands	Decree, 10/09/1986	
Spain	Royal Decree 53/1992, 24/01/1992	
Denmark	Regulation 383, 1986	"as low as reasonably achievable"
United Kingdom	Regulations 1985	"restrict so far as reasonably practicable"
Greece	Decree, 19/07/1991	"as low as reasonably achievable technological feasibilities, results of cost-benefit analysis and in general every other social and economic factor being taken into account"
Germany	Ordinance, 30/06/1989	"...as low as possible, taking due account of the state of the art and paying attention to the merits of each individual case"
France	Decree, 02/10/1986 modified 1988 and Decree, 28/04/1975 modified 1988 (occupational radiation protection)	"...as low as reasonably possible"
Italy	President Decree, 13/02/1964, never revised	"...to reduce workers exposures, taking into account good current practice"
Portugal	Regulatory-Decree 9/90, 19/04/1990	"...as low as possible"

Different views exist in the various countries on the extent to which this requirement should be expanded into precise prescriptive regulations. Both the form of the ALARA principle in national regulations, and the will of the Authorities to enforce its implementation appear to have a very important impact on the radiological protection culture in the different countries and on organisations' and individuals' perceptions of what ALARA means. This point has repeatedly emerged from the CEC courses on optimisation. In countries like France and Belgium the Authorities tend not to intervene, while in Sweden or Spain the Authorities require any utility to provide collective or individual dose predictions per important job, discuss with the utility the possible protection actions to optimise the exposures, and check the results against the predictions. In the United Kingdom, the authorities use the general regulatory requirement to underpin improvements they require, and also specify levels of individual dose, which if exceeded, require the employer to carry out an investigation to determine if appropriate action had been taken to keep doses as lows as reasonably practicable. Another important point is the extent to which the ALARA requirement has been tested in a court of law. At present ALARA requirements have been addressed in the law courts of two European countries. The first is the UK, where 'reasonably achievable' is replaced by "reasonably practicable", a term that has been used for many decades in a wide variety of safety legislation. As a result there are case precedents that can be used in a court of law, and several convictions in respect of failure to meet the ALARA requirement have been recorded [10, 11]. The second country is France where, in 1993, for the first time [12], a court convicted a manager on the grounds of not meeting the ALARA principle. This case is currently the subject of an appeal.

Management Commitment, Workers Motivation and Training

It is now obvious that real success in the application of ALARA demands that organisations take a more positive role than only responding to regulatory pressure [13, 14]. A strong management commitment, through for example Corporate Codes of Practice, is as fundamental as the commitment of individuals at all levels within the organisation. It is therefore important for each organisation, to ensure that ALARA is totally inserted within its culture as a "way of thinking", through its various components such as training, information, communications and incentives. At this point it would be appropriate to identify questions that, as yet, have not received a consensus view in Europe. Do rewards and incentive schemes have a role to play in optimisation, and if so in what form ? Are there problems associated with the use of such schemes, and if so what are the solutions to these problems ?

Contractors Involvement

For Light Water Reactors, more than 80% [15] of the collective dose is received by contractors' employees, and it is therefore impossible to achieve ALARA without effective and efficient cooperation from the contractors. As a result more and more frequently' European utilities are introducing ALARA oriented requirements into contractual arrangements e.g., dose prediction, dosimetric goals, radiation protection feedback reports etc. They also analyse the contractor's proposals concerning the development of "tools" and process modifications with respect to dose savings and their corporate value of the man sievert. In Sweden, at Vattenfall, the corporate man sievert value is specified in the contract; it is then mandatory for the utility to accept any contractor's proposal which leads to dose savings costing less than this man sievert value.

The standard of radiation protection shown by contractors is increasingly becoming an important part of their ability to compete and win contracts. This is to be welcomed, however it is not without problems. For example the retention of intellectual property rights by contractors for their expertise in processes and the development of specialised "tools", can inhibit the dissemination of feed back experience.

Organisational Structures

A key element of management's contribution to ALARA is having an organisational structure capable of ensuring that ALARA is implemented. Whilst many different approaches are no doubt possible, it is worth considering two distinct approaches which have each been shown in Europe to be capable of applying ALARA in the workplace.

In the first approach, ALARA is accepted as an integral part of the overall radiological protection programme, and normal existing management structures are sufficient. The operational (or project) management team carries the formal responsibility for all aspects of safety, and the established culture of the organisation naturally extends this to encompass ALARA. In such cases the existing Health Physics organisation is likely to be effectively integrated into the overall management framework and to carry significant influence. This type of arrangement is, for example, the situation generally pertaining in the UK, Sweden and Finland.

A second approach is to create specific ALARA structures to provide an effective focus for pursuing ALARA. This could for example involve a special management ALARA Committee, with objectives such as setting targets (e.g., collective dose goals), taking strategic decisions on the impact of radiological protection actions on costs and production, and arbitrating in conflicts between designers, health physicists, engineers and operators. This committee could be supported by the appointment of a specified individual as ALARA Coordinator with responsibilities for the implementation of all aspects of the ALARA programme during the operation. This type of approach has been used with considerable success in some parts of the French, Spanish and Belgian nuclear industries. Nevertheless this leads to the following question. Is the use of ALARA structures only a temporary step on the way to integrating ALARA into the overall radiation protection programme and structure, or can both approaches complement each other? Irrespective of which approach is taken, the involvement of Health Physicists in the early stages of projects is strongly advocated. This can be justified in pure business terms in that it maximises the effectiveness of the investment of effort.

Adapted "Tools" and Procedures

Whilst ALARA success is mainly due to "Attitude", the use of adapted specific "tools" and procedures can be very helpful. In line with any ALARA programme these "tools" and procedures have to correspond to different functions dealing with the three phases of any operation (see figures 2 & 3). Most of these "tools" such as decision aiding techniques, particularly cost benefit analysis (CBA), or pre-job and post-job ALARA reviews together with corresponding check-lists, are now in current use, both in Europe and America. Special attention will be paid here to the "ALARA procedure", the status and levels of alpha values in the different European countries, the analytical "tools" and, finally, networks for exchanges of feedback experiences between utilities.

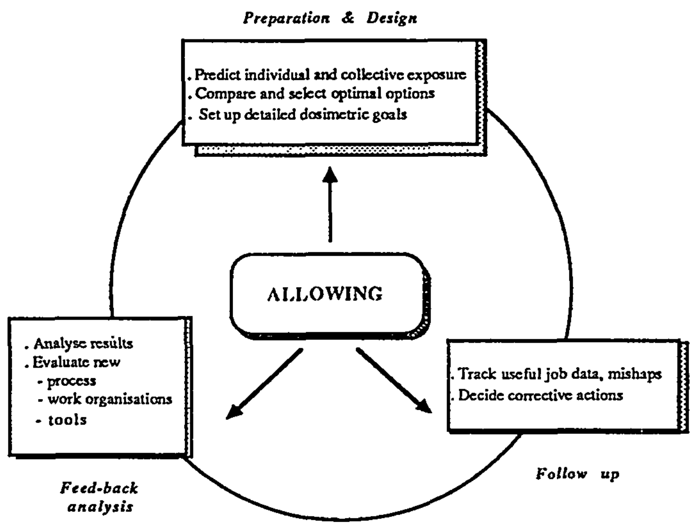

Figure 2. Functions of ALARA "tools" with regard to the phases of operations

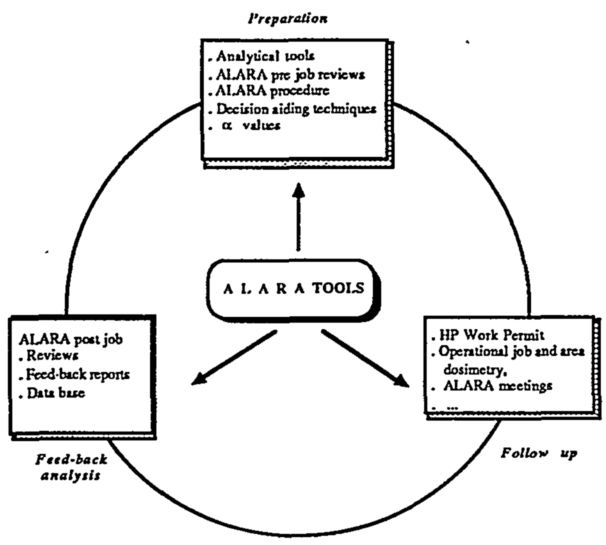

Figure 3. ALARA "tools" and phases of operations

The ALARA Procedure

So that ALARA decisions can be made in a systematic fashion, the ALARA Procedure was developed, and subsequently incorporated into ICRP publication 55 [16] on Optimisation. Its function is to provide a way of structuring and standardising judgements. It is stressed that this is only a schematic representation of a logical approach to clear decision-making, and as such it is also a representation of what many experienced health physicists already do in practice.

The keys steps in the ALARA Procedure [7] are as follows: to define the problem fully at the outset, setting boundaries to the analysis; to identify alternative courses of action (options) and the important factors in terms of doses and costs; to quantify, where necessary; and to make some comparison of the options identified. At this point a quantitative decision-aiding technique may be of use, but that will depend on the problem. Sensitivity analysis may or may not be required depending on the nature of the problem. The product of the Procedure is the ALARA result. However, this may not be the same as the final decision, because the decision-maker quite legitimately may conclude that other factors not directly considered in the analysis are also important and need to be taken into account. However, the procedure should ensure that all the radiological protection factors that are considered important are explicitly included in the study. This helps to make decision-making, and the rationale behind final preferences, more transparent.

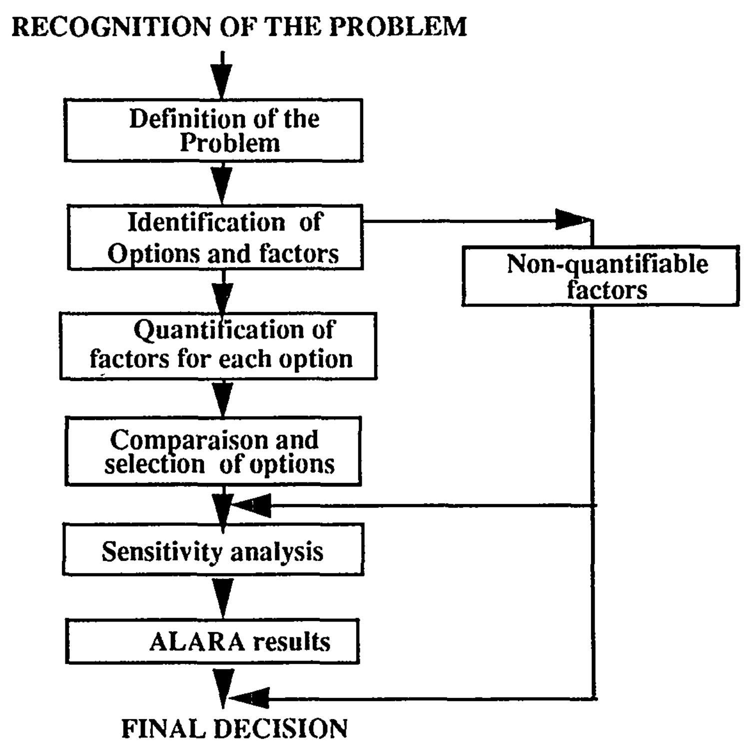

Figure 4. The ALARA Procedure steps

Alpha Value Status in European Countries

In order to assess what are reasonably achievable radiation protection options from an economic point of view a monetary valuation of unit collective dose (the cost of the man sievert) is obviously essential. This is often referred to as the 'alpha' value and has been addressed in many ICRP publications [1,8,17,18]. In the European countries, there are many different man sievert values corresponding to different rationales.

In a few countries, such as the UK and the Nordic countries (Denmark, Finland, Iceland, Norway and Sweden), national organisations have recommended values of the man sievert. For example in 1991 the radiation protection authorities of the Nordic countries recommended [19] value equivalent to approximately US $ 100k per man sievert. This single value was deemed suitable for all radiological situations and types of exposure ie, public and occupational in the nuclear, medical and industrial fields. In the UK, the NRPB, using its previous model [20], has recommended a base line value of £ 10k (US $ 15k) per man sievert together with a set of values for different radiological situations [21]. For example for occupational exposure the NRPB has taken into account the average annual levels of individual exposure of a few millisievert and using a multiplier to integrate the aversion to risk, recommended a minimum value of £ 50k (US $ 75k) per man sievert.

In the countries with recommended values, as well as in other countries, many utilities have set up their own corporate alpha values. It is noticeable that these corporate values have generally been several times higher than the nationally recommended ones. Some utilities have adopted a value covering all doses ranges. An example is the case of British Nuclear Fuels in the UK where a £ 100k (US $ 150k) per man sievert was used for a long term refurbishment project [22]. Another case is that of Vattenfall in Sweden, where a baseline value of SEK 4000k (US $ 700k) per man sievert [23] is used internally and contractually ie, if for a particular operation: a contractor proposes a radiological protection improvement costing less than this value, Vattenfall is contractually obliged to implement that improvement. In France, Electricite de France (EdF) has adopted a set of values rather than a single value. The valuations adopted are based on a model developed by CEPN [24]. There are five values ranging from FF 100k (US $ 18k) per man sievert for individuals with annual doses lower than 1 mSv to FF 15 000 k (US $ 2 700 k) per man sievert value for individuals with annual doses between 30 and 50 mSv.

What is clear from the above is that national and corporate values of the man sievert are now common and well founded. Variations in values are to be expected and they reflect the factors folded in, the social and economic pressures and available resources, together with the dose distributions pertinent to the defined exposed population they are applied to.

Analytical Softwares

In order to facilitate its use, particularly by non specialists, most of the ALARA procedure steps may be formalised in a computerised way. One example is DOSIANA [25] which was developed to assess and analyse predicted doses, as well as real dosimetric data. More recently a user friendly software to carry out cost effectiveness/cost benefit analysis (OPTI-RP) has been developed by CEPN and NRPB within the CEC joint project. This uses `ACCESS' under `Windows' , can accommodate both single and sets of alpha values, can accommodate temporal distributions of costs and can cope with a great number of options. It is totally independent of both national and computer contexts and will be available by the end of 1994 in two versions: one in English and one in French.

Feed back Databases and Networks

Both the utilities and authorities in Europe have put considerable effort into improving feedback exchanges between all the "actors" in the nuclear field in order to achieve as quickly as possible the necessary input to the ALARA management of the radiological risk. For about ten years, the Commission of the European Communities has set up a data base and an annual European meeting of utilities'representatives, in order to facilitate exchanges of feed-back experience in the radiation protection field and to analyse the evolution and levels of collective doses and individual doses distributions. In 1992, the European countries became part of the OECD-NEA Information System on Occupational Exposure (ISOE). By 1994 all 112 European light water reactors are participating to the ISOE system, providing it both with statistical data and job related informations (good practices, radiation protection problems) and using it as an operational network to ask questions, and speed up feed back experience retrieval.

II - ON GOING RESEARCH

Despite advances in the practical implementation of ALARA, there is still significant potential for improvement. In this part of the paper we will address four topic areas that are the focus for on going research, namely work management, decommissioning, internal exposures and potential exposures.

ALARA and work management

The application of the ALARA principle to occupational exposure implies the adoption of an analytical approach in order to identify the relevant factors contributing to individual and collective exposures. In Europe over the last two decades much has been done to reduce the ambient dose rates. However there is considerable evidence that much remains to be done to reduce the duration of exposure and the number of exposed workers required to carry out particular tasks. All procedures and actions which can influence these last two factors come under the heading of 'Work Management' [26].

In a recent study, CEPN has looked at three different categories of "work management factors": those linked to working conditions (ergonomics of work areas, protective suits...), those characterising the operators (qualification, experience level, motivation...), and those directly dependent on the organisation of operations (tasks planning, general preparation of work...). In order to quantify the impact of different working condition parameters, a detailed survey was carried out in five French nuclear power plants, and was supplemented by a literature review on the influence of "hostile" environments on working conditions. Also tests were carried out to quantify the impact of various types of protective suits used in French nuclear installations on a variety of types of work. All these factors have been included in a model aimed at quantifying the effectiveness of protection actions, from both dosimetric and economic points of views. The main results of this study will be presented during this workshop [27].

The direct impact of the approach to organising operations is more difficult to quantify. Nevertheless, some studies on causes of mishaps occurring during outage maintenance jobs in French NPPs has shown that up to 30% of the doses from mishaps can be attributed to organisation problems (planning, scheduling ...). A study of the organisation of outages in 4 different nuclear power plants from various countries has confirmed [28] the importance of commitment and motivation toward the ALARA principle. Also it highlighted the value of the total integration of radiation protection criteria in the overall outage process, from planning stage to feed back experience, through both effective co-ordination and collaboration of all groups involved in the outage, and an effective use of well documented feed back data bases for jobs, doses, dose rates, mishaps etc.

ALARA and Decommissioning

ALARA thinking is well on the way to pervading many aspects of the operational and maintenance phases in the life of nuclear installations. The extension of this approach to the whole life of an installation, although perceived as essential is not as well developed in Europe. During the next few decades, many nuclear facilities will be decommissioned and possibly dismantled. Decommissioning and dismantling operations have their own specific problems, differing in various aspects from normal maintenance operations in nuclear reactors. As a consequence radiological protection optimisation with respect to decommissioning/dismantling is an area warranting further attention in various aspects.

These aspects are mainly related to:

- the decommissioning strategy;
- the decommissioning methodology;
- the dismantling operations;
- the management of the radioactive waste generated during the decommissioning.

The decommissioning strategy defines the major milestones in the decommissioning process as a function of time. Conventional endpoints are:

- evacuation of free activity and confinement of that remaining;
- dismantling of the most contaminated structures and confinement in a reduced volume;
- evacuation of all radioactive materials and complete restoration of the site.

Economical, technical and radiological arguments feed into decisions on the scale and timing of each step. The economic and radiological attributes relate to the costs and doses associated with survey, maintenance and finally the dismantling operations. Also economic benefits, e.g. from the restoration of the site, and radiological risks from uncontrolled intrusion and from degradation of barriers as well as radiological impacts of waste management options play a role. The values of these attributes depend strongly on time, and consideration has to be given to the capitalisation of monetary provisions, to the degradation of equipment and the loss of human know-how with time. SCK/CEN is currently examining the variation with time of the relevant attributes involved in ALARA decision making concerning the decommissioning strategy. Important trade-offs playing a role in the decision-making, have been identified eg:

- the use of a telemetric monitoring system (exposure during the installation) versus local survey (exposure during monitoring);
- decontamination prior to dismantling (doses during the decontamination) versus no prior decontamination (higher doses during dismantling);
- the costs, doses and equipment reliability related to the use of robots.

Dismantling can be broken down into the major tasks of isolation, cutting, transport, interim storage and clean up. A serious lack of experience is evident, particularly with respect to the technological aspects and radiological protection factors of cutting operations. As a consequence a data base collecting experience from dismantling operations, is considered as a very valuable aid for future optimisation analyses. SCK/CEN has developed a draft-structure for such a data base and has demonstrated its applicability with respect to dismantling operations at its BR-3 reactor. In routine maintenance activities, external gamma exposure is the major radiological attribute, however in dismantling operations, skin doses, internal contamination and waste generation (quality and quantity) are also important and can often dominate the decisionmaking. With respect to waste generation it should be noted that the decommissioning of an installation produces approximately as much radioactive waste as the amount generated during the total operational life of the installation. Therefore it is recommended that decommissioning and subsequent waste management be considered as a whole in the optimisation process, resolving conflicts between the nature of the waste generated by the best decommissioning option and the requirements on radioactive waste related to the best waste handling and storage option.

ALARA and Intake

Much of the thinking in optimisation has tended to focus on external exposure. Although the principle of optimisation clearly applies to all forms of exposure, it is often found in practice that internal exposure is treated in a very different manner to that for external exposure; with the approach being nearer to minimisation. This stems from two principal factors.

Firstly unlike external exposure it is often quite difficult to predict the levels of intake and hence the doses; because so many variables come into play. The problem is compounded by the difficulties encountered for many radionuclides in accurately measuring intakes that have occurred. This is further exacerbated for low ALI radionuclides, principally those of the actinides, as almost any measurable intake gives rise to a significant fraction of the dose limit. Thus we are often faced with the trade off of the certainty of increased external exposures, of the order of a few mSv per task due to loss of efficiency and manipulative skills from wearing protective clothing against the potential, albeit often less frequently than once in a decade, to receive an ALI or so if protection measures or procedures fail. Secondly feedback from the CEC training courses indicates that workforces appear to be significantly more averse to receiving an intake giving a committed effective dose of say 1 mSv, than for the same equivalent dose from external irradiation.

Often the costs of protecting against intakes are significantly greater than for comparable external irradiation protection measures. Thus the same resources if applied to external exposure would be more efficient in reducing the overall risks. To develop thinking in this area guidance is being prepared by NRPB which takes the basic structured approach of the ALARA Procedure and addresses the special considerations that apply to internal exposures e.g.. the exposure pathways, the protection options available, trade-offs between chronic routine exposures and probabilistic exposures, quantification of costs and intakes etc. Once the draft guidance has been completed, case studies will be carried out for a variety of uses e.g.. large scale handling of Low Specific Activity material, radiochemical laboratory design, work in nuclear power plants, decommissioning work, use of glove boxes, etc.

From discussion sessions at the 4th European Seminar [6] and the CEC Training Courses it appears that workforce perceptions and management decisions are based almost entirely on worst case predictions with inherent large pessimistic assumptions. Therefore improvements in applying ALARA will be dependent on stimulating better measurement regimes or means of assessing intakes.

ALARA and Potential Exposure

One of the major changes brought about by ICRP 60 [8] was the introduction of an explicit reference to potential exposures in the system of radiological protection. Subsequently, ICRP 64 [29] (and an imminent INSAG publication) have provided the conceptual basis for this topic. Any situation involving potential exposure will present a range of scenarios, each being characterised by a distribution of individual exposures, a collective exposure and a probability of occurrence. This topic will be further addressed during this workshop [30].

Theoretically the optimisation process will apply to situations where potential individual exposures above the normal operational dose limits might be encountered. Indeed one may have to consider deterministic effects (albeit with a very low probability) as well as stochastic effects. The optimisation process will require the ability to compare options where it will be possible to modify the distribution(s) of individual exposures for one or several scenarios and consequently the number and types of expected effects and/or to modify the probabilities of occurrence of the scenarios. To address this, firstly there will be a need to develop a means of weighting the different types of effects into a kind of unique index of harm in order to be able to compare the consequences of the different options independently of their probabilities. It will then be necessary to develop other tools to take into account the perception or utility of probability-consequences pairs and of their modification. Then and only then will it be possible to introduce optimisation of potential exposure in a structured way within a decision making process. It has also to be pointed out that regulators may need to define the boundaries of any optimisation process dealing with potential exposure.

It seems likely that the increased emphasis on potential exposures, will push managements into more explicit consideration of probabilistic events, which in its own right may lead to operational improvements. The development of thinking in this area may also have an impact on the previously mentioned problem area of ALARA and intakes.

CONCLUSION

The last few years have seen the spread and establishment of an ALARA culture within many European countries and organisations. This is reflected in organisational arrangements; attitudes of workforces, managements and regulators; databases for accessing past experience and most importantly work efficiency coupled with lower dose distributions. A number of papers at this meeting provide examples.

However it needs to be noted that to date the principal focus has been routine operations involving mainly external exposure. There is still an ongoing need to develop thinking in respect of the special problems posed by decommissioning and work involving intakes and potential exposures.

ACKNOWLEDGEMENTS

The work presented in this paper was carried out during a Joint Research project partly funded by the Commission of the European Community DG XII.

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SIX STEPS TO A SUCCESSFUL DOSE-REDUCTION STRATEGY

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ABSTRACT

The increased importance of demonstrating achievement of the ALARA principle has helped produce a proliferation of dose-reduction ideas. Across a company there may be many dose-reduction items being pursued in a variety of areas. However, companies have a limited amount of resource and, therefore, to ensure funding is directed to those items which will produce the most benefit and that all areas apply a common policy, requires the presence of a dose-reduction strategy.

Six steps were identified in formulating the dose-reduction strategy for Rolls-Royce and Associates (RRA):

1. collating the ideas,
2. quantitatively evaluating them on a common basis,
3. prioritising the ideas in terms of cost benefit,
4. implementation of the highest priority items,
5. monitoring their success,
6. periodically reviewing the strategy.

Inherent in producing the dose-reduction strategy has been a comprehensive dose database and the RRA-developed dose management computer code DOMAIN, which allows prediction of dose rates and dose. The database enabled high task dose items to be identified, assisted in evaluating dose benefits, and monitored dose trends once items had been implemented. The DOMAIN code was used both in quantifying some of the project dose benefits and its results, such as dose contours, used in some of the dose-reduction items themselves.

In all, over fifty dose-reduction items were evaluated in the strategy process and the items which will give greatest benefit are being implemented.

The strategy has been successful in giving renewed impetus and direction to dose-reduction management.

INTRODUCTION

Why have a dose-reduction strategy? Several years after the widespread implementation of ALARA, most major dose-reduction activities are being applied, and there is a plethora of proposals which will make smaller, but still significant, reductions in accrued dose. Although individual modifications or improvements to plants are assessed and an ALARA decision made, these may have knock-on effects to the dose benefit to be gained from proposals in other areas. There is the need to direct expenditure to those items which will produce the most overall benefit and to ensure a common policy across all areas of design, operation and maintenance. In addition, there is a need to take a long-term view on dose-reduction activities to ensure that dose will continue to be driven down in the future in order to meet the continuing downward trend on what is considered acceptable. This approach led to the formulation of a dose-reduction strategy at Rolls-Royce & Associates, consisting of a prioritised compilation of dose-reduction proposals across all areas with recommendations for implementation. Six steps were identified during the creation and implementation of the strategy and these are described below.

THE STRATEGY

Step 1 - Collating the Ideas

Creating a list of dose-reduction ideas to be considered for implementation was achieved by a variety of techniques. A literature search was carried out resulting in over 1,800 abstracts which were reviewed to identify, in particular, any dose-reduction items that had not previously been considered, as well as to confirm that all the major techniques were already being pursued. As a result of this work, some practices that had not previously been investigated, such as the injection of zinc ions into the reactor primary coolant and the anodic oxidation of the primary pipework (both of which may reduce cobalt deposition), were added to the ideas list.

The majority of the ideas came from various "brainstorming" sessions held between the staff involved such as operators, maintainers, designers, and Health Physics. These sessions took advantage of their experience and knowledge of the reactor plants and provided a forum for all ideas to be considered. The ideas ranged from detailed proposals such as greater use of mechanical couplings as opposed to welded, and relocation or redesign of various plant items to improve ease of maintenance, to more general suggestions such that workers should clean up as they go along.

In all, approximately 80 ideas were generated from the literature search, group sessions, and items previously proposed as part of the normal on-going improvement of the plants. The ideas were distilled down to 60 firm dose-reduction proposals. Group sessions were again used to segregate the list into three sections based on whether they were expected to produce a high, medium or low dose saving (unquantified at this stage).

Step 2 - Quantitatively Evaluate on a Common Basis

The dose-management system at Rolls-Royce & Associates incorporates a comprehensive database recording both measured doses from the various reactor plants as well as predicted doses generated by the ALARA engineers (see Figure 1). The information held on the database can be analysed and displayed to provide, for example, individual dose, task dose, worker group dose, comparisons of measured and predicted dose, etc. One of the predictive tools used is the RRA-developed computer code DOMAIN, which calculates dose rates and task doses from the reactor plant or indeed any physical structure containing gamma activity. DOMAIN can also be used to investigate the effect of various options such as installing extra shielding, draining components or decontaminating the activity.

From the database information the measured task or worker group dose related to each dose-reduction item was determined and used to estimate the potential dose saving resulting from its implementation. Some of the proposals will affect more than one task and, therefore, the dose savings from several tasks have to be combined, e.g., extending the use of metallic lagging on the primary plant will have a knock-on effect in reducing the time required for cleaning and inspection as well as reducing the dose for removing and replacing the conventional lagging.

For items involving a change in the actual dose rate environment (as opposed to a reduction in occupancy time, where simple calculations are generally adequate), the DOMAIN code was used to model the situation and calculate the dose with and without the change, e.g., it was used to calculate the effect on various maintenance tasks of adding temporary shielding around components.

For the majority of dose-reduction proposals, the cost penalty associated with implementation has been estimated from knowledge of the costs incurred with similar tasks. For those items with the greatest potential for dose saving, a more rigorous costing exercise was applied. Some items of relatively low dose saving were difficult to quantify in terms of cost and for these cases the value of unit dose, in UK terms the £/Sv value, was used to determine what cost could be reasonably justified against the estimated dose saving.

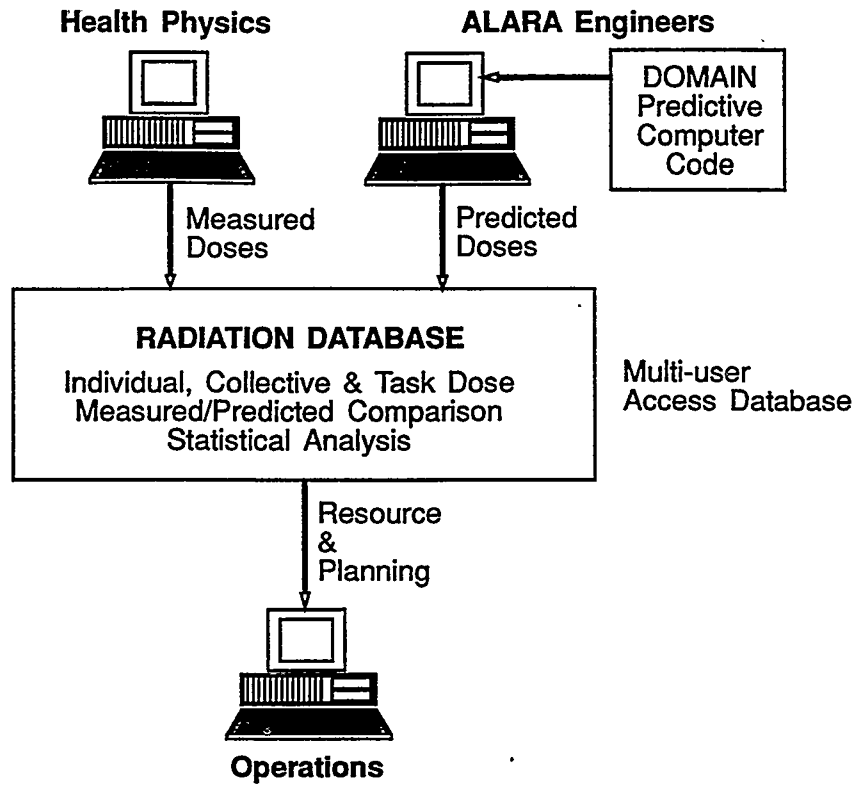

Figure 1. Dose Management System

Step 3 - Prioritise in Terms of Cost Benefit

Having determined dose savings and cost penalties, the proposals were ranked. The top ten items, listed in Table 1, were recommended for implementation during this first phase of the dose-reduction strategy.

Table 1. Top Ten Dose-Reduction Proposals

	Proposal	Dose Saving (man-Sv)	£000/Sv Value
1.	Extend application of metallic lagging	8	25
2.	Set dose targets for all tasks	5	6
3.	Revise painting requirements	2	10
4.	Install CCTV for inspection and general monitoring	2	40
5.	Reduce frequency of Health Physics manual inspections	1	10
6.	Reduce number and length of electrical maintenance procedures	0.6	35
7.	Provide permanent supports for erection of temporary shielding	0.5	200
8.	Extend application of automated equipment for NDE	0.5	200
9.	Label or color-code areas for dose	0.25	200
10.	Use computer simulated models as training aids	0.5	400

The latest recommended value of the £/Sv by the UK National Radiological Protection Board (NRPB) is that it would be reasonable to spend up to £50,000 per man-Sv of dose avoided based on the assumed health risk. If benefits from dose reduction such as worker reassurance, public perception and good public relations are taken into account, then expenditure of £100,000 per man-Sv avoided has been readily accepted previously at RRA. The positions of the dose proposals in Table 1 on a cost benefit vs dose saving graph are illustrated in Figure 2.

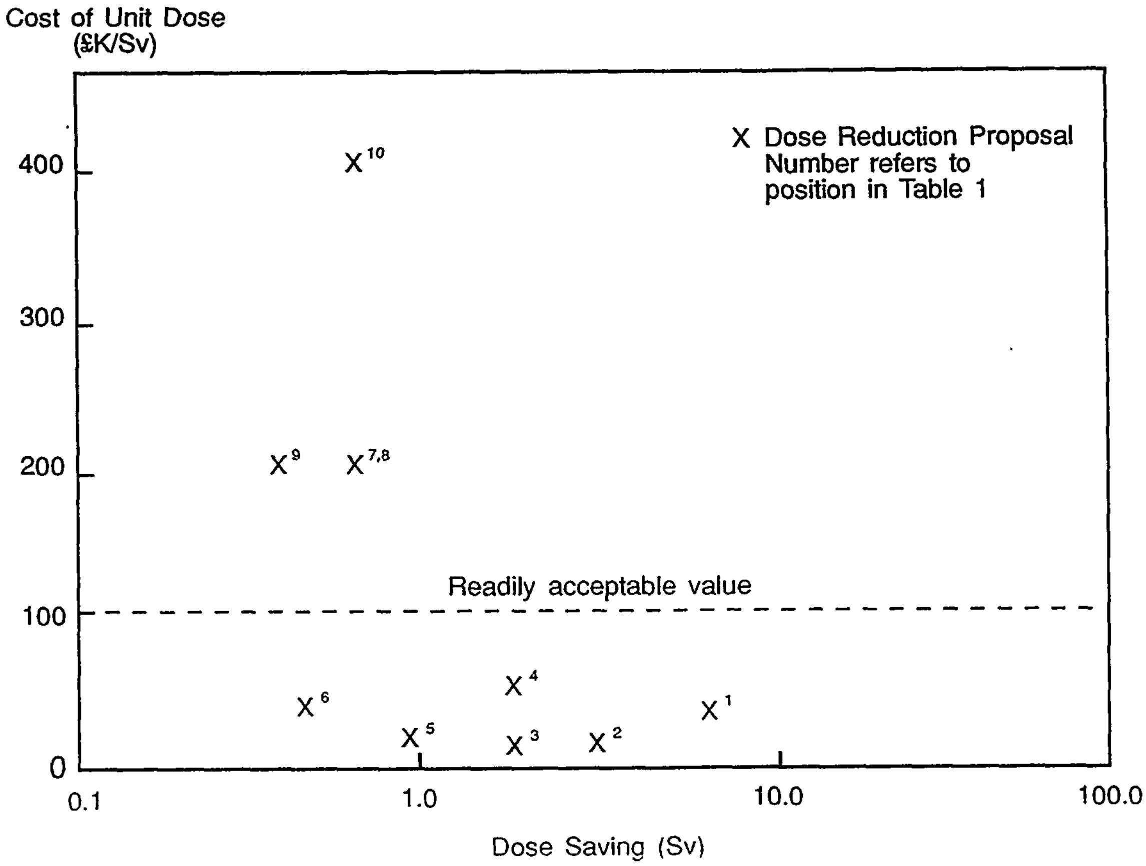

Figure 2. Location of top ten proposals in cost benefit terms

The figure shows that the top six items in terms of dose saving all lie below the £100K/Sv value. The four items above the "acceptable" £/Sv value, although producing a relatively low dose saving were either low cost items or were extensions to dose-reduction items that were common practice. For example, on item 7, temporary shielding is already being used extensively during maintenance and, therefore, the provision of permanent shielding supports was still judged worthwhile.

It is also noticeable that there are no high dose saving (>10 Sv) items on Figure 2. These would be measures such as decontamination which save several tens of Sv of dose and which are already in use. The implementation of these high dose saving measures has often had knock-on effects to other tasks (decontamination reduces the dose on nearly all tasks), and therefore, the available dose saving for future measures is reduced. The dose-reduction measures further down the priority list would all appear on the extreme left-hand side of the graph, with the majority of them some way above the £100K/Sv level. Careful consideration will need to be given as to whether their implementation can be justified.

Estimates of the total benefit to be gained indicated that implementing the top ten items would produce a dose saving of 20 man-Sv at an overall cost of £0.8M (an average valve £40,000 per man-Sv avoided).

Step 4 - Implementation of the Highest Priority Items

The dose-reduction strategy has been regarded as a single project as regards progress monitoring, even though the individual items are carried out by different areas, and this has ensured a high visibility as the overall strategy requires significant funding. The top ten items are currently in various states of implementation, e.g., new equipment is being tested that will carry out some electrical maintenance remotely (item 6); closed circuit television (CCTV - item 4) is being used for a variety of tasks such as general work area viewing, monitoring the performance of automated machinery and remotely reading instrumentation; and the use of automated equipment for painting is being investigated (item 2). Development of items 9 and 10 (color coding of areas and use of computer simulated models) has involved the use of the dose prediction code DOMAIN. From a set of measured dose rates around the reactor plant DOMAIN can calculate the source activity and hence determine dose rates anywhere around the plant. This capability is being used to generate dose rate contour maps both to identify low dose rate "awaiting" areas in the reactor plant during maintenance, and to combine with virtual reality models of the plant to enable display of the dose rate and accumulated dose during simulated walk-throughs of the plant for training purposes.

Step 5 - Monitoring Success

Just as important as the generation of ideas and their implementation, is to demonstrate how effective the measures are proving to be. This will give confidence in the strategy and help to provide realistic goals. One of the most useful techniques we have found to monitor dose reduction achievement is the "task learning curve." This involves compiling the measured dose from the radiological database for each task or worker group which has been subject to a dose-reduction measure, normalising this dose to account for different dose rate environments or other variations, and recording the data in chronological order in graphical form. Figure 3 shows the task learning curve generated for the application of an ultrasonic examination technique on a

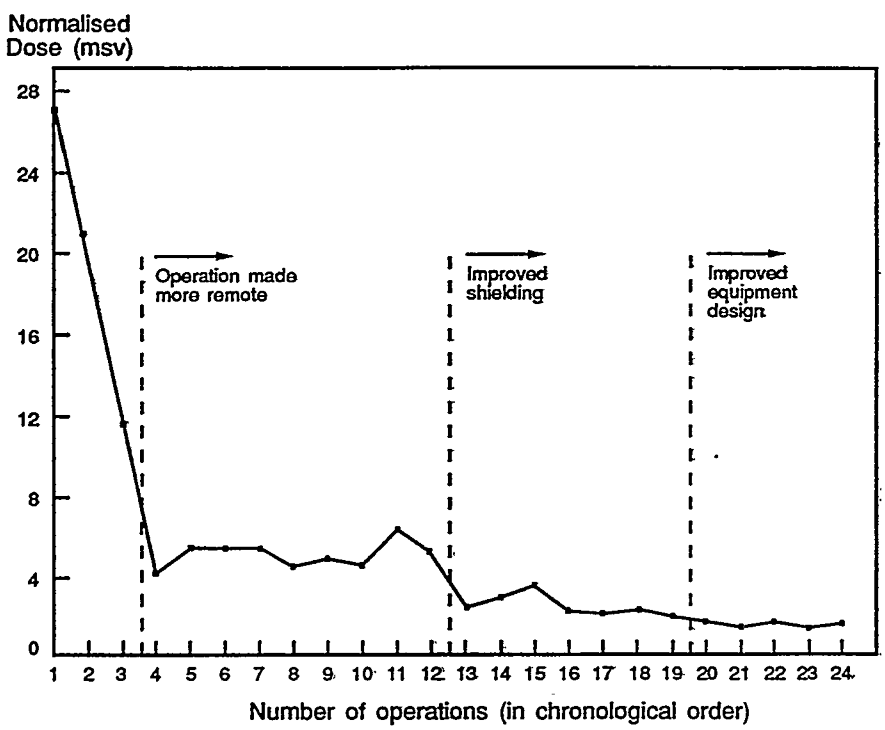

Figure 3. Task Learning Curve for a Non-destructive Examination Technique.

primary plant component. This shows a large reduction in the task dose (normalised to an average primary plant working dose rate) when the technique was first carried out due to initial teething problems being solved and increased familiarity with the equipment. What is also noticeable is the apparent change in dose following the introduction of various dose-reduction measures:

i) greater use of CCTV leading to more remote operation, task dose fell to approximately 6mSv.

ii) increased use of temporary shielding giving greater overall coverage, task dose fell to an average of 3mSv.

iii) improved equipment design requiring less assembly in the radiation area, task dose fell to an average of 2mSv.

Dose-reduction strategy items are being monitored in this way, if applicable, to produce a quantifiable measure of the dose saving achieved. Other benefits arising from this type of approach is that any higher than expected doses will be readily apparent, and the curve can be used for predicting the dose for future applications. A quantifiable measure of effectiveness will be difficult to produce for some of the individual dose-reduction proposals, e.g., setting dose targets and color-coding areas, since they are aimed more at increasing the overall ALARA awareness. However, the overall effectiveness of the strategy will be able to be judged by the total dose to all employees since the strategy was introduced compared to previous years.

Step 6 - Periodically Reviewing the Strategy

Reviews of the dose-reduction strategy are required to summarise the progress of the strategy to date, to ensure that the strategy will meet its objectives and to recommend the next phase of proposals to be implemented. At Rolls-Royce & Associates the first review is being carried out and will recommend an additional further five to ten proposals from the original list be studied further, as well as investigating whether any new dose-reduction methods or ideas are worth examining.

CONCLUSION

The creation of a dose-reduction strategy to bring together and assess, as a unified project, dose-reduction proposals over all aspects of the reactor plant has worked well both in raising the profile of ALARA and in directing effort and resource in the most cost effective and beneficial way. Dose savings achieved as a result of the strategy will continue to be monitored and new proposals implemented in order to continue the downward trend in accrued dose.

Author Biography

Margaret Bennett is a Senior Engineer in the Health and Safety Department at Rolls-Royce & Associates (RRA), whose operations include design and support to the UK's nuclear submarine fleet. She is responsible for providing ALARA advice and developing the implementation of ALARA throughout the Company. She has worked for RRA for 14 years and, prior to her work in the ALARA field, has been involved in various aspects of PWR reactor design and radiation safety. She has a BSc in Physics from St. Andrews University, Scotland.

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4-5

PROGRESS REPORT ON THE MANAGEMENT OF THE NEA ISOE SYSTEM

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ABSTRACT

The Information System on Occupational Exposure (ISOE) was launched by the Organisation for Economic Cooperation and Development (OECD), Nuclear Energy Agency (NEA) on 1 January, 1992, to facilitate the communication of dosimetric and ALARA implementation data among nuclear utilities around the world. After two years of operation the System has become a mature interactive network for transfer of data and experience. Currently, 37 utilities from 12 countries, representing 289 power plants, and 12 national regulatory authorities participate in ISOE. Agreements for cooperation also exist between the NEA and the Commission of the European Communities (CEC), and the Paris Center of the World Association of Nuclear Operators (WANO-PC). In addition, the International Atomic Energy Agency (IAEA) is acting as a co-sponsor of ISOE for the participation of non-NEA member countries. Three Regional Technical Centres, Europe, Asia, and Non-NEA member countries, serve to administer the system. The ISOE Network is comprised of three data bases and a communications network at several levels.

The three ISOE data bases include the following types of information: NEA1 - annual plant dosimetric information (annual operational collective dose, and annual outage collective dose, man-hours, and number of workers broken into 20 job categories and 75 sub-categories, etc.); NEA2 - plant operational characteristics for dose and dose rate reduction (primary water chemistry, cobalt replacement programs, ALARA organisation structure, start-up and shut-down procedures, etc.); and NEA3 - job specific ALARA practices and experiences.

The ISOE communications network has matured greatly during 1992 and 1993. In addition to having access to the above mentioned data bases, participants may now solicit information on new subjects, through the Technical Centres, from all other participants on a real-time basis. Information Sheets on these studies are produced for distribution to all participants. In addition, Topical Reports on areas of interest are produced, and Topical Meetings are held annually.

INTRODUCTION

In order to facilitate the exchange of techniques and experiences in occupational exposure reduction, the Nuclear Energy Agency (NEA) of the Organisation for Economic Cooperation and Development (OECD) launched the Information System on Occupational Exposure (ISOE) on 1 January 1992 after a one year pilot program. This three level data base joins utilities and regulatory agencies throughout the world, providing occupational exposure data for trending, cost-benefit analyses, technique comparison, and other ALARA analyses.

The ISOE Structure

The ISOE system consists of three data bases of occupational exposure information. The first, NEA1, contains for each participating reactor various radiation protection performance indicators: total annual collective dose, non-outage annual collective dose, outage annual collective dose divided into 20 job categories and 75 sub-categories, annual collective man-hours and number of workers associated with each job category and sub-category, and annual individual dose distribution are included. Although not all reactors provide data for all categories, all the data provided are updated annually.

The second data base, NEA2, contains for each participating reactor information concerning methods and techniques used for dose and dose rate control. Primary water chemistry, cobalt replacement/reduction programs, primary water filtering, surface preconditioning, decontamination, work practices, ALARA organisation and management, tools and procedures, and motivation and training practices are listed. The dosimetric effect of each practice is quantified as best possible. This type of information normally evolves rather than changes, thus this data base is updated by the participating utilities on an as needed basis. Information for this data base will be collected for the first time in 1994.

The third data base, NEA3, contains details on the dosimetric results of specific operations. Items as large as the removal of the reactor temperature detector bypass system, or as specific as reactor vessel head control rod drive penetration inspections have been the subjects of NEA3 reports. Important radiological aspects of the operation, and the name, address, and phone number of a contact person for further information are listed. The participating utilities are encouraged to complete NEA3 reports as often as they perform operations with interesting radiation protection aspects.

ISOE Software

To facilitate access to and interrogation of the data bases, the user-friendly Windows environment is used. The NEA1 and NEA3 questionnaires are computerized, and NEA1 is available in English, French, German, Spanish, Italian, Dutch, and Japanese. The NEA2 questionnaire will be computerized for use in 1994. A multi-layered key-word search routine facilitates the interrogation of the NEA3 data base, and can be used to generate lists of reports in an area of interest. Finally, the interrogation of the NEA1 data base, for numerical analyses of occupational dose data, will in 1994 be performed using a Windows-based system.

CURRENT STATUS OF PARTICIPATION

As ISOE nears the end of its second full year of successful operation, its list of participants continues to grow:

> Currently, 37 utilities from 12 countries, and 12 national regulatory authorities participate. Additional data for reactors in non-participating countries is collected from published reports such that the data base now represents 185 PWRs, 84 BWRs, 20 CANDU reactors (see Appendix I for a full list of participants).
>
> The Commission of the European Communities (CEC) and the NEA have signed a cooperative agreement such that the ISOE data base now also serves the European Community's data needs, and such that ISOE and CEC programs in occupational exposure remain complementary.
>
> Several European regulatory authorities are investigating the use of the ISOE format for their national occupational exposure reporting systems to avoid duplication of effort by utilities.

The Paris Center of the World Association of Nuclear Operators (WANO-PC) and the NEA have signed a Memorandum of Understanding to assure co-ordination of the activities of the two organizations in the field of occupational exposure.

The International Atomic Energy Agency (IAEA) and the NEA have established a cooperative agreement whereby the IAEA co-sponsors ISOE, acting as the ISOE Technical Centre for the participation of non-NEA countries. China, Mexico, and Hungary are participating in the one year trial run of this program.

The Nuclear Power Engineering Corporation (NUPEC) has agreed to act as the ISOE Technical Centre for Asian NEA member countries, notably Japan and Korea.

The Centre d'Etude sur l'Évaluation de la Protection dans le Domaine Nucléaire (CEPN) acts as the ISOE Technical Centre for European NEA member countries.

A North American Technical Centre, which will serve the United States and Canada, will be established during 1994, after a one year, small scale trial run.

Thus ISOE has a wide following and is the most complete occupational exposure data base in the world. The value of such a widely used system is its ability to efficiently facilitate the exchange of occupational exposure reduction experiences and practices among participants.

THE USE OF THE ISOE SYSTEM AND NETWORK

There are several diverse ways in which ISOE can be used by its participants. The ISOE System, consisting of the three data bases and their associated software, can be used for statistical and comparative studies, and as a source of good practices and experiences. The ISOE Network, which consists of all Participating Utilities and Authorities, Regional Coordinators for certain countries, and the ISOE Technical Centres, serves as an open line of communication for the real time exchange of data, experiences, policies, practices, etc. In addition, ISOE Expert Groups are established from time to time to perform specific studies based on participant's needs. More regularly, the Annual ISOE Steering Group meeting includes a Topical Session during which current issues of interest to the participants are presented and discussed.

Use of The ISOE System

As described earlier, the three ISOE data bases contain annual operational dosimetry data (NEA1), plant configurational and administrative data (NEA2), and operational experience reports (NEA3). These data bases can be used individually, or together, to perform interesting studies.

NEA 1

At the end of each calendar year, operational occupational exposure data is collected from all Participating Utilities for the NEA 1 data base, and is summarized and analyzed. This data is most useful for trending and comparative studies. The evolution of average annual collective dose per reactor, as shown in Figure 1 for PWRs and Figure 2 for BWRs, is an example of the type of trending which can be performed. These types of studies are published each year by the NEA in an Annual ISOE Report[1,2]. More detailed analyses are also possible.

Figure 1: Average Annual Collective Dose per Reactor (PWR)

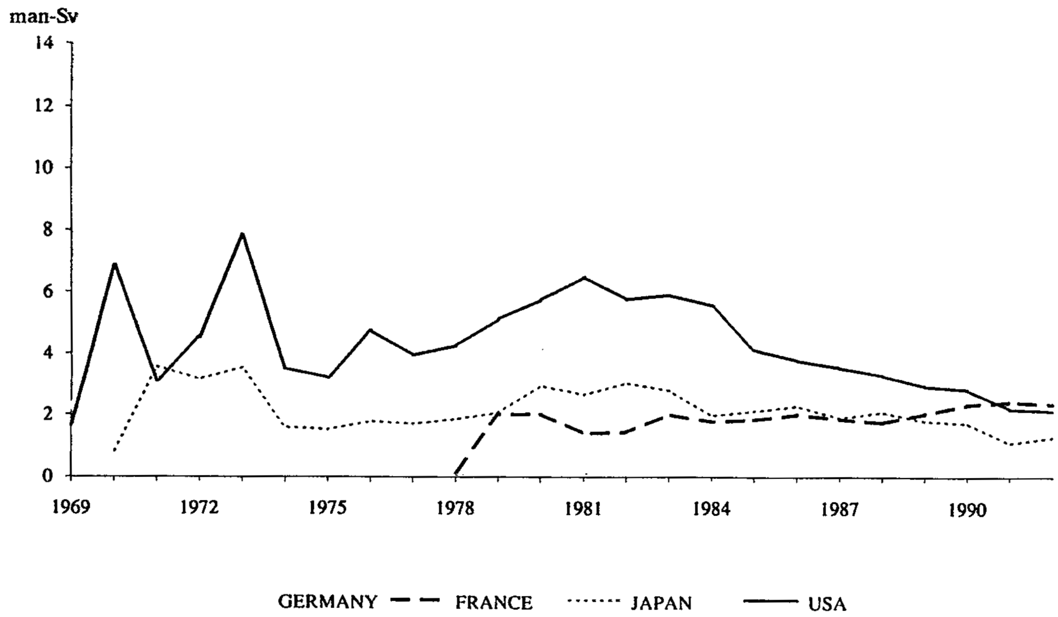
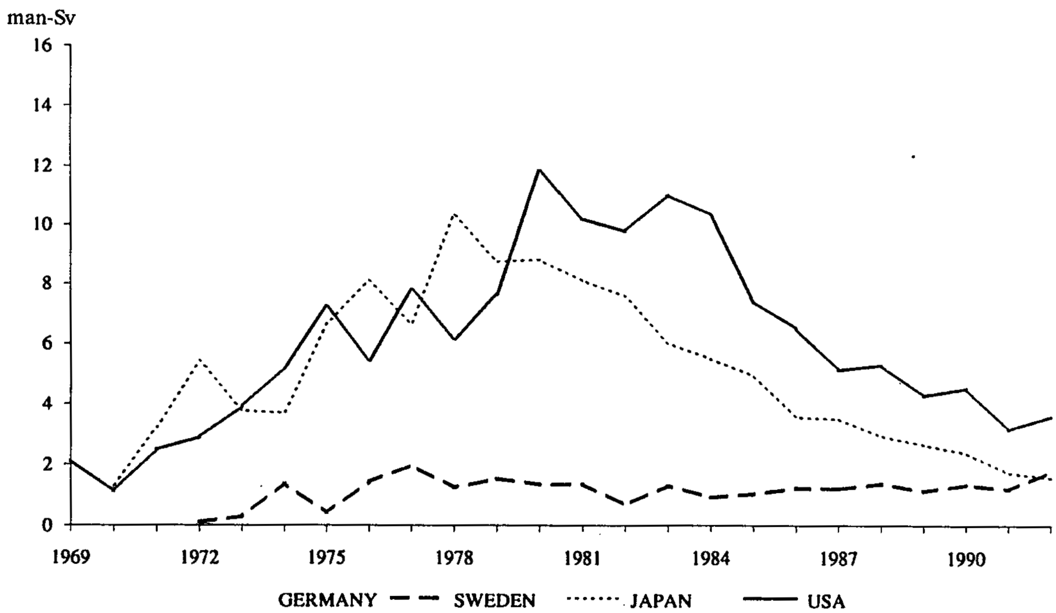


Figure 2: Average Annual Collective Dose per Reactor (BWR)



For Example, ISOE data through 1991 was used by the Centre d'étude sur l'Evaluation de la Protection dans le domaine Nucléaire (CEPN) to study, for the NEA, the effect of reactor age and size on occupational exposure[1]. Tables 1 to 3 show the results of this study for PWRs, BWRs, and CANDUs in Europe, North America, and Asia. Although it is difficult to draw concrete conclusions from such a study, partly due to the irregular statistics of small sample sizes, the trend for modern and intermediate age reactors, of large and medium size, is that average annual collective doses increase with age and decrease with size. The decreasing of average annual collective dose inversely with age may be due to the progressive buildup of corrosion products. The decrease with size may be because larger plants are often of more modern design, thus incorporating design improvements.

Another study was performed the following year, again by the CEPN, on the effect of fuel cycle length on average annual collective doses[2]. The average full cycle time, operation time, and refuelling outage time were plotted against average annual collective dose per country (see Tables 4 and 5, and Figures 3 to 8). Fuel cycle lengths were taken from the ELECNUC[3] data base. In order to correctly account for different full cycle lengths, the annual collective dose used for comparison purposes has been averaged over three years (1990 to 1992). Again, concrete conclusions are difficult, however, it is clear that these simple averages, for all ages and sizes of reactor together, do not support an argument that longer cycles result in smaller collective doses. Looking simply at Figures 7 and 8, refuelling outage length versus annual collective dose, the trend appears to show an increasing dose with increasing refuelling outage length. However significant fluctuations in average annual collective dose make it difficult to link this increasing trend uniquely to increases in refuelling outage length.

In addition to annual collective dose data, the NEA 1 data base contains doses by task. European countries are currently the only participants who routinely supply this data. Doses for three such tasks, General Work, Scaffolding, and Insulation, are listed as their percentage of total outage dose in Tables 6 to 7, for PWRs and BWRs, for the years 1990 through 1992. These tasks, known collectively as Services, typically account for significant fractions of the total outage dose. As can be seen from these tables, there is significant variation from country to country. Although further study is necessary to fully understand these variations, these tables demonstrate that some participants have found effective ways to control Services doses.

All three of the studies discussed above show interesting trends in occupational dosimetry, and are intended to demonstrate the range of studies that can be performed using the ISOE NEA 1 data base. Further detailed studies, sorting data by reactor make, age, model, etc., may provide more definitive conclusions, and can be performed by participants based on their needs, using the ISOE data base and software.

NEA 2

Data for the NEA 2 data base will be collected for the first time during 1994. Interesting aspects of this data base, such as the type of primary system water chemistry used, start-up and shut-down procedures, or the use of "standard" shielding and scaffolding configurations, can be used in a comparative fashion by participants. Combined with historical data from the NEA 1 data base, the dosimetric success of various operating regimes and procedures can be studied.

NEA 3

The last of the data bases, NEA 3, is a repository for brief reports on good, and bad, practices, procedures and experiences. The data base can be interrogated at any time by participants, using key-word search software, to learn from the experiences of others. As important as the data contained in each NEA3 report is the name and address of the author for follow-up and in depth questions.

Table 1
Average Annual Collective Dose (man-Sv) for PWRs
as a Function of Reactor Size and Age
for Europe, North America, and Asia

Plant Age	Small Plants (<700 MWe)	Medium Plants (700-1000 MWe)	Large Plants (>1000 MWe)
Europe			
Modern	-	1.25 (3)	0.9 (18)
Intermediate	-	2.3 (25)	1.4 (10)
Old	2.3 (10)	3.2 (19)	5.7 (3)
North America			
Modern	-	2.3 (1)	1.9 (11)
Intermediate	-	2.9 (1)	2.5 (7)
Old	1.9 (8)	2.4 (24)	1.8 (8)
Asia			
Modern	0.4 (1)	-	0.8 (1)
Intermediate	1.2 (1)	0.9 (4)	-
Old	1.7 (2)	2.1 (3)	5.5 (2)

Plant Age
Modern: 1 - 5 years
Intermediate: 6 - 10 years
Old: > 10 years

Table 2
Average Annual Collective Dose (man-Sv) for BWRs
as a Function of Reactor Size and Age
for Europe, North America, and Asia

Plant Age	Small Plants (<700 MWe)	Medium Plants (700-1000 MWe)	Large Plants (>1000 MWe)
Europe			
Modern	-	-	-
Intermediate	-	3.0 (2)	1.3 (7)
Old	1.9 (8)	1.6 (5)	1.0 (1)
North America			
Modern	-	1.9 (2)	1.7 (5)
Intermediate	-	-	3.1 (6)
Old	4.6 (7)	3.8 (10)	2.6 (5)
Asia			
Modern	-	0.3 (1)	0.4 (1)
Intermediate	0.5 (1)	-	2.6 (3)
Old	3.7 (4)	3.6 (3)	2.7 (2)

Table 3
Average Annual Collective Dose (man-Sv) for CANDUs as a Function of Reactor Size and Age
for North America

Plant Age	Small Plants (<700 MWe)	Medium Plants (700-1000 MWe)	Large Plants (>1000 MWe)
North America			
Modern	-	0.4 (2)	-
Intermediate	0.4 (6)	0.4 (3)	-
Old	1.2 (4)	0.7 (4)	-

<u>Plant Age</u>
Modern: 1 - 5 years
Intermediate: 6 - 10 years
Old: > 10 years

Table 4
Average Collective Dose per Reactor Year,
Full Cycle, Operation Cycle, and Refueling Cycle Length
for PWRs for 1990 and 1992

Country	Average Collective Dose per Reactor year (man-Sv)	Average Full Cycle Length (days)	Average Operation Cycle Length (days)	Average Refueling Outage Length (days)
Belgium	1.51	386	347	39
Finland	1.30	382	344	38
France	2.38	433	338	95
Germany	2.12	401	323	78
Netherlands	1.68	380	320	60
Spain	2.02	393	349	44
Sweden	1.01	360	316	44
Switzerland	1.5	361	312	49
Europe	2.14	413	336	77
Japan	1.39	478	363	115
United States	2.44	559	477	82

Table 5
Average Collective Dose per Reactor Year,
Full Cycle, Operation Cycle, and Refueling Cycle Length
for BWRs for 1990 and 1992

Country	Average Collective Dose per Reactor year (man-Sv)	Average Full Cycle Length (days)	Average Operation Cycle Length (days)	Average Refueling Outage Length (days)
Finland	0.90	367	350	17
Germany	2.18	440	381	59
Netherlands	0.88	353	308	44
Spain	4.20	546	482	64
Sweden	1.44	363	331	32
Switzerland	2.04	366	327	39
Europe	1.89	394	353	41
Japan	2.20	507	382	125
United States	3.76	610	511	99

Table 6
Average Percentage of the Total Outage Dose
Spent on Services for PWRs in Europe

Country	1990 (%)	# of Plants	1991 (%)	# of Plants	1992 (%)	# of Plants
General Work						
Belgium	21.44	7	15.96	7	8.31	5
Finland	3.80	2	3.95	2	15.20	2
France	15.80	39	14.24	43	14.53	43
Germany	8.85	2	8.66	4	15.96	5
Netherlands	11.28	1	15.38	1	5.43	1
Spain	15.10	6	17.15	4	14.12	5
Sweden	4.73	3	13.51	2	8.28	3
Scaffolding						
Belgium	1.95	1	2.13	3	2.86	5
Finland	-	-	-	-	3.48	2
France	2.92	39	2.91	43	3.09	42
Germany	1.50	3	2.47	5	3.53	5
Netherlands	4.89	1	7.86	1	5.93	1
Spain	1.81	5	2.27	4	3.16	6
Sweden	1.52	3	0.72	2	1.39	3
Insulation						
Belgium	6.71	5	8.98	3	5.60	6
Finland	-	-	-	-	10.18	2
France	5.68	39	6.40	43	7.36	43
Germany	6.36	3	8.83	5	4.59	5
Netherlands	11.72	1	12.79	1	6.03	1
Spain	5.98	6	6.04	4	7.83	7
Sweden	3.50	3	4.12	2	9.97	3

Table 7
Average Percentage of the Total Outage Dose
Spent on Services for BWRs in Europe

Country	1990 (%)	# of Plants	1991 (%)	# of Plants	1992 (%)	# of Plants
General Work						
Germany	17.37	2	11.71	3	27.40	4
Netherlands	21.73	1	23.43	1	-	-
Spain	16.22	2	13.77	1	12.06	1
Sweden	6.92	9	9.99	9	8.83	7
Scaffolding						
Germany	3.05	3	0.51	1	7.15	3
Netherlands	-	-	-	-	-	-
Spain	2.17	2	1.42	1	3.77	1
Sweden	2.57	9	3.40	8	4.43	7
Insulation						
Germany	6.35	5	9.61	3	28.02	4
Netherlands	-	-	-	-	-	-
Spain	4.86	2	5.75	1	5.49	1
Sweden	8.88	9	11.15	9	18.86	7

Figure 3: Average Full Cycle Length and Average Collective Dose per Reactor Year (PWR, 1990 - 1992)

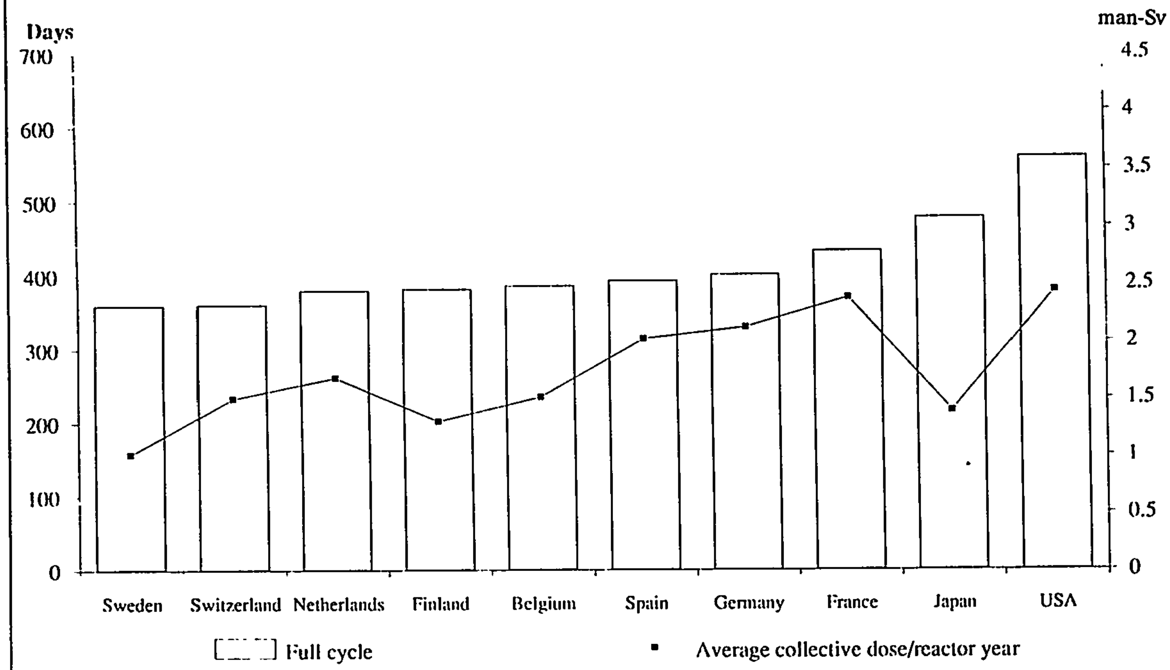


Figure 4: Average Full Cycle Length and Average Collective Dose per Reactor Year (BWR, 1990 - 1992)

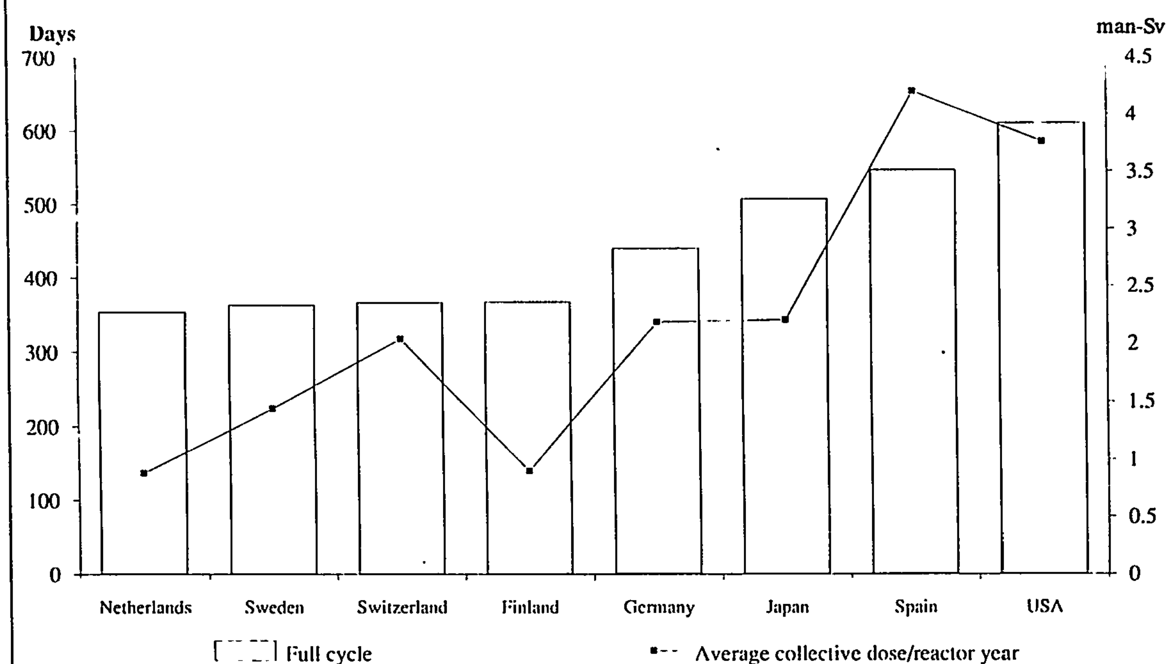


Figure 5: Average Operation Cycle Length and Average Collective Dose per Reactor Year (PWR, 1990 - 1992)

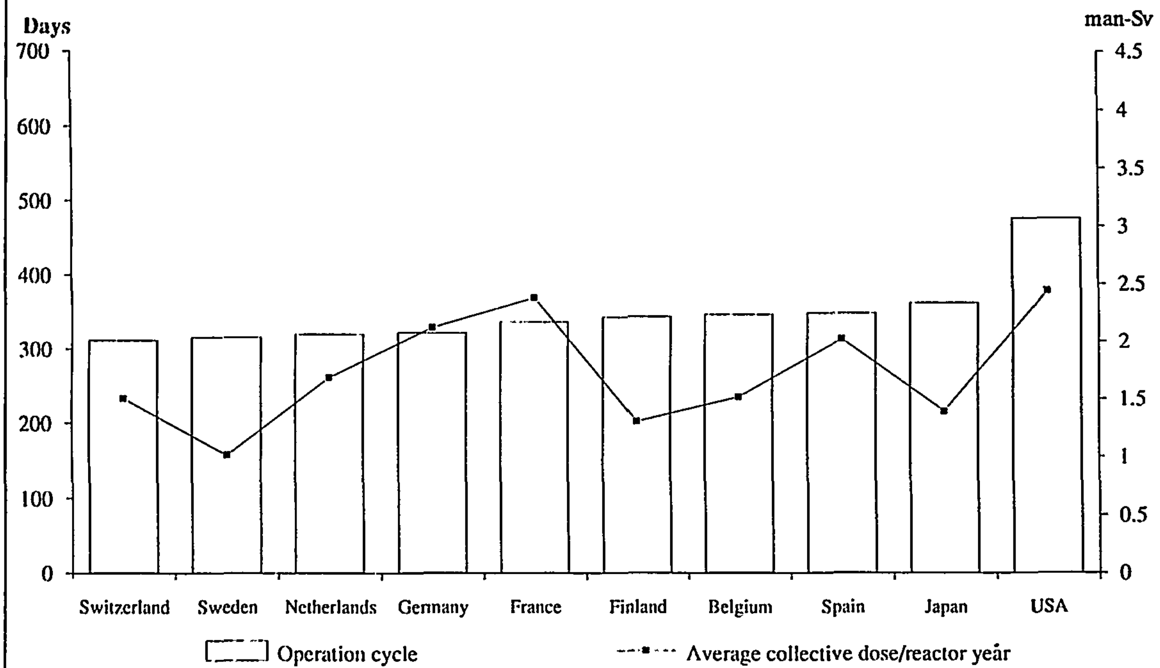


Figure 6: Average Operation Cycle Length and Average Collective Dose per Reactor Year (BWR, 1990 - 1992)

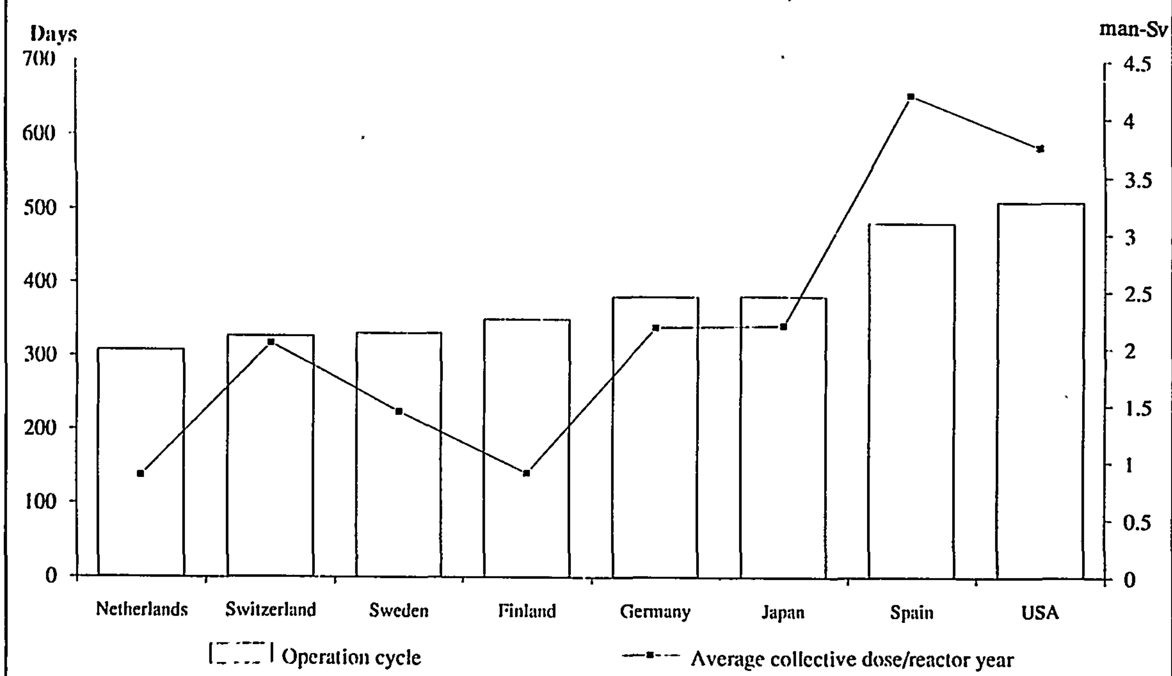


Figure 7: Average Refueling Outage Length and Average Collective Dose per Reactor Year (PWR, 1990 - 1992)

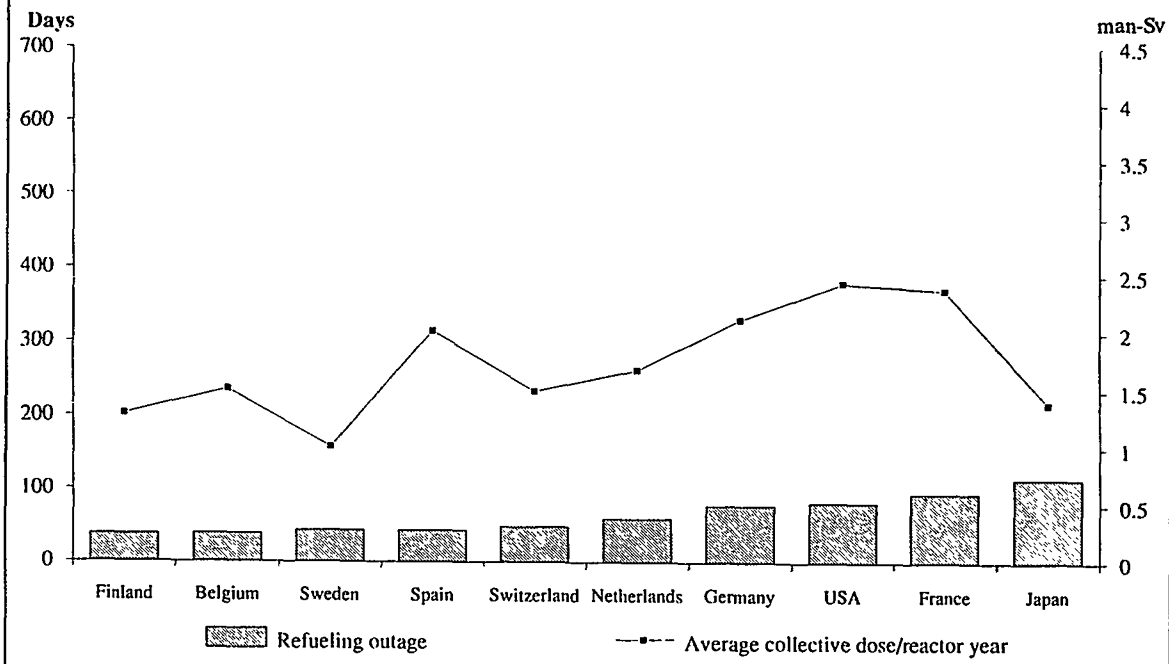
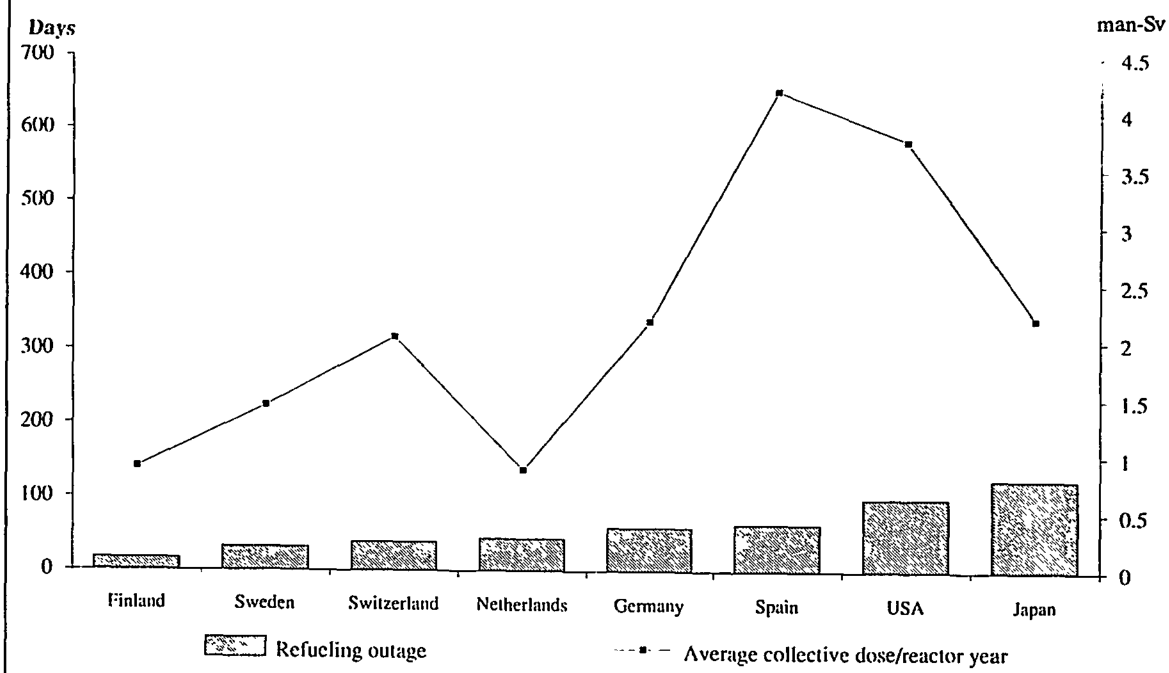


Figure 8: Average Refueling Outage Length and Average Collective Dose per Reactor Year (BWR, 1990 - 1992)



In addition, in the case where several NEA 3 reports have been submitted on the same subject, Topical Reports can be prepared comparing the various experiences. For example, several NEA3 reports were submitted by French and Swedish reactors, summarizing their experiences in reactor vessel head inspection and repair, prompted by the discovery of cracks in the thermal sleeves of control rod drive vessel head penetrations. The French found dose rate reduction factors to be superior using mechanical brushing (reductions from 2 to 10) than those attained using high pressure water decontamination (factor of 1.65). Ambient dose rates both under and on the vessel head were reduced by factors of 3 to 5 by the use of specially adapted shielding. In conjunction with this work, both the Swedish and the French found the removal and replacement of thermal insulation on the vessel head to be dosimetrically costly, such that the Swedish have replaced old style insulation with modern quick-disconnect insulation to facilitate future inspections and refuellings. The French have designed special scaffolding to speed installation. Robotics is in development in both countries.

Another interesting Topical Report concerned the removal of the reactor temperature detector bypass system. Seventeen NEA3 reports on the subject were found and compared.

A Topical Report to be completed in 1994 will compare the steam generator replacement operations at Doel in Belgium, North Anna in the United States, Dampierre and Bugey in France, and Beznau and Ringhals in Sweden.

The ISOE Network

The ISOE network consists of all participating utilities and authorities, the ISOE Technical Centres, and national ISOE coordinators. Participants interested in the experience of others in specific areas not already covered by NEA 3 reports may request that the Technical Centres solicit the needed information. Participating utilities, authorities, and national ISOE coordinators are then contacted by the Technical Centres and the resulting information is passed on to the requestor and made available to all other participants. Recent examples of the use of this network system have included a utility's request for information concerning the decontamination of the residual heat removal (RHR) system for the replacement of an RHR heat exchanger channel head; an authority's request for information concerning the dosimetric impact of vessel head inspections in France, Switzerland, Sweden, and Belgium; a utility's request for information regarding experience in reactor vessel decontamination, a utility's request for experience in refuelling pool decontamination, and a utility's request for experience in the repair of fuel storage rack anti-seismic snubbers. In all these cases, the ISOE network was questioned by the European ISOE Technical Centre (the CEPN), and the information collected from participating utilities was passed on to the requestor within a very short period. Topical Reports will be written on these subjects and distributed to all participants.

ISOE Expert Groups

Based on the needs of the ISOE participants, as decided by the ISOE Steering Group, Expert Groups may be established to study specific questions. Two such Expert Groups are currently at work.

The first Expert Group is investigating dosimetry recording and reporting practices to better understand, and thus analyze, the data supplied to ISOE. For example, whether or not background is subtracted from reported doses, what dose recording level is used and how are doses below this level reported, are extremity or skin doses recorded and reported, are neutron doses recorded and reported, etc., are the types of questions which need to be answered so that valid analyses of the ISOE data can be performed.

The second Expert Group is trying to quantify the impact of "work management" on occupational exposure. In that most radiation protection practices must be justified, often in monetary terms, in order to gain management support, techniques and approaches to quantification are being studied. In addition, this Expert Group will also be addressing the somewhat related question of the impact of regulatory requirements on occupational exposures. This study is intended to provide data for the ongoing discussion of nuclear safety versus occupational exposure.

Both of these Expert Groups will produce ISOE Technical Reports, based on their studies, which will be distributed to participants and, based on the recommendation of the ISOE Steering Group, may be issued as NEA reports.

ISOE Technical Sessions

Each year the ISOE Steering Group meets to discuss administrative and organizational issues associated with ISOE. In addition, a Topical Session, like a small workshop, is held during which "invited papers" on topics of current interest are presented by participants and discussed. Topics such as lessons learned during steam generator replacement, failed fuel prevention programs, and rework prevention programs, will be discussed at upcoming meetings.

CONCLUSIONS

After two years of operation and expanding participation, the ISOE system appears to have reached the "critical mass" necessary to efficiently provide its users with a very broad range of detailed information. Continued growth, and efforts by all participants to deliver timely and useful information, will help to foster the dedication necessary to keep ISOE an up-to-date conduit for occupational exposure experience throughout the world.

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Author's Biography

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THE ECONOMICS OF RADIOLOGICAL PROTECTION

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(The following is a transcript of Mr. Lochard's presentation.)

The objective of my presentation this morning is to give you an overview of the status of economic thinking in radiological protection. I will not say anything new about this topic, but I will try to put everything into perspective and give you a flavor about the state of the art in this field.

I would like to remind you of the key reason why economics is at the heart of ALARA thinking. If we are dealing with the stochastic risk related to radiation, the basic problem is that we don't know anything about the real shape of the dose-risk relationship -- is there a threshold or no threshold. The attitude of the ICRP, and I would say the entire radiological protection community, was to adopt a prudent attitude and to assume that there was no threshold. The result of this basic assumption is to enter into a risk reduction approach. As a first step, you could imagine that the zero risk objective is the good one, but you have to be careful. This is where the economical aspects enter into the system. Looking for zero risk is not the right way, because on one side this leads to a misallocation of protection resources in society, and secondly, you can have transfers of risk from one group to another. This is an important topic. When you try to reduce a risk to zero, you generally generate some risk for other groups. For these two basic reasons, efficient allocation of resources and avoidance of risk transfers between groups in the society, ALARA is the proper route for dealing with radiological risk (Figure 1). I mentioned this point yesterday during the panel. The result of this attitude is that a residual risk always remains. Whatever you do, if you don't go to the zero risk level, you will leave some residual level of risk. This residual level of risk is a question to be looked at carefully.

In terms of economic models, the story began with ICRP Publication 22 and the introduction of the so-called cost-benefit model, which is, in fact, an adaptation of the famous optimal pollution control model developed by economists in the framework of welfare economics to deal with externalities related to environmental pollution. The cost-benefit model in radiological protection is just an adaptation of this classical model which all economists learn in their first year of school. The key feature of this model is to try to find the minimum total cost including, on one side, the cost of detriment, i.e., the economic evaluation of the detriment related to radiological risk, and on the other side, the cost of protection (Figure 2). The whole system is driven by the fact that the cost of protection is following the law of diminishing marginal returns, which is also a key law in the economic thinking. This is not a physical law, but an empirical law. In many situations, the more money you spend, the less efficient it is to reduce the risk. This is the basic model on which the whole economics of radiological protection is based.

I will now develop two sides of this model, i.e., the cost of detriment and the cost of protection. As far as the detriment cost is concerned, there has been an important evolution since ICRP 22. At the beginning, people were focused on the reduction of collective exposure using a single monetary value of the man-Sv. There was a slow evolution towards taking more into account individual levels of exposure. To say that in a condensed way, if we want to integrate the most recent developments from ICRP, the main objective when looking for the reduction of exposures is, of course, to reduce collective exposures, but at the same time, to reduce the dispersion in individual exposures as well as the highest individual exposures. This is clearly mentioned in ICRP 60 through all the developments related to the concept of dose constraints. The challenge now, in terms of monetary valuation of the man-Sv, is to find a way of dealing with these three objectives at the same time. One solution is to use models for the valuation of the man-Sv like the one described on Figure 4.

Referring to this figure, you have on the ordinate the monetary value of the unit of collective exposure, and on the abscissa is the individual level of exposure expressed as annual dose. For doses under d_0 you don't take care about the level of individual exposure because the differences are not meaningful. You can imagine for d_0 something like 1 mSv, for example. However, when you are dealing with higher individual doses, the "alpha value" is increasing with the level of individual dose according to an aversion factor, which is noted here in the formula as "a." Note that "a" must be greater than 1 to cope with the three objectives: reducing at the same time the collective exposure, the dispersion of individual doses, and the highest individual doses.

There is another important issue in terms of economics within the valuation of detriment, which is the problem of how to deal with future detriment. Many times we are faced with a situation of choosing to spend money today for avoiding doses in the future. The traditional way of dealing with this situation is to use a discount factor, and engineers tend to use the classical interest rate approach. This is an important point. It was demonstrated recently by economists that the interest rate is not applicable to nonmarket goods, and obviously with the radiological detriment we are dealing with a nonmarket good. It was also demonstrated that the tradeoff between costs and exposures distributed in time relates on some sort of willingness to pay from individuals. The market cannot give us any good numbers. We have to refer to the so-called social values, and we need to develop contingent valuation approaches in this matter. In this perspective, it seems that using a discount rate in the range of 1-5% is appropriate to deal with future detriment. This is an area which needs to be investigated further, especially in the field of radiological wastes where we deal with very long time frame.

To finish the first part of this presentation on the value of the detriment, I would like to emphasize the need to differentiate values of man-Sv according to risk situations. If we take, for example, occupational exposure, we can assume that workers are informed in advance about the risk in the industry, and they are willing to join the industry because of the benefit that they obtain. This is totally different from the public that is living around an installation for which the risk can be seen as imposed. We can also think about the medical exposures where you get a direct benefit from the exposure you receive voluntarily. All these types of exposures have a clear impact on the risk perception and should be translated in one way or another into different alpha values according to the risk situation. This is an area we need to develop further in the future. There is, however, a consensus among those who are dealing with these types of problems to consider the willingness to pay approach developed by economists as a means to establish alpha values related to risk perception.

The next part of my presentation is related to the cost of protection. The cost-benefit model proposed by ICRP in Publication 22 and repeated in all successive publications is based on the assumption that protection and production costs are independents. In other terms, it means that if you improve the protection, this does not affect operation and maintenance costs. This assumption is misleading. We have many empirical analyses, especially in the nuclear industry, suggesting that there is a clear correlation between the improvement of protection and the reduction of operational costs. Traditionally, we have thought that improving radiation protection is spending more money and reducing the benefit of the activity or the practice. In fact, there are some possible synergies between improving radiation protection and also reducing operational costs. This is clearly shown on Figure 8. In the ordinate we have the outage collective dose related to the 1300 MWe French PWRs for the last outages in 1993, and in abscissa we have the cost of these outages. We can see a clear correlation between the collective dose associated with outages and the cost. This needs to be further analyzed to see how it relates to the lengths of outage, but it is a very encouraging curve showing that there is probably a very good correlation between good protection and a reduction of total cost of operation.

The last point I would like to make is related to the compensation of residual risk. It is clear that the cost-benefit model proposed by ICRP is a way to internalize the radiological detriment up to the point where the marginal cost of protection equals the marginal cost of detriment. But there is always a residual risk. When you are ALARA, you have reached the acceptable level of residual risk, but a residual risk still remains, and one way to spend more money if a radioinduced disease appears after exposition. This compensation can be based on the attributable risk approach, which is another facet where the economists have something to say about radiological risk management.

To conclude, I would like to come back to the curve that was produced by CEPN some years ago when the research group was working on the use of robotics in nuclear power stations (Figure 10). Classically, economists have always presented cost-effectiveness where the reduction of exposures is just more costly. In fact, because of the correlation that I have mentioned between protection and production costs, there is a large potential for reducing exposure and saving costs at the same time before entering into the situation where reducing risk is spending more money. Our experience so far demonstrates that further improvements in dose reduction within or outside nuclear industry can still be achieved without significant increases in costs.

Author Biography

Jacques Lochard is currently the Director of the Nuclear Protection Evaluation Center (CEPN). CEPN is a nonprofit organization, founded in 1976, for research and consulting in the area of optimization of radiological protection and comparative assessment of health and environmental risks associated with energy system. Mr. Lochard's main contribution in radiation protection has been the development of methodologies and implementation tools in the field of optimization of radiological protection. Mr. Lochard is currently the Secretary of the French Society of Radiation Protection (SFRP). He is also widely involved in the international radiation protection scene. He is a member of the Executive Council of the International Radiation Protection Association (IRPA); a member of the Committee on Radiation Protection and Public Health (CRPPH) for the Nuclear Energy Agency of the OECD, and Secretary of Committee 3 of the International Commission on Radiological Protection.

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Figure 1

THE FOUNDATION OF THE ALARA PRINCIPLE

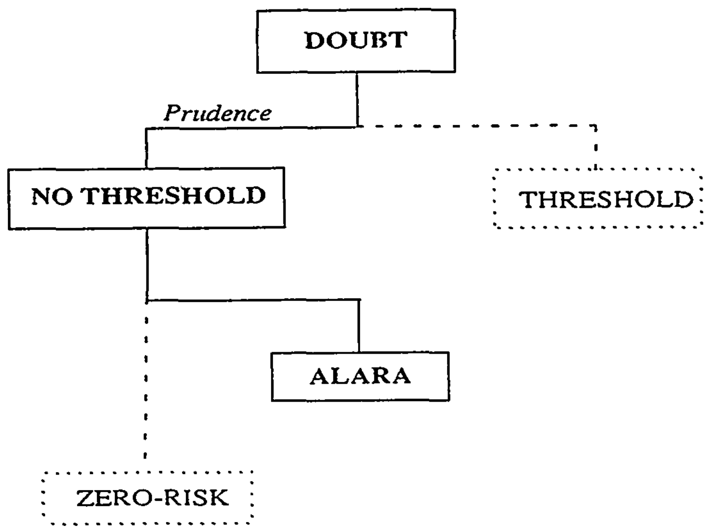


Figure 2

THE OPTIMAL POLLUTION CONTROL MODEL APPLIED TO RADIOLOGICAL PROTECTION

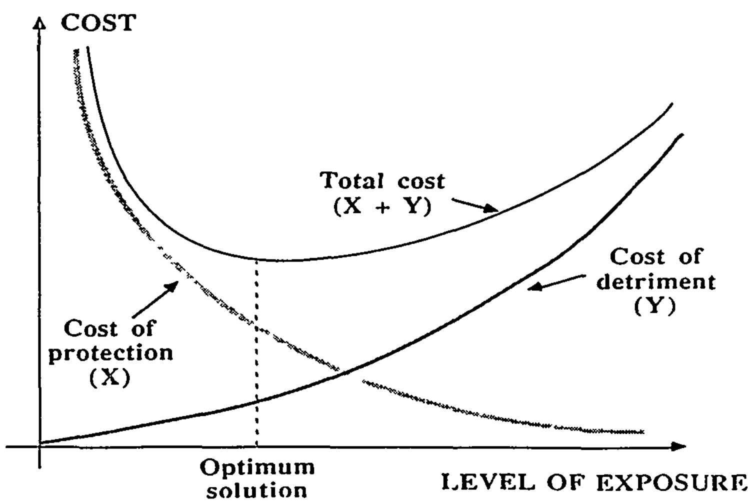


Figure 3

DETRIMENT COST AND RISK AVERSION
[From ICRP 26 to ICRP 60]

- ☐ REDUCTION OF COLLECTIVE EXPOSURE
- ☐ REDUCTION OF THE DISPERSION OF INDIVIDUAL EXPOSURES
- ☐ REDUCTION IN PRIORITY OF HIGHEST INDIVIDUAL EXPOSURES

Figure 4

A MODEL FOR THE MONETARY VALUATION OF THE DETRIMENT

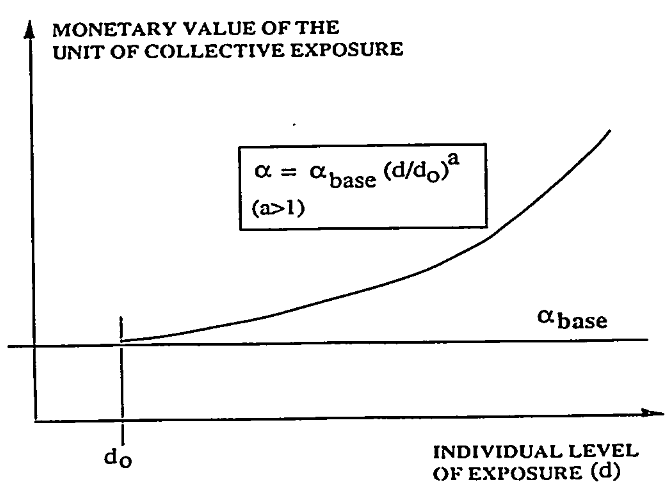


Figure 5

THE VALUATION OF FUTURE DETRIMENT

- ❑ INTEREST RATE IS NOT APPLICABLE TO NON MARKET GOOD
- ❑ THE TRADE-OFF BETWEEN COSTS AND EXPOSURES DISTRIBUTED IN TIME IS RELATED TO THE WILLINGNESS TO PAY FROM INDIVIDUALS
- ❑ A DISCOUNT RATE IN THE RANGE OF 1 TO 5 % IS IN ACCORDANCE WITH EMPIRICAL DATA

Figure 6

MONETARY VALUES OF THE MAN-SIEVERT AND RISK SITUATIONS

- ❑ BECAUSE OF RISK PERCEPTION CONSIDERATIONS THERE IS A NEED TO DIFFERENTIATE 'ALPHA VALUES' ACCORDING TO EXPOSURE SITUATIONS [occupational, public, medical...]
- ❑ THE USE OF THE WILLINGNESS TO PAY METHOD SEEMS TO BE THE MOST APPROPRIATE APPROACH TO ESTABLISH RISK PERCEPTION RELATED 'ALPHA VALUES'

Figure 7

THE PRODUCTION FUNCTION AND THE COST OF PROTECTION

- ❑ THE INDEPENDENCE BETWEEN PROTECTION AND PRODUCTION COSTS ASSUMED IN THE ICRP COST-BENEFIT MODEL IS MISLEADING

- ❑ EMPIRICAL ANALYSIS ARE SUGGESTING A POSITIVE CORRELATION BETWEEN THE IMPROVEMENT OF PROTECTION AND THE REDUCTION OF OPERATION COSTS

Figure 8

COLLECTIVE DOSE AND OPERATIONAL COSTS [1300 MWe French PWRs refuelling outages in 1993]

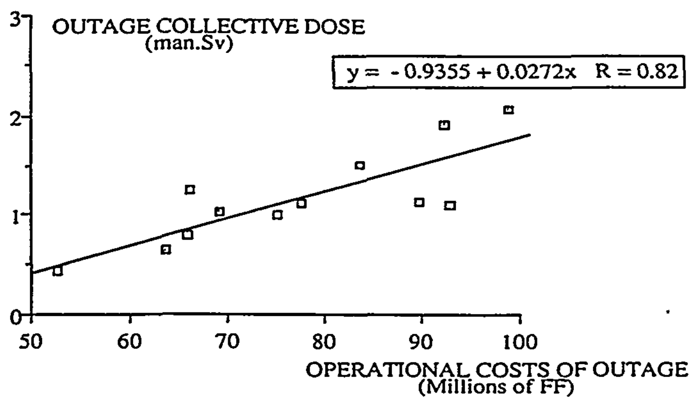


Figure 9

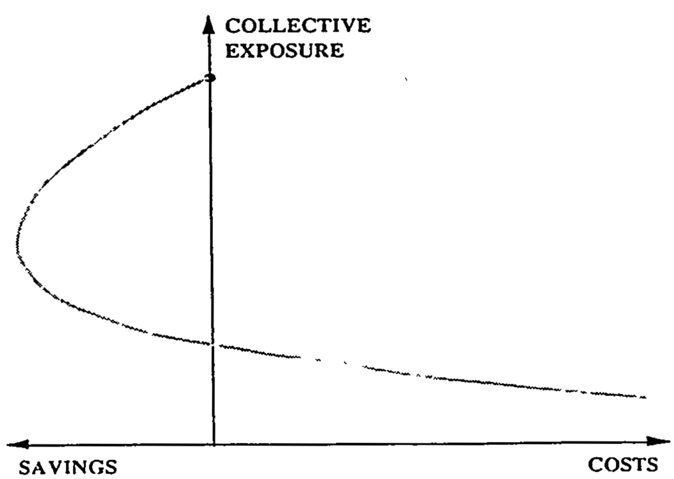
THE COMPENSATION OF RESIDUAL RISK

- ❑ THE USE OF MONETARY VALUES FOR ALLOCATING PROTECTION RESOURCES IS AN EX-ANTE INTERNALISATION OF THE AVOIDED DETRIMENT

- ❑ COMPENSATING RADIO INDUCED DISEASES WHEN THEY OCCUR IS A MEANS TO INTERNALISE THE RESIDUAL RISK

Figure 10

THE 'BENEFITS' OF RADIOLOGICAL RISK REDUCTION



SESSION 5

PANEL DISCUSSION ON ECONOMICS VS. EXCELLENCE

Chair:

John W. Baum

SESSION 5

PANEL DISCUSSION ON ECONOMICS VS. EXCELLENCE

Chair: John W. Baum

JOHN BAUM is a Senior Scientist at Brookhaven National Laboratory where he is Division Head of Radiological Sciences and manager of the ALARA Center. Dr. Baum has several years of experience in applied health physics, and for four years, was lecturer in Radiological Health at the University of Michigan. He has been at BNL for the past 29 years doing research in radiation protection and dosimetry. He is a certified health physicist, a member of the NCRP, and has worked with NCRP, ICRP ICRU, ANSI, and ASTM committees. He chairs an NCRP Committee on ALARA at Nuclear Power Plants.

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EXPERT PANELISTS

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PANEL DISCUSSION ON ECONOMICS VS. EXCELLENCE

Baum: We have a very interesting and distinguished group of panelists. We have not only health physicists, but economists, chemists, and other specialists on our panel. Jacques Lochard is an economist, and one of our panelists has a degree in business administration. It should be a very interesting discussion. I would like to briefly introduce the subject, which is "Economics vs. Excellence," and perhaps that is a misnomer because in my mind, ALARA equals optimization equals excellence. This is the goal. The various speakers during the previous sessions have talked about goals, dose goals, and so on. There is always a question in my mind, are those goals seeking an optimum or is it just dose reduction? We are interested, of course, in the optimization process. It will be very interesting to see what our panelists have to say on this subject.

We have with us, beginning on your left, Brian Richter, who is a Senior Cost Analyst in the Division of Regulatory Application with the NRC's Office of Nuclear Regulatory Research.

Next, we have Harvey Cybul, who is presently serving as the Manager of the Radiological Protection Department at the Institute of Nuclear Power Operations.

Robert Giordano, is currently the General Electric Nuclear Energy Radiation Protection/ALARA Senior Program Manager.

Next is Floyd Spivey, who is currently the ALARA Manager at the Tennessee Valley Authority's Browns Ferry Nuclear Plant.

Jacques Lochard, who you are well acquainted with, is the current Director of CEPN, which stands for Nuclear Protection Evaluation Center.

Christopher Wood, you all know well from EPRI, is the Senior Program Manager in the Nuclear Power Division at the Electric Power Research Institute.

Finally, Alan Homyk is the Radiation Protection Manager at Con Edison's Indian Point 2 Nuclear Power Station. Alan is the man with the master's degree in business administration.

Welcome, panelists. I would like to open the session with a brief talk from each of the panelists, and then we will open it for questions and discussion. Brian would you like to begin?

Richter: I am providing a synopsis of what the NRC is doing with respect to its effort at revaluing the unit of radiological exposure. For approximately the last two decades, the NRC and its predecessor agency, the Atomic Energy Commission, have used a conversion factor of \$1,000/person-rem as the monetary valuation of the consequences associated with radiological exposure. That is, an increase or decrease of person-rem is valued at \$1,000/person-rem in order to allow a quantitative comparison of the values and impacts associated with a proposed regulatory decision. As an aside, I might add that regulatory actions needed to insure adequate protection of the public health and safety are not subject to a value-impact assessment and thus the \$1,000/person-rem value is not operative in these circumstances except in assessing possible alternative approaches to achieve the necessary level of protection. Nevertheless, this value has been used as a reference point in value-impact based regulatory decision making involving routine emissions, accidental releases, and, of course, 10CFR20 ALARA programs. Over the years, the NRC has become increasingly aware of alternative estimates and mythological approaches for arriving at a conversion factor. In addition, questions have surfaced on the continued validity of the $1,000/person-rem conversion factor because basic parameters such as the value of the dollar and risk factors have changed dramatically over this period. Such factors have potentially significant effects on the value of this conversion factor. In the NRC's view, a thorough reassessment of the $/person-rem value and its application in NRC regulatory decision-making is needed. Therefore, the NRC is going to introduce for public consideration a proposed revision to the dollar valuation of radiation exposure that would be used by the NRC as a reference point to guide regulatory decision making affecting the public health and safety. It is expected that a Federal Register notice will be sent to the Commission this fall to seek public comment on the revised dollar value per person-rem and its supporting analysis. I might mention that John Baum's recent report, Value of Public Health and Safety Actions and Radiation Dose Avoided (NUREG/CR-6212), is going to be a key component of that action. While the Commission has not seen the proposed revision, it contains a key change from the present policy that the $1,000 value covers all off-site impacts. The proposed Federal Register notice calls for the revised value to cover health effects only. Of course, off-site non-health effects will also be addressed with several options presently under consideration. Also, the Commission appears to be changing its course in the handling of the monetary worth of the unit of radiation exposure in another way. The Commission published regulatory analysis guidelines of the U.S. Nuclear Regulatory Commission, Draft Report for Comment, that's NUREG/BR-0058, Revision 2, in August 1993. These draft guidelines call for a present worthing, or discounting, to be used for all values and impacts including radiation exposures. Lastly, it should be noted that, in order to be consistent with the Commission's policy on metrication, it is planned that the revised value when published in the final Federal Register notice will be expressed in $/person-cSv with the value in standard units following parenthetically. However, for purposes of continuity, $/person-rem shall be the unit used throughout the paper seeking public comment.

Baum: Thank you, Brian. Harvey, would you like to say a few words?

Cybul: When we talk about achieving excellence in radiological protection, namely in dose reduction, how much is good enough? What can we afford? These are the questions that are facing the U.S. nuclear utilities today. Increased competition caused by deregulation and the emergence of the independent power producers has caused utilities to look very seriously at where they can cut costs. The biggest single area of focus is operating and maintenance costs. That equates really to staff reductions. If we start reducing the staffs, we have the same or more work to be accomplished with fewer people. Potential exists, therefore, for higher individual exposures. We must live with the existing plant designs that we were given. We can look to the Europeans and the Japanese who have newer plants and are achieving excellent results in dose reduction, but for the U.S. plants with the older designs, there are certain limitations that are driven by economics. For example, can an older BWR afford $1 million/year for zinc injection? Does the dose saving realized allow them to compete economically with an independent power producer for their megawatts? Accelerated changeout of control rod blades, for example, is another area where, if money were not an issue, it would certainly be a dose reduction. But is it affordable? Full system decontamination certainly offers an attractive alternative to some of the other methods we are using now, including shielding, but can we afford it? What are acceptable dose limits? We talked yesterday and today about different methods of reducing dose. We have talked about the prospect of changing regulations to reduce the allowed dose, but what is the number, what is the target that we need to have in the back of our minds when we are making these critical decisions? The alternative to reducing dose is *not* reducing dose. The alternative also means do we remain economically competitive, and if we don't remain economically competitive, we shut down. We can have the greatest dose-reduction program going, but if we can't sell electricity cheaply enough, we are out of business. That really is the limit that faces most of our utilities today. Don't let me paint all negative pictures. There are still improvements to be achieved. A lot of people talk about ALARA reviews. I try to present it in the opposite respect. I say, "Let's look at how we can plan work smartly. Let's do a good job of pre-job planning, let's look at all the things we can do to improve the productivity of the worker, let's use mock-ups, make sure our procedures are good. When we go in and do the job, let's do the job right the first time. We do it quickly, efficiently, and we get out." That equates to increased productivity and increased capacity factor -- and we save dose in the process. That's the thinking with which we have to approach most of our work. I also think that in looking at how we can save dose and be economically competitive, we have to recognize that we have a work force that is very intelligent. We have to challenge that work force. We have to ask them to get out of their box -- their comfort zone -- and look at new and innovative ways. Certainly technology affords us many opportunities. All too frequently when you find how the jobs are planned, workers doing the job are not involved in the planning and they are just told to go out and do the job. They have some good ideas. Another thing I think we have to do is change our paradigm as far as contamination is concerned. In the U.S. industry, we put so much emphasis on avoiding contamination, and every time a contamination event occurs, we spend so much time *documenting it*, that we have created a perception that getting contamination is more significant that picking up extra dose. We do a job and we will get 25-30 mrem more than we should have, nothing is documented and nothing is said. The worker gets contamination on his hands and he has to go to the plant manager and explain why he fouled up. So that it is an important consideration. I think in the future we are going to have to look very hard at do we even put people in contaminate-protective clothing? Do we put rubber gloves on to do delicate work, when maybe we should let them take the rubber gloves off and work with their bare hands and maybe get a little contamination? These are new thoughts for a lot of people, but in order to be economically competitive, that's the kind of thinking that has to be brought to the forefront.

Baum: Thank you, Harvey. Robert, how does G.E. look at these matters?

Giordano: We have seen how things are done in the radiation protection aspects in plants all over the world. We have had to live with the requirements of the *specific utilities*. We certainly will do that because that's part of our contract. But as Harvey was mentioning, looking at the competitive costs and the competitive nature of our business these days, the industry generating electricity business, we have to keep down the operation and maintenance (O&M) costs. Harvey alluded to a number of different items. When I go to one utility in Europe and simply see the size of their parking lot, and see the size of the parking lots at some of the locations in the United States, they are both generating power, some of them are doing a better job from the dose reduction value, some of the them are doing a better job from the capacity factor value. What's the difference? Certainly culture is some of it, the union situation is some of it, but there's room. With the pressures coming down to reduce the O&M costs, I would certainly not be surprised to see the VP or the plant manager say to the rad protection department, why are you so much overhead? What can you do at that utility that had to have a radiation control technician escort a worker into the drywell and watch him go up a ladder while decontamination was being performed holding a meter and then walk out? Then that worker, when he was done, had to call down so that the same radiation control technician who was on standby outside the drywell would walk back in to make the same surveys while the man climbed down the ladder and walked out the path. I think that is an area where there would be some challenges as to how the money is actually being spent. We are doing things in the operating plant area to reduce the number of surveillances and frequency of surveillances to assure the plant's safety. Margaret Bennett's paper had a neat little bullet "to direct effort and expenditure efficiently." We have to take more of a look at this area. As I've seen at a number of different places, we can do better with the resources that we have. We are going to be challenged in that area to understand what's best. Why am I doing surveys every shift? Why do I go up to 80-120 additional contract radiation control technicians in preparation for an outage? That money could better be applied to the zinc injection opportunities, to the long-term source reduction activities, to the things that are preserving my plant to be generating the megawatts to keep it going. In the interest of time, I will stop there. Maybe that will be enough to stimulate some other thoughts.

Baum: Thank you, Robert. Floyd, TVA has had a lot of economic challenges in the last few years. Would you like to comment?

Spivey: Mr. Cybul mentioned several of the items that we go through daily at TVA, so I will not repeat them. As an ALARA Supervisor at a 3-unit BWR, in the last 2-3 years, economically I have had the tunnel-vision blinders of "man-rem savings, man-rem savings" taken off substantially. The items that Harvey Cybul and Bob Giordano have both talked about are very active at TVA. When I went in to justify a job, a chemical decontamination project for our last outage, for example, I probably spent at least 80 man-hours justifying to the plant manager why we had do something that, without question, everyone on site knew had to be done to save man-rem. But I still had to spend 80 to 100 hours to justify it because it had to be cost effective. I think the biggest key, the thing that has changed in the last 2 to 3 years that interests me is the meaning of the words "reasonably achievable." Indeed, we do want to do every job with the least amount of man-rem that we can, but it can't be just man-rem- reduction-driven only. My comments are utility-driven without question. I will give you an example. Recently we had a power reduction. We wanted to go in and clean water boxes in our condensers. We agreed to go to 70% power. For that it was "X" number of man-rem to clean those boxes. My boss, being ALARA proactive, wanted to go to 50% power for that 48-hour period. It was hard for me to go to upper management and be against my radiation control manager. When I went in it was 1.6 man-rem savings at 50% vs. 70%. That was like $24,000 TVA ALARA dollars. The loss of megawatt production dollars was $254,000 lost in megawatts. So the 1.6 man-rem was a trade that I had to support -- and that's something different for an ALARA Supervisor -- to support not going down to 50% power. That's the reality we deal in every day. If it was a $25,000 savings and a $50,000 megawatt, then you may argue. But it is hard for me to argue against a $230,000 net savings, earnings that we can make in megawatt production. Everything is not driven from the dollar sense, but you must be competitive as Harvey and Bob said, or people like me and some of you other folks that are at utilities will be unemployed, because if we don't make megawatts competitive, we won't make megawatts.

Baum: Thank you, Floyd. That relates very closely to what Jacques Lochard was saying earlier about the relationship between radiation protection and the costs of production, which incidentally I referred to as the gamma factor in alpha, beta, gamma dollar/rem values in the NUREG report to which Brian referred. That NUREG is out on the ALARA Center table in case anyone wants a pre-publication copy of it. Jacques, would you like to say a few words?

Lochard: I will be very brief, because I think I already delivered my message. I don't want to abuse the time. I think the potential synergy between improvement of radiological protection and improvement of production activities is a key issue for the whole industry. We have to look for new ways of integrating radiological protection within the production process and to stop to see radiation protection like the fireman's station. Living in a nuclear power station or any nuclear installation is living with radiation, and we have to integrate this dimension in the project planning of the installation. I think the right way is to adopt a management approach, a management perspective in dealing with doses as we deal with money, keeping in mind that we have the problem of communicating with the operators and also the public, which is a very important point.

Baum: Thank you, Jacques. Chris?

Wood: I'd like to follow up on Harvey's and Floyd's comments. We are very concerned that, with the pressure to reduce O&M costs, ALARA is going to suffer. We are going to see that, with fewer staff on site, individual exposures are going to increase, and total exposures may increase as well. I'd like to take one of the last slides that Jacques showed in his presentation this morning (see paper 4-6). This curve here, exposure savings against cost. The point about ALARA is that the first things you do are undoubtedly going to be cost effective, they are going to produce savings. An example there is that perhaps a part system decontamination is going to cost $400,000 and save 300 rem. It is obviously cost effective. Now, if you go to the full system decontamination in Harvey's example, for some plants, you know the French plants that can replace a generator for 130 man-rem, that's going to be to the right of the curve. It's expensive and not going to give a great deal of savings. For other plants it will be cost effective. My role at EPRI is to develop tools to help utilities optimize and try to get to that point -- the inflection on the curve here. In my presentation yesterday, I showed a cartoon for hydrogen water chemistry advisor, and that was done for a different reason. It was working out the most cost-effective way of tackling a stress corrosion cracking problem. We are applying the same approach now to radiation protection, and we will be looking at things like cobalt replacement. What is the optimum amount of cobalt replacement? Obviously, it is not cost effective to change out the stellite in valves that are operating perfectly well, but it is cost effective to use cobalt-free alloys if you are going to change out valves anyway because they need replacing.. So that is the approach that we are going to be adopting to develop a very user-friendly computer tool that will allow the utilities to plug in their own plant-specific numbers and work out the optimum course of action.

Baum: Sounds good. Alan, I know that you have had a number of opportunities to do cost-benefit calculations in your work and you've had a lot of good applied experience in addition to your business administrative background. What do you have to say?

Homyk: I guess the point I want to make is that there's not a decision we make, at least at our utility and many others, that doesn't somehow involve the cost-benefit issue. It's really an attitude and a culture at our plant that we are really running a business. Not that we don't want to be safe and excellent, but the bottom line is to keep the plant running, and to keep nuclear power going we need to be cost effective. And that really permeates every decision we make in our life at the plant -- every ALARA decision and every operational type decision. But we need to remember that problems also pose opportunities. I will give you an example. Last outage, I told my health physics manager, "I want to bring ten less techs on site this outage. I don't think we need that many. I think there are tools out there that will allow us to use less staff." So we looked at our present system for job coverage and remote worker wireless monitoring. He said, "I think I can save people through this new technology." I said, "Are you confident enough that you can guarantee me that? In other words, I want to go to the VP and say that we are going to pay for that system in one outage by the ten less techs you brought on site." Well, he had enough confidence to make the promise and deliver even more. We need to think about how funds can be re-targeted to other areas. If you are bringing 100 rental techs on site, maybe you can bring in only 80 and then use the difference to fund some technology for your people -- the remote continuous air monitors, the remote sensors for dose rates, and things like that. We also need less compartmentalization -- why do only chemists take oxygen samples of areas? Why can't the HPs do that? You can do a little bit of on-the-job training, put it on the qualification card, attend a half-hour of classroom training, and know as much as you really need to know to run an oxygen sensor. Why not save the extra man and dose associated with going inside containment under power. Again, Mr. Giordano talked about survey reduction. There are a lot of simple things that can be done. Why are we taking so many surveys? For example, our Unit 1 has been non-operational for 20 years, and if you never once have had any change in gamma rad levels well maybe you don't have to do monthly surveys. There are other obvious, common sense type things that we all can be doing. You may take thousands of air samples a year and never have had any airborne activity in the areas -- so work smart. Prioritize your efforts. There are many simple things we can do. We don't need more money, we just need to work smarter. I'll give you one final example. I used to talk to the maintenance manager about leaks. There would be a film of boric acid on a component and he'd say to me; "I don't want to fix it. It's not cost effective. Why don't you just wipe it off?" I said, "I'm getting dose wiping it off. I'm generating rad waste wiping it off. Because there's a film there, I have to post it as a contaminated area, so I'm generating laundry, anti-contamination clothing, and the film may eventually result in airborne activity -- it's a big cost." So don't think about the cost of doing things, think more about the cost of <u>not</u> doing things. I can't think of many leaks on the nuclear side of a plant that aren't cost effective to fix. That's what we've found, and we trend and track them. We put a drip catcher under every leak as many plants do. But we go beyond this and measure the leak rate. We take a chemist's graduated cylinder, we quantify the leak, and put it in terms of $/year of not fixing that leak. We had a number of expensive leaks about five years ago that have been eliminated. You can get people's attention if you communicate effectively and use common sense. I guess the thought that I want to leave you with is that many of these things, as Mr. Giordano said, reduce cost, foster excellence, and make your job a little easier.

Baum: Thanks very much, panelists. Those were all very important thoughts and comments. Before opening it up for other questions and discussion, I would like to tell Alan that he doesn't realize how many hundreds of thousands of dollars he lost by not being here yesterday. This process may be the most cost-effective thing you can do. Does anyone have a question or a comment for the panel, or would the panelists like to question each other?

Aldridge: One of the costs that really drive ALARA is the public perception. What does the public think of the activities that we are doing. Within the DOE and the Health Physics Society, we are taking a lot of monies and a lot of dedicated time to get out and teach and educate the public. You folks in the commercial nuclear power industry, are you taking those kind of activities in a positive, proactive role? If so, what are you doing?

Unknown: Just one comment. In every decision we make, we have to weigh, in addition to the true radiological consequences the perception consequences, and many times that is what drives the decision. It's a very important part of our decision process.

Wood: I guess the industry's official response will be that the USCEA that was which is now part of the Nuclear Energy Institute did put out the educational information. My reaction is that it almost completely fails. It is regarded as propaganda and is not accepted by the public. My wife says that I would do much better spending a portion of my time going around to the high schools and trying to educate the kids, which I did after the Chernobyl accident. I went to my daughter's class to talk about Chernobyl and what it meant. I have some sympathy for what you are saying.

Homyk: There are a couple of things I am aware of that are being done. Again, at Indian Point 2, we just try to get people into the plant. Sometimes people have the notion that nuclear plants are evil, and if they can just come up and visit the plant, a lot of times that can diffuse fears that they have. We have a steady stream of visitors at our plant. I saw a couple of nice things going on at other utilities. One is that GPU has a tape aimed at not only the workers, but the families of the rad workers, because if you think about it, every one of us touches many people who know that we work at a plant. Their philosophy is to use a tape and a nice little manual to communicate the risks of radiation. The questions people might have are communicated to the workers so they can communicate the answers to their families. Again, think about the contacts all the family members have. So it's a good effective way to communicate. You can almost think about it in terms of disciples getting the information out. The other idea is that Chris Wood is heading up through EPRI is a radiation worker handbook that addresses basic questions people might have. Such as, what is the risk of bringing contamination home? Really simple questions that might be asked by family members. Those are the two things I think would be effective.

Lochard: I would also like to comment on this issue, because globally we have been quite bad within the last decades about communicating about radiological risk. For example, during yesterday's debate about the limits, I mentioned that the majority of the people, not only the public, but workers, are living with the idea that the limit is something like -- you are safe if the limit is respected, and unsafe if it is not respected. This is because the key ideas driving radiation protection management have not been really explained to the public at large and also to the workers. For a few months in France we have this experience where we do extensive training of workers to commit all the operators in French power stations to the new ALARA program which has been launched by EDF and was presented this morning. For most of those who attend the courses, they are just discovering this idea of dealing with radiological risk, the idea of allocation of resources, the problem of risk transfers. If you really explain all the issues in very simple ways, operators become aware that the problem is not just reducing limits and claiming being safe at any cost. We are living in an unsafe environment, and we have to deal with it and manage it in the best way. I think this is a real challenge. If we are publishing ALARA techniques, ALARA models, and all these alpha values, and so on, without communicating with the public and the workers, it will fail.

Baum: You are coming from France, you don't know what an unsafe environment is!

Giordano: I've seen several of the utilities have speakers' bureaus. These are low-cost, highly effective communication activities that members of the utilities take up on their own. As Chris was saying, to go talk to the schools and to get the message out that way. There are open houses at some of the utilities. When I was associated with the DOE world and the

Shippingport decommissioning project, we actually had an Explorer scout troop that came on site, with their parents on occasion, but certainly by themselves, where they had a project that was associated with construction and dismantling and these scouts learned things. But they also, subliminally if you will, understood a little bit more about what was actually happening. It's not something unique to the commercial world. There may be some things that the commercial world is able to do because they don't have some of the governmental restraints on them, or how they are speaking for the government. There is a lot going on in that area. I know that the Health Physics Society has a session on communication with the public that you may also want to look into.

Borst: I wonder if you could expand on these two items. First, if we ask any of the rad workers in our plant, "What is ALARA," you'll get "time, distance, and shielding." When we talk to management about ALARA practices, exposure savings, they want to hear dollars. In almost every case, if we could do the job for less exposure, that means we've done it more efficiently, and that's the dollars that management can really relate to. As a utilities industry, we need to drive our presentations to management in the dollars aspect more so than in straight exposure, which is kind of intangible to begin with. Secondly, a couple of years ago, all the U.S. utilities said that their goals are to be best quartile. Well, obviously, not everybody can be in the best quartile. Some of those have backed off and said "Let's shoot for better than average and go from there" when they realized they couldn't make that. One thing that management failed to pass on to the program managers is that they want us to be best quartile, reduce our exposure by a half, but we don't get anything to do it with. So I hope we can ask our management, when they give us that direction, "how badly do you want me to be best quartile, and how much money are you going to give me to do the kinds of things that we really need to do?" We've all gone through the efficiency phase, reaching our bottoming out point of what we can do with nothing. We need some extra help in automation, remote technology, and things like that.

Baum: Frank Rescek, you also have a degree in business administration. What words of wisdom can you give us?

Rescek: I just had a comment or a follow-on. Commonwealth Edison recently has built the Powerhouse, up by the Zion Station, which is open to the public and scout troops. The public went through there and it's been a tremendous success -- one of the bigger drawing attractions for that type of center in Chicagoland. There are visitors from all over who come there and get to see and learn about various forms of energy and energy production. That's one thing that Commonwealth has done to educate the public. Also, though, I wanted to hear your thoughts on the NRC's role in this area. I think it's very important that, I understand, the regulations are first to ensure that you've met a level of protection that's safe, but then the regulations also don't want to convey a misconceptions of the risk to the public as well. Could you comment on that as well?

Richter: Are you referring to adequate protection and that role vs. when $/person-rem comes into play?

Rescek: I'm not referring to $/person-rem, but, for example, the decommissioning and decontamination rule-making process with the public and the EPA. The first draft that came out on that rule focused on goals going to zero dose and then using the 3 mrem as maybe the standard to show that you are close enough to zero. The whole way that it was written would give the public a misperception that it makes sense to try to go to zero.

I think that the slant of the language may confuse the public that one could achieve zero risk.

Richter: Unfortunately, I have not been involved in that action. I wish that Don Cool were still here. I'm not sure if any of my colleagues from the NRC might have something to add on that.

Baum: Frank Congel is going to say a few words.

Congel: The words that you pointed out that were indeed in the earlier draft have been changed. In fact, one of the reasons Don is not here is because we are going through another review process. The proposals for decommissioning standards have changed, but it is a number, it is measurable. The values, whether they will be approved or not, I can't say, because they haven't gone to the Commission yet. The way the values are going to be stated is a target value of 3 mrem would be a de facto compliance with an ALARA standard. However, the standard that you would have to meet above that before you did an ALARA analysis, is going to probably be on the order of 15 mrem. This is carefully coordinated with EPA. I can't emphasize enough that this is preliminary and is a staff proposal that has not been formally approved. The point is that your comment is important and is being regarded by the NRC in the same light, and that the word changes are being proposed right now.

Baum: John Connelly from DOE has a question, and I have to make sure to let him ask it, because he is our project manager on the DOE ALARA Center effort, and he has been very good about not asking questions a couple of times because we were getting near the end of a session. Please, John, go ahead.

Connelly: Actually, this is a comment, and something some of the people may not know. A lot of money is being spent because of public perceptions. About 10 years ago, I was called in by a local middle school teacher to teach a class to 7th and 8th graders because a TV show about "the day after the bomb was dropped" scared his students. When I looked at his textbook, I could see why he was disturbed about trying to teach his students. The school text book was anti-nuclear. It showed a mushroom cloud and discussed the LD-50 (lethal dose to 50% of the population). The book then showed a nuclear power plant and indicated, "well, it's safe -- maybe." The only thing I had going for me was that I was a health physicist, and that I had worked at that particular plant for 10 years, so I could stand up there and talk with some credibility to his students. I brought a portable frisker (contamination monitor), a sealed source, and some instructional slides. I turned the students around by telling and demonstrating to them what was real. I also told them about the positive uses of radioactive material in various fields. I think a lot of the school books are stacked against the nuclear industry, and the schools are training these young people to be anti-nuclear because of a lack of accurate text books. There have been studies done that show the press to be typically anti-nuclear. Some of the press act this way because they are dealing with a technical subject that they' don't know anything about. If you read the average article in the newspaper, they will misuse the information supplied to them.

Secondly, I took the Myers-Briggs Type Indicator test at a Health Physics Society meeting. What I learned is that health physicists are typically introverts, and only 1% of the population matches this profile. Since we don't match with the average person, we don't click on this level. Therefore, it's no wonder that health physicists have difficulty communicating with the general public, especially when they are talking about an emotionally charged subject.

Thirdly, I took a risk management course and I learned that the average technical person that takes a risk test will come out pretty close to the correct risk. The more familiar they are with a particular subject, the more accurately they assess the associated risk. If you compare them to the general public, it's widely different. The general public changes the risk by orders of magnitude. If you give them a risk number associated with an activity, and there is some obvious benefit to them, such as driving a car, they automatically lower the risk by about 2 orders of magnitude. If they do not perceive an immediate benefit to themselves, such as the construction or operation of a nuclear facility, they increase the risk by about 2 orders of magnitude. Suddenly, there is a great divergence between actual and perceived risk that most of us don't realize. So, it doesn't matter how technically correct you are, the general public will change actual risk to their own perceived risk depending upon their own personnel needs.

Zodiates: Nuclear Electric is nowadays facing the economic realities in the U.K., it is one of the daughter companies from CEGB. We are facing the same problem. We have to educate the public to understand the risks and the benefits of nuclear power. For quite a long time now, more so recently, each of our power plants have visitor centers which are geared mainly toward the younger people -- children and teenagers -- with a lot of touch screens and a lot of models, so schools from the area can visit the centers and they can become more familiar with our power plants. In addition to that, we have spent some time producing educational packs for primary schools to assist them with their curriculum on energy production, and therefore, try to educate the youngest generation. Recently, we started carrying out surveys of the public to find out which parts of the public react against nuclear power and therefore target other types of complaints and alleviate their fears. The most important item, at least in my opinion, is not doses to the operators, but discharges and our long-term waste disposal. You suffer from the same problem, because looking at yesterday's newspaper that was delivered in the hotel, there is a big article about waste disposal; if I remember correctly, it was Minnesota State. Those are the items that the public becomes aware of and fear from nuclear power. It is time for our politicians to solve this problem and therefore alleviate the fears of the public.

Haynes: I would like to pick up a bit on some of my comments yesterday. Ontario Hydro is a large nuclear utility that has made a major investment in self-protection philosophy for a long, long time. It has a lot of experience in training its workers extensively in radiation protection, in the fundamentals, including ALARA, the biological basis of it, and the benefits of it. Our labor unions are very much involved in formulation of policy in radiation protection and ALARA specifically, in dose limitation, and I just want to reemphasize the importance of having buy-in from your work force. I think it is well worth the investment. I recall that a couple of years ago there was a public debate in the city of Toronto in which some politicians had some idea about declaring Toronto a nuclear-free zone or some such nonsense, and Ontario Hydro was front and center in that debate, both the management and the labor unions. Without a doubt, the most convincing advocate of the safety of nuclear power was the labor leaders. Clearly, the public much more readily identified with an individual like that speaking in favor of the nuclear option. They see him as a far more credible source than corporate management. Although sometimes it's very painful to deal with organized labor in formulation of policy on specific issues in radiation protection, overall I think it is well worth the investment.

Baum: We have just one or two more minutes. Would any of the panelists like to make a concluding comment?

Giordano: One thing that I did not hear that occurred to me was when I was listening to the animated dialogue at the 10CFR20 workshops, I was pleased to hear Jim Wigginton and company talk about the intended unified approach to administering this new regulation to avoid the things that have occurred in the past where well meaning, well intentioned local inspectors were establishing criteria at ground levels that grew into little cost-benefit types of actions that were implemented by the plant. I know that process is ongoing. My point is that we have to be careful, and we have to think about what is suggested, and to make sure that it is right for safety and right for the business. When I hear an ALARA individual tell me within the last couple of months that, "Yes, the law is that you don't have to badge someone if they are 10% of the limit or 500 mrem, but my local inspector wants that number to be 300," it makes me wonder how far we are headed down that path. In the continuous improvement verbiage, we have to keep a handle on ourselves and make sure that we are not ratcheting ourselves into increased costs to provide safety with very little benefit in that area. I'll pass that on as a concluding remark.

Baum: Thank you very much. The message that I'm getting from this is that it's not enough for us to be excellent, but we are going to have to educate the workers and the public to the fact of what is excellence, and how close are we to it. Thank you, panel.

SESSION 6

DISCUSSIONS

Chair:

Joel Rabovsky

OCCUPATIONAL DOSES AND ALARA - RECENT DEVELOPMENTS IN SWEDEN

Thommy Godås and Christer Viktorsson
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ABSTRACT

Sweden has traditionally experienced very low doses to workers in the nuclear industry. However, this trend has since last year been broken mainly due to significant maintenance and repair work.
This paper will describe occupational dose trends in Sweden and discuss actions that are being implemented to control this new situation.

INTRODUCTION

Nuclear power in Sweden is generated by 12 reactor units at four sites. Nine of the reactors, all BWRs, were supplied by the Swedish company ASEA-Atom, now ABB-Atom. The remaining three, which are PWRs were delivered by Westinghouse.
Nuclear accounts for half of the electric power generated in Sweden. The 12 reactor units have a total installed capacity of approximately 10 GWe.

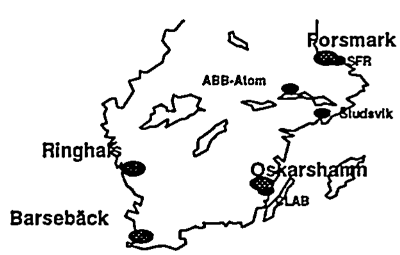


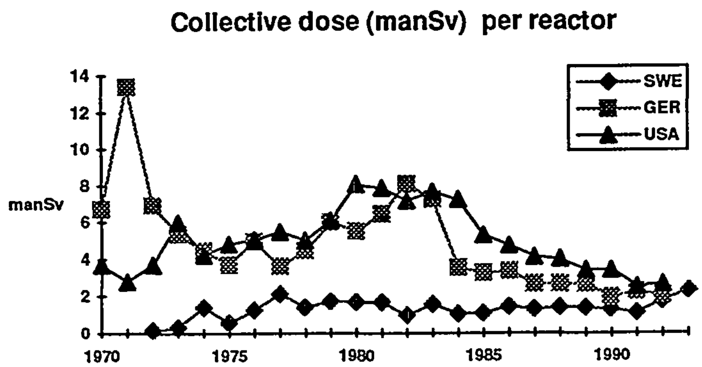
Figure 1. Nuclear power facilities in Sweden.

Unit	Type	MWe	Commissioned	Owner
Barsebäck 1	BWR	600	1975	Sydkraft
Barsebäck 2	BWR	600	1977	Sydkraft
Forsmark 1	BWR	970	1981	Forsmarks
Forsmark 2	BWR	970	1981	kraftgrupp AB
Forsmark 3	BWR	1155	1985	
Oskarshamn 1	BWR	440	1972	OKG AB
Oskarshamn 2	BWR	600	1974	OKG AB
Oskarshamn 3	BWR	1160	1985	OKG AB
Ringhals 1	BWR	820	1976	Vattenfall
Ringhals 2	PWR	860	1975	Vattenfall
Ringhals 3	PWR	915	1981	Vattenfall
Ringhals 4	PWR	915	1983	Vattenfall
Swedish NPPs		10000		

Figure 2. Nuclear power plants in Sweden.

OCCUPATIONAL DOSES IN SWEDEN

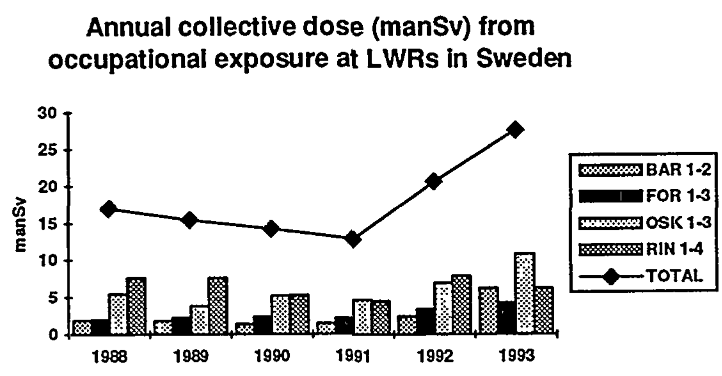
Sweden has traditionally experienced very low doses to workers in the nuclear industry. This can be seen in a comparison with the collective doses in other countries operating light water reactors.



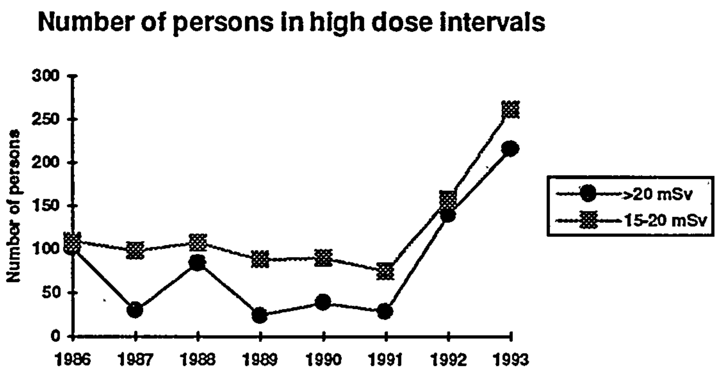
The decrease of doses that we have seen in several countries during the 80s, can partly be the result of new reactors taken into operation but probably more as the result of the "new view" on radiation protection.

We understand that the ALARA-way of thinking was early accepted in many countries and have been developed even more during the years.
Of course, most of the countries started their dose reduction programs when the dose levels were high and from which it must have been relatively easy to reach spectacular decreases.

Unfortunately, in Sweden we have seen the opposite development in recent year. From the positive trend up to 1991, we found the collective dose for 1992 to be "all time high", 20,5 manSv and it become even worse in 1993, namely 27,5 manSv.



We have also seen an increasing number of persons with relatively high doses, for example in 1993 there were 216 persons exceeding 20 mSv, compared to 38 in 1991.



The reasons for increasing doses in Sweden can be

- Ageing reactors requiring significant maintenance and repair works; and
- Increasing safety requirements resulting in extended test programs.

Additionally, some "routine" may have gone into the radiation works resulting from the many years of "easy" operations.

Of course, the dose increase that we have seen is most likely the result of a combination of different causes, but it is obvious that the lack of goals for the radiation protection activity contributes strongly to this increase.

In spite of the many large and difficult repair works which have taken place at the reactors during 1993 and particularly during 1992 we would argue that most of the doses can be related to "ordinary work" during the outages.

Let us look back to the steam generator replacement at Ringhals 2 in 1989, a work which was carried out with success following careful planning. The collective dose from the replacement became as low as approximately. 3 manSv, at that time a very low collective dose for such a significant works. However, today is such a dose common for similar works, some have been even more successful to keep the dose at a low level, but we believe Ringhals showed the way how to handle this kind of a problem.

During the spring of 1993, approximately 400 pipe bends had to be replaced at Oskarshamn 1 because of problem with stress corrosion. Also, this job was carried out after careful planning. Combined with a successful decontamination which resulted in a decontamination factor DF of 20, the collective dose could be held on a very low level.

What the result will be from the current renovation work at Oskarshamn 1 is too early to say, but the dose reduction actions taken so far have been extremely successful. They include a large decontamination of the reactor vessel, the main recirculation loops and several other parts of the reactor system, as well as shielding of the inner wall of the reactor vessel, especially at the core region.

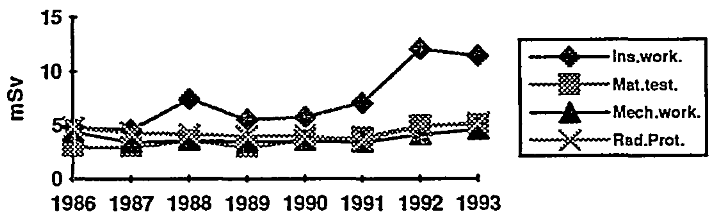
In July 1992, a safety valve in the automatic depressurization system at Barsebäck 2 opened inadvertently at 30 bar and blew stream to upper drywell causing a simultaneous clogging of both trains of the emergency core cooling system. Five of the Swedish reactors, those with external recirculation pumps and small stainer areas, were later that year taken out of operation.

At four of the five reactors a decision was taken to replace most of the fibre insulation with metallic insulation. At the fifth reactor fibre-glass insulation was chosen. All of them increased their stainer areas.

From a radiation protection point of view, we have had some unsatisfactory experience from the use of metallic insulation, it requires some time-consuming and troublesome handling.

We have also seen a considerable increase in the individual doses to insulator personnel. Therefore, we had some doubts in the whole operation.

Average individual dose for different worker cateogries



However, the replacement took place and it was not a success neither from a radiation protection nor a technical point of view.

Lack of planning, wrong drawing support combined with an extremely tight time schedule gave a collective dose of approximately 7 manSv in total for the four reactors installing metallic insulation.

ACTIONS TO TURN THE TREND

At the Radiation Protection Institute (SSI), we cannot accept a prolonged negative dose trend, and therefore, we have worked hard to find countermeasures to turn the trend.

In the new regulations on occupational exposure, which will be issued in the very near future, new requirements were included. First of all we decided to introduce a new individual dose limit, 100 mSv in 5 consecutive year in addition to the annual individual dose limit which is 50 mSv.

We have also required an extended education and training program in radiation protection, addressed especially to foreman and team-leaders, working for the utility as well as for contractor. We believe that this program will increase the understanding and motivation of the personnel to more heavily engage in dose reduction.

Additionally, we believe in an ALARA, or work management, approach, i e where the utilities systematically review their strategy towards radiation protection and develop goals in the area of occupational doses. The SSI has initiated such a review and in discussion with the utilities we have asked them to develop plans for dose reduction based on the ALARA way of thinking. This review should result in individual and collective dose goals concrete means to reach those goals as well as system for feedback analysis. Finally SSI requires an organisational structure to manage and monitor the occupational dose control program.

The SSI established long ago, an ambition level for collective doses to workers in the nuclear field in Sweden. This level, which was set to 2 manSv/GW installed electricity, is emphasized even more in the new regulations and the Institute now requires the utilities to plan all their works according to this level.

Using our research funds, we have recently started a significant development program in the field of dose reduction. The Swedish "reactor maker" ABB-Atom is on one behalf studying the reasons for the increasing dose levels, estimating the expected dose situation during the years to come, as well as giving advice on concrete actions to reduce occupational doses.

Moreover, we support the international cooperation in this field, and therefore, we participate in the NEA ISOE program. In order to improve our possibility to exchange information with other countries, we have decided to adjust our regulations for reporting occupational doses. This will mean that we plan to include parts of the ISOE reporting system in the Swedish regulations. We are also reviewing the electronic dose recording software (ASPIC), development by the ISOE system, with the view of introducing it in Sweden.

FINAL REMARKS

We are convinced that the negative trend we have experienced in the area of occupational doses will be broken already in 1994. This, however, will require hard work, including increased emphasis on ALARA and on ways to manage work in radiation fields. In order to be effective, this "culture" in radiation protection will require the cooperation between various professional groups with the utilities as well as a continuous dialogue between the utility and the regulator.

Author Biography

Thommy Godås is a Senior Radiation Protection Physicist at the Division for Nuclear Inspection and Emergency Preparedness at the Swedish Radiation Protection Institute. Mr. Godås is also responsible for the Institute's supervision of the Oskarshamn Nuclear Power Plant, which includes 3 BWR reactors and the Central Storage for Spent Nuclear Fuel (CLAB).

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ALARA AND PLANNING OF INTERVENTIONS

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INTRODUCTION

The implementation of ALARA programs implies integration of radiation protection criterion at all stages of outage management. Within the framework of its ALARA policy, Electricité de France (EDF) has given an incentive to all of its nuclear power plants to develop "good practices" in this domain, and to exchange their experience by the way of a national feed back file.[1] Among the developments in the field of outage organization, some plants have focused on the planning stage of activities because of its influence on the radiological conditions of interventions and on the good succession of tasks within the radiological controlled areas. This paper presents the experience of Chinon nuclear power plant.

At Chinon, we are pursuing this goal through careful outage planning. We want the ALARA program during outages to be part of the overall maintenance task planning. This planning includes the provision of the availability of every safety-related component, and of the variations of water levels in the reactor and steam generators to take advantage of the shield created by the water. We have developed a computerized data base with the exact position of all the components in the reactor building in order to avoid unnecessary interactions between different tasks performed in the same room. A common language between Operation and Maintenance had been established over the past years, using "Milestones and Corridors".

A real time dose rate counting system enables the Radiation Protection (RP) Department to do an accurate and efficient follow up during the outage for all the "ALARA" maintenance tasks.

Planning Jobs on the Safety Related Components Taking into Account the Cold-shutdown Technical Specifications: Developing a Common Language Between All the People.

Detailed Cold Shutdown Technical Specifications have been developed and are to be strictly followed during an outage. They address the availability of on-site and off-site supply of power, emergency core cooling system (ECCS), reactor water level, reactor heat removal, reactor spray system, etc.

Before an outage, and as early as possible, typically six months before a given outage, each maintenance department gives its own maintenance program to the planning department. As early as possible, typically four months before the start of a given outage, we take these programs into account with the Technical Specifications and we establish:

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[1] L. STRICKER, ALARA Policy at Electricité de France, Third International Workshop on the Implementation of ALARA at Nuclear Power Plants, BNL, Long Island, May 8-11, 1994.

- The so-called "milestones" are key moments in the outage, such as the first opening of the reactor primary circuit, the end of defuelling, and the start of Mid Loop Operation, etc. These milestones are labelled using letters, A, B, ... Z. Generally speaking, each major alteration in reactor water level corresponds to one milestone, such the isolation of a specific ECCS safety line or electrical power source.

- The so called "corridors" between two milestones. To every maintenance task, a corridor is given and this task can only be performed in this given corridor. For example, corridor JK, FN..., so that every foreman will know when a specific task should be performed.

The very important aspect of this is that everybody, from the control room operator to the valve mechanic will use the same language, during the outage long, and they will understand each other far better.

Figure 1 is an example of such planning.

Establishing the "Shuttle Notes" So Every Technician and Job Specialist Can Explain His Normal and Specific Logistic Needs.

Working "On Line":

Over the last two years we have been developing an "on line activity" concept, which means that maintenance technicians are made fully accountable for a specific maintenance task. This includes performing a safety risk analysis, establishing the maintenance procedure and various paperwork, contracting with the help of the bargaining division, meeting with the other workers needed to carry on the activity, then monitoring performance on the field, and recording the experience for future outages. Furthermore, we request each of them to prepare and follow the job, not only considering safety, quality assurance, cost effectiveness, and also addressing the radiation protection and industrial safety side of the activity. These people have all the needed background and tools on hand to do it! Part of this includes preparing the "shuttle notes," which are a communication tool.

These shuttle notes list all the logistic side of a maintenance activity as well as its location:

- necessary scaffolding
- radiation shields to be put in place
- needs for in service inspection techniques after welding
- the actual location where the maintenance task will take place
- all of this taking into account the recorded know-how from previous outages

All of these tasks are written using diagrams representing the Reactor Building floors with the exact location of the job. To help technicians in establishing these, they have been given Reactor Building maps, floor by floor, where all the pieces of equipment are precisely located. Each of the maintenance specialists, such as valve mechanics, primary pump mechanics, I&C technicians, have a map of the Reactor Building floor indicating where they are going to work, together with all their needs.

Figure 2 is an example of such a diagram.

Figure 1

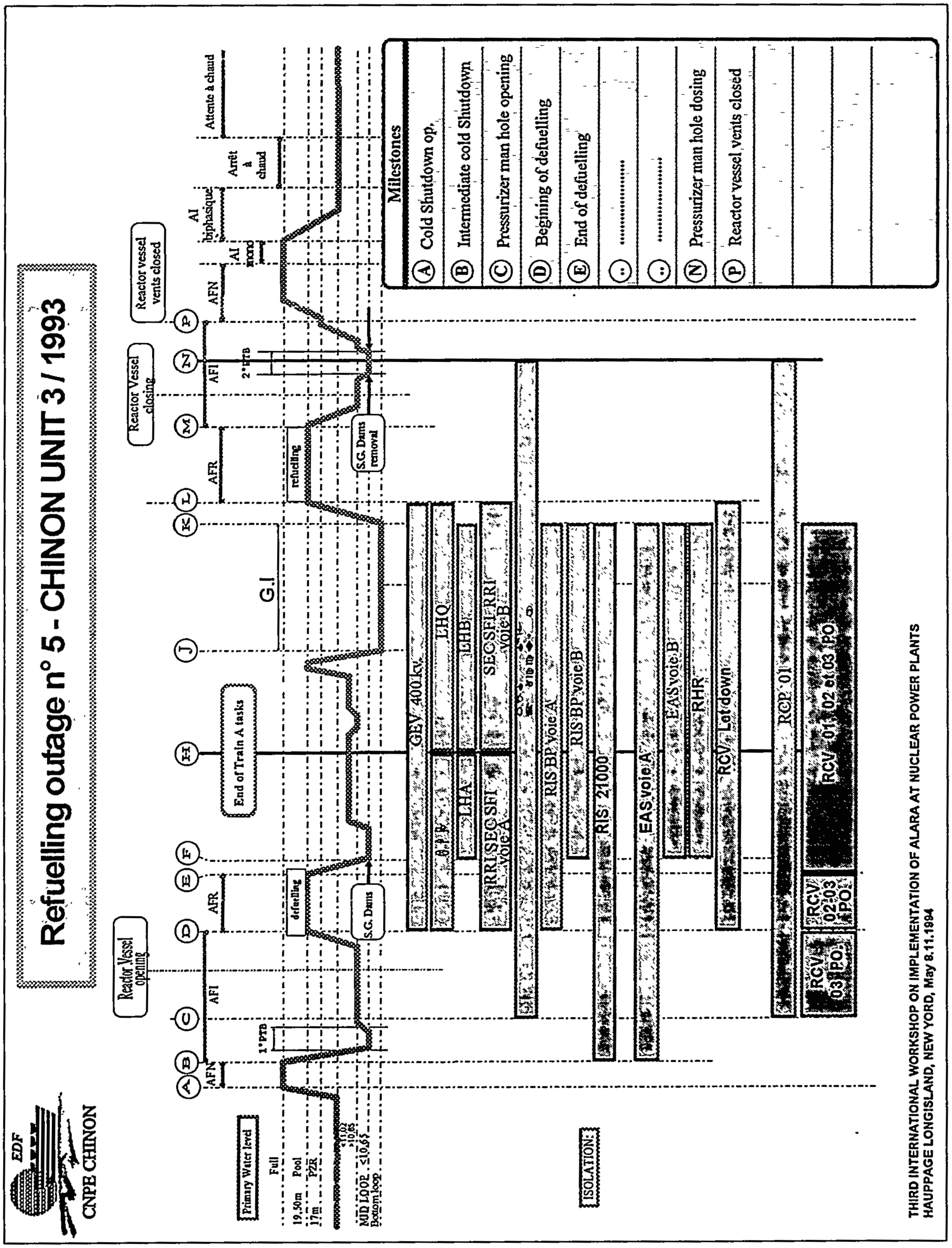
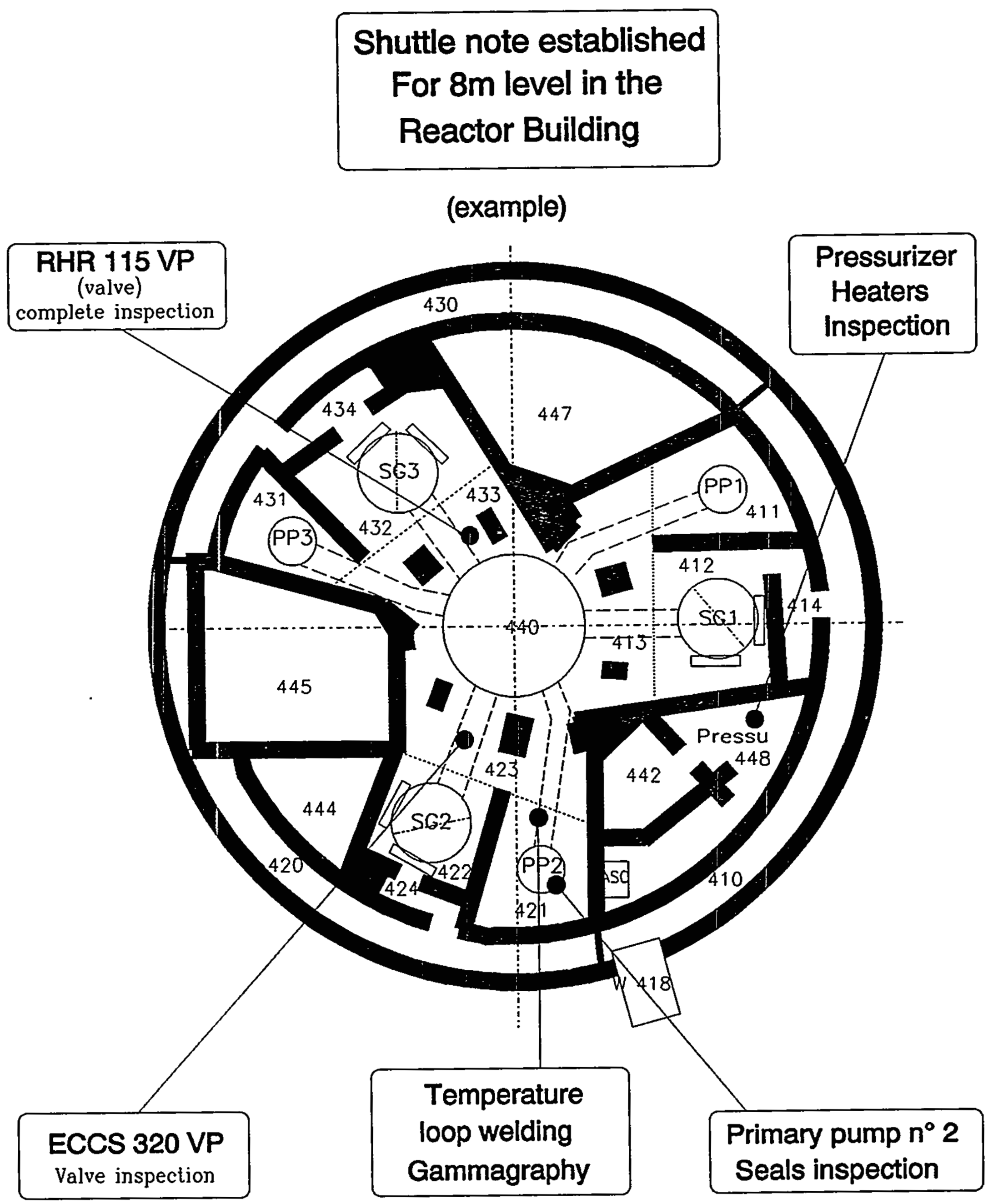

Refuelling outage n° 5 - CHINON UNIT 3 / 1993
EDF
CNPE CHINON
Primary Water level
Full
19,50m Pool
17m P2R
Reactor Vessel opening
Reactor Vessel closing
Reactor vessel vents closed
End of Train A tasks
S.G. Dams
S.G. Dams removal
defuelling
refuelling
AFN
AFI
AFR
AI mono
AI biphasique
Arrêt à chaud
Attente à chaud
1° PTB
2° PTB
G.I
ISOLATION:
GEV 400 kv
LHQ
LHA
LHB
SEC SFI RRI voie B
RRI SEC SFI voie A
RIS BP voie A
RIS BP voie B
RIS 21000
EAS voie A
EAS voie B
RHR
RCV Letdown
RCP 01
RCV 01 02 et 03 PO
RCV 02-03 PO
RCV 03 PO
Milestones
A Cold Shutdown op.
B Intermediate cold Shutdown
C Pressurizer man hole opening
D Begining of defuelling
E End of defuelling
N Pressurizer man hole dosing
P Reactor vessel vents closed
THIRD INTERNATIONAL WORKSHOP ON IMPLEMENTATION OF ALARA AT NUCLEAR POWER PLANTS
HAUPPAGE LONGISLAND, NEW YORD, May 8.11.1994

Figure 2



Shuttle note established
For 8m level in the
Reactor Building
(example)
RHR 115 VP
(valve)
complete inspection
Pressurizer
Heaters
Inspection
430
447
434
SG3
433
PP1
411
431
432
PP3
412
414
SG1
440
413
445
Pressu
448
423
442
444
SG2
420
422
PP2
424
421
410
W 418
ECCS 320 VP
Valve inspection
Temperature
loop welding
Gammagraphy
Primary pump n° 2
Seals inspection

The Planning Department Collects All These Shuttle Notes in Order to Establish the Final Planning for Every Maintenance Department.

This is a very important step, as all the work that cannot be performed at the same time and at the same place will be obvious to the planning technicians. They now have very valuable documents to work on to plan all the jobs in the reactor building, and they are ready for the next step, which is to organize meetings with all the support people, i.e., radiation protection, scaffolding, shielding technicians. During these meetings, they establish:

- the kind of shielding to be used, when to put it in place, how to set it for the maximum advantage, and to minimize the dose rates.
- which jobs are chosen to conduct a more specific ALARA process, based on experience from previous outage at Chinon Nuclear Power Plant (NPP) or from other EDF plants.
- the detailed planning using the safety "corridors."

RESULTS

This common language used among all the employees at the plant has allowed us to achieve promising results, through better assessment of when to perform each task, and not only the main ones.

For example :

1. In March 1994, deinsulation of all the primary circuit prior to a 10-year hydro test costed us only 8.6 man-Sv instead of an usual 300 man-Sv because of carefully choosing the time to perform it.
2. Between 1992 and 1993, the dose rate for a similar annual refuelling outage was reduced by 30% (from 2.3 man-Sv to 1.6 man-Sv).

   We hope to get a 50% reduction by pursuing the same approach to all the tasks.

Brief Introduction to the Chinon Power Station

The Chinon Nuclear Power plant is a 4 Unit 900MWe Pressurized Water Reactor (PWR) power station owned and operated by EDF, the utility which operates the 56 nuclear units in France. The first Chinon PWR unit was put on line in 1983, the last one in 1988.

An overall availability factor of 81% was achieved in 1993 for the four units. A typical cycle between refuelling outages lasts 10 to 12 months. The outage lasting 45 days (average value for 1993, down from 52 days in 1992) with extensive controls performed on safety related equipment. The overall performance is to be considered as average as compared to other French PWRs.

In 1993, the total radiation exposure reached about 9 man-Sv (900 man-rem) with no so "clean" units: we experienced some control rods clad failures in 1990 that polluted the unit 1 primary circuit with "Silver 110," and some valve stellite seats induced "hot spots" on Unit 2 in 1993. Our current goal is to achieve a utility wide goal of 1.6 man-Sv per unit per year by 1995. Some newer plants (1300 MWe PWRs) have achieved promising results in that respect, such as a Golfech Unit with a 0.6 man-Sv (60 man-rem) refuelling outage.

Author Biography

Alain Rocaboy is currently Plant Manager for two PWR 900 MWe Units at Chinon Power Station in France. Before joining Chinon in 1991, he has been working at the Blayais Nuclear Power Plant for 13 years where he was successively in charge of operations, Deputy Plant Manager and Safety Superintendent. He started his career within EdF in 1968, at an experimental Heavy Water-Gas Cooled Reactor in Brittany. He has a Professional Engineering degree from the French National School of Electronic Engineers of Brest.

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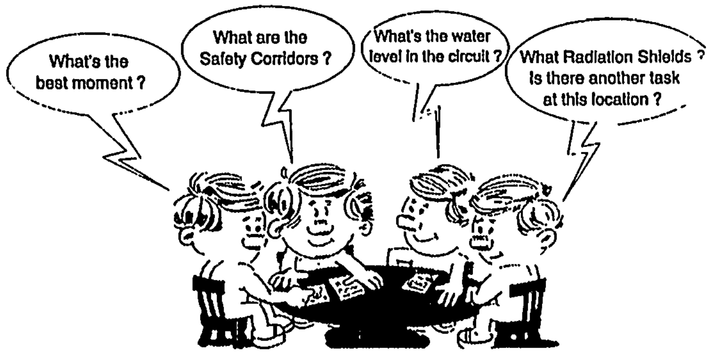
ESTABLISHING OUTAGE'S SCHEDULE

SHUTTLE NOTES



WHAT'S AN OUTAGE ?

A MAINTENANCE PROGRAM

- A SERIES OF OPERATIONAL ACTIVITIES
- EQUIPMENT REQUALIFICATION POST MAINTENANCE TESTING
- SPECIFIC MAINTENANCE TASKS

WHAT ARE THE CONSTRAINTS ?

NUCLEAR SAFETY FOR THE GENERAL PUBLIC

SAFETY AND RADIATION PROTECTION FOR PLANT EMPLOYEES
TIME LIMITS
COST

↓

OPTIMIZING SCHEDULING ACTIVITIES

OUR RESULTS

REMOVAL INSULATION FROM PRIMARY CIRCUIT 10-YEAR HYDRO-TEST MARCH 94	
ABSORBED DOSE	
Without Chinon's approach	With Chinon's approach
300 Man mSV	86 Man mSV
→ - 75%	

	ABSORBED DOSE	
	1992	1993
SIMILAR ANNUAL REFUELLING OUTAGES	2,3 Man SV	1,6 Man Sv
	→ - 30%	

PEACH BOTTOM ATOMIC POWER STATION RECIRC PIPE DOSE RATES WITH ZINC INJECTION AND CONDENSER REPLACEMENT

David C. DiCello, Andrew D. Odell, and Todd J. Jackson
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SUMMARY

Peach Bottom Atomic Power Station (PBAPS) is located near the town of Delta, Pennsylvania, on the west bank of the Susquehanna River. It is situated approximately 20 miles south of Lancaster, Pennsylvania. The site contains two (2) boiling water reactors of General Electric design and each rated at 3,293 megawatts thermal. The units are BWR 4s and went commercial in 1977. There is also a decommissioned high temperature gas-cooled reactor on site, Unit 1.

PBAPS Unit 2 recirc pipe was replaced in 1985 and Unit 3 recirc pipes replaced in 1988 with 316 NGSS. The Unit 2 replacement pipe was electropolished, and the Unit 3 pipe was electropolished and passivated. The Unit 2 brass condenser was replaced with a Titanium condenser in the first quarter of 1991, and the Unit 3 condenser was replaced in the fourth quarter of 1991. The admiralty brass condensers were the source of natural zinc in both units. Zinc injection was initiated in Unit 2 in May 1991, and in Unit 3 in May 1992.

Contact dose rate measurements were made in standard locations on the 28-inch recirc suction and discharge lines to determine the effectiveness of zinc injection and to monitor radiation build-up in the pipe. Additionally, HPGe gamma scans were performed to determine the isotopic composition of the oxide layer inside the pipe. In particular, the specific activity ($\mu Ci/cm^2$) of Co-60 and Zn-65 were analyzed.[1]

The results of the Unit 3 measurements after 2.8 effective full power years (EFPY) of operation on the new recirc pipe show dose rates higher than expected for zinc plants (164 mR/hr vs. 120 mR/hr projected). The latest measurement was made after running 1 cycle with a new Titanium condenser and zinc injection. On Unit 2 the latest dose rate measurements were made with 2.4 and 3.5 EFPY of operation with the 3.5 EFPY measurements post condenser replacement and zinc injection. Dose rates on the Unit 2 pipe continue to be in the typical zinc plant range (114 mR/hr average).

The Zn-65 and Co-60 specific activities on reactor recirc piping and concentrations in RWCU influent water are used to monitor zinc injection system impact on dose rates. Zinc is currently being injected at a rate of .15 ppb in feedwater (which is approximately 180-220 grams per week) to achieve 2-5 ppb in reactor water. This injection rate is based on the stable zinc concentrations present in feedwater prior to condenser replacement. GE recommends injecting zinc up to 0.6 ppb in feedwater for non-zinc plants (which correlates to around 5-10 ppb stable zinc in reactor water). PECo is investigating the optimum zinc injection rate in both Peach Bottom units to address the upward trend in dose rates on Unit 3.

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[1] Data collection and analysis provided by Radiological & Chemical Technology, Inc., 1700 Wyatt Drive, Suite 16, Santa Clara, CA 95054.

Authors' Biographies

David C. DiCello is Manager, Radiological Engineering, at the PECO Energy Company's Peach Bottom Atomic Power Station (PBAPS). He manages the overall implementation of the ALARA program at the station and supervises a staff of six (6) Radiological Engineers. He previously served as the Health Physics Technical Support Supervisor at PBAPS and as an in-plant Radiological Engineer at PECO Energy's Limerick Generating Station. Prior to joining PECO Energy, he worked as a Corporate Radiological Engineer for Long Island Lighting at the Shoreham Nuclear Power Station. Previously he worked at Princeton University as the Assistant University Health Physicist. He has a B.S. in Biological Services from the University of Pittsburgh and an M.S in Radiological Health from the University of Pittsburgh - Graduate School of Public Health. He is both comprehensively and power reactor certified by the American Board of Health Physics (ABHP).

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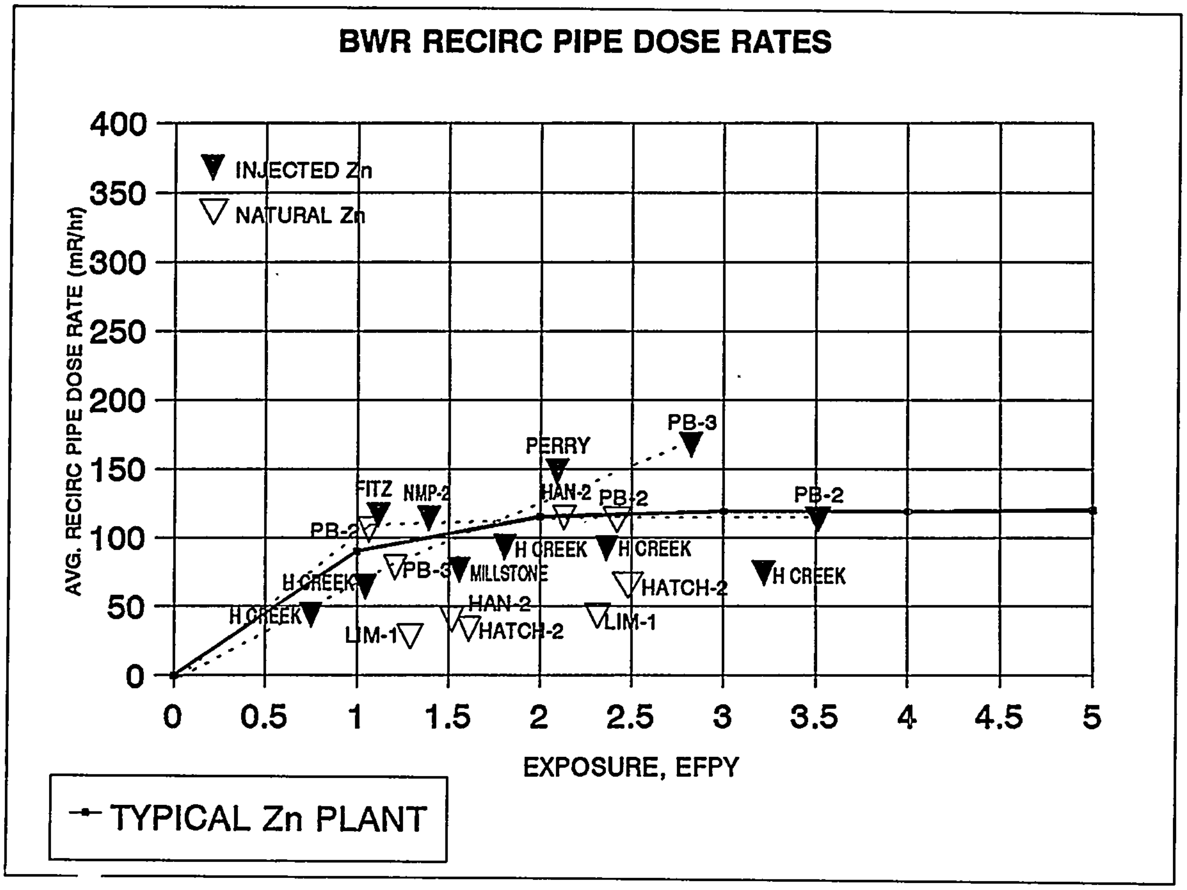
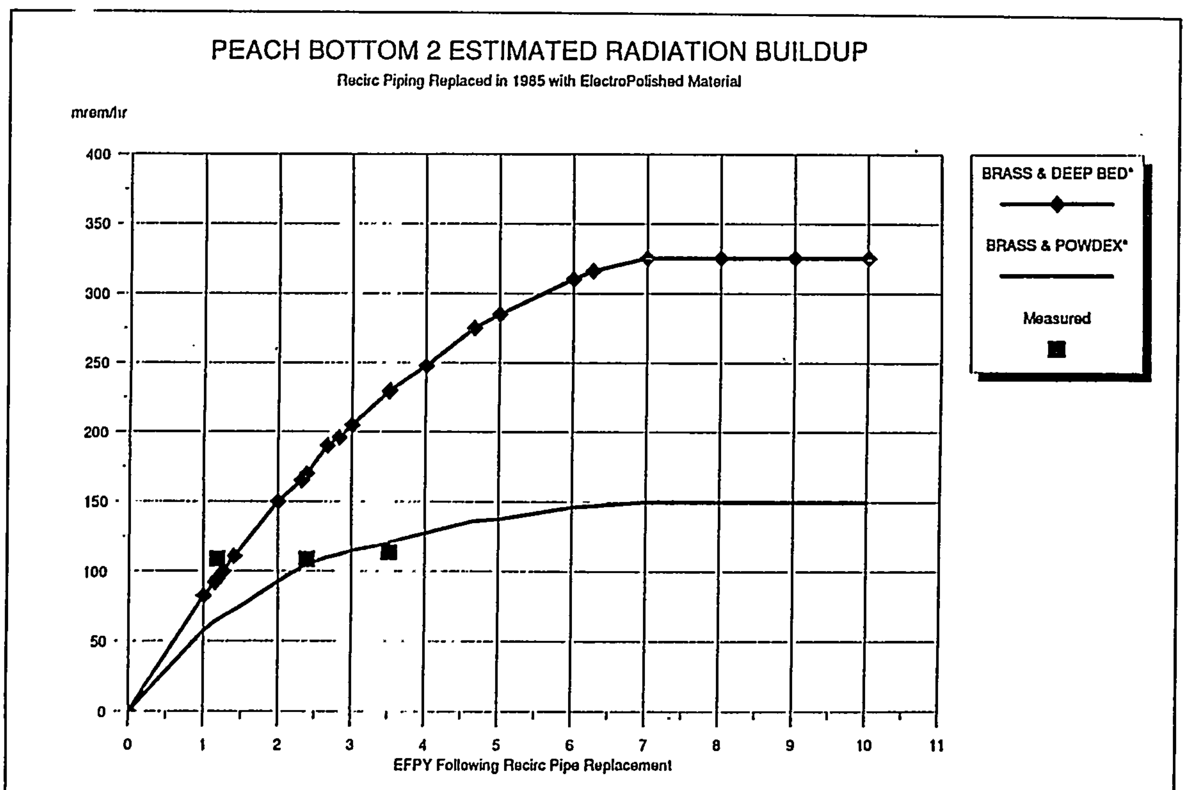
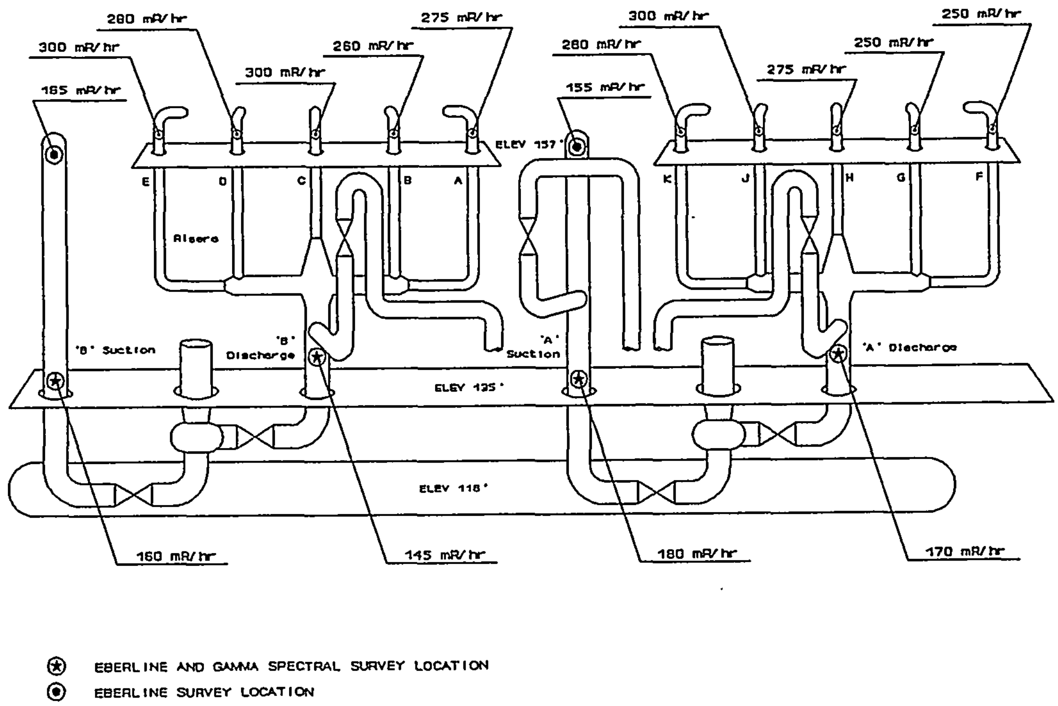
Todd J. Jackson is an Engineer in the Technical Services Branch of PECO Energy's Nuclear Generation group. He is responsible for coordinating radiation buildup and zinc injection effectiveness monitoring for Peach Bottom and Limerick stations. He was formerly the Chemistry Manager at Limerick station from 1988-93. Prior to joining PECO, he was Radiological Engineering Manager for Westinghouse Radiological Services and Hydro Nuclear Services. He began his career as a Radiation Specialist with the USNRC in Region I. He earned BS and MS degrees from Rensselaer Polytechnic Institute in biology and environmental engineering/radiological health.





*Ref EPRI NP-4474, "BWR Radiation Field Control Using Zinc Injection Passivation", March 1986

Unit 3 2.8 EFPY
Gamma Scan and Dose Rate Target Locations



Unit 2 2.4 and 3.5 EFPY

Comparison of Recirculation System Dose Rates

Elev.	Azimuth, degree	Survey Point No.	Location	Dose Rate, mR/hr March 1991	Dose Rate, mR/hr Nov. 1992
135'	0	7	A Suction	90	130
135'	270	18	A Discharge	115	90
135'	180	15	B Suction	110	105
135'	90	12	B Discharge	120	130
AVERAGE - 28" RECIRCULATION PIPE				109 ± 13	114 ± 20
157'	30	23	A Riser	200	200
157'	60	24	B Riser	180	220
157'	90	25	C Riser	180	200
157'	120	26	D Riser	190	200
157'	150	27	E Riser	180	270
157'	210	29	F Riser	180	250
157'	240	30	G Riser	200	200
157'	270	31	H Riser	170	220
157'	300	19	J Riser	210	250
157'	330	20	K Riser	220	200
AVERAGE - 12" RISERS				191 ± 16	221 ± 26

Unit 3 1.4 and 2.8 EFPY

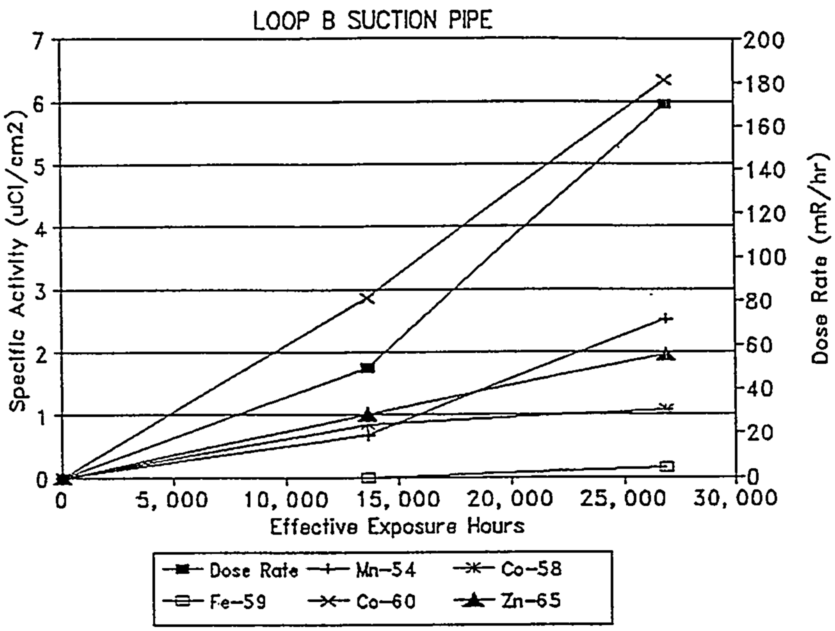
HISTORICAL SPECIFIC ACTIVITY BUILDUP AT PEACH BOTTOM-3
FOLLOWING RECIRCULATION PIPE REPLACEMENT

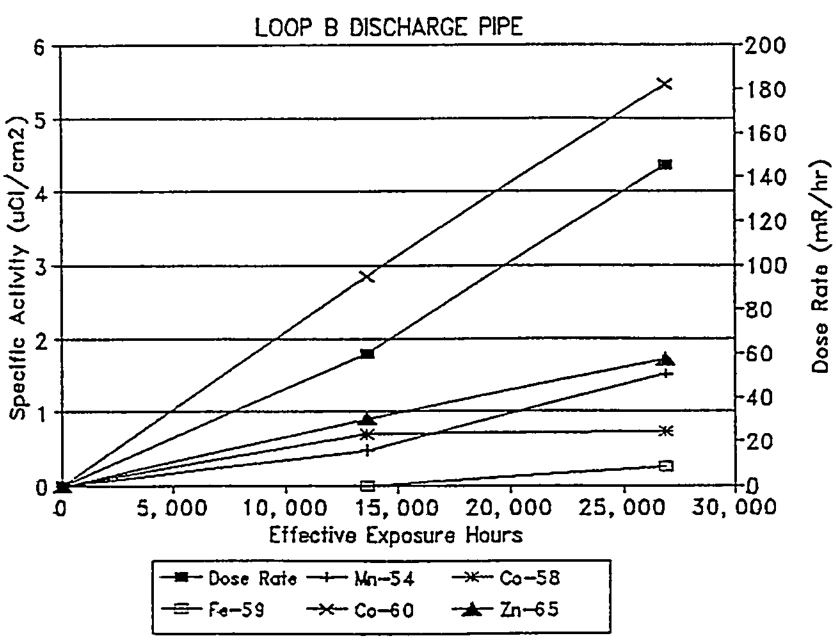
Target Location	Dose Rate, mR/hr	Specific Activity[1], $\mu Ci/cm^2$					
		Mn-54	Co-58	Fe-59	Co-60	Zn-65	Total
				(December 1991)			
A Suction	50	0.74	0.75	n/d	2.95	0.95	5.39
B Suction	50	0.66	0.84	n/d	2.86	0.99	5.35
A Discharge	60	0.50	0.95	0.32	2.62	0.72	5.11
B Discharge	60	0.48	0.70	n/d	2.85	0.91	4.94
				(October 1993)			
A Suction	180	2.58	0.88	0.14	6.05	1.73	11.38
B Suction	170	2.52	1.08	0.16	6.33	1.95	12.04
A Discharge	160	2.04	0.94	0.25	6.14	1.99	11.35
B Discharge	145	1.52	0.73	0.25	5.45	1.72	9.67

(1) - Measurements made with pipes water-filled and insulation in place.
n/d - not detected

Unit 3 1.4 and 2.8 EFPY

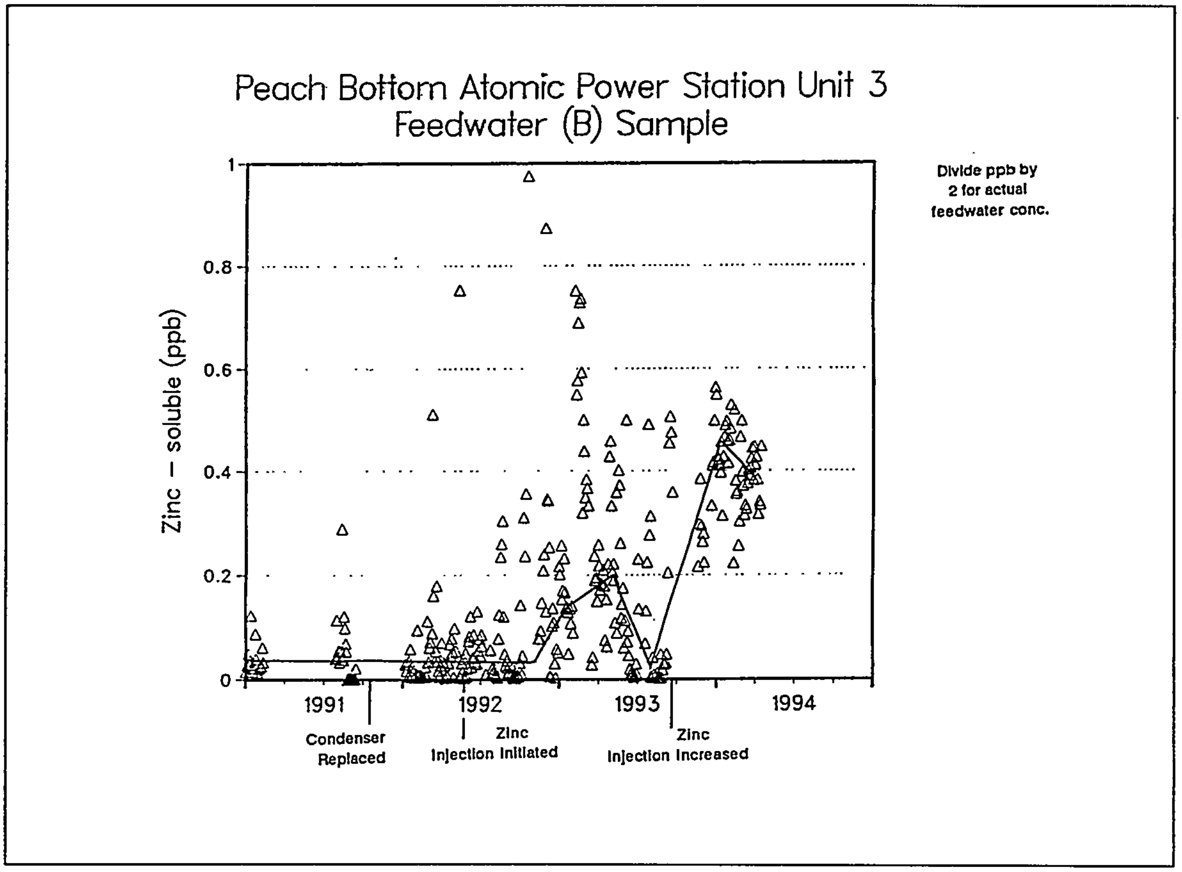
Historical Specific Activity Buildup in Recirculation System Loop B Suction and Discharge Piping

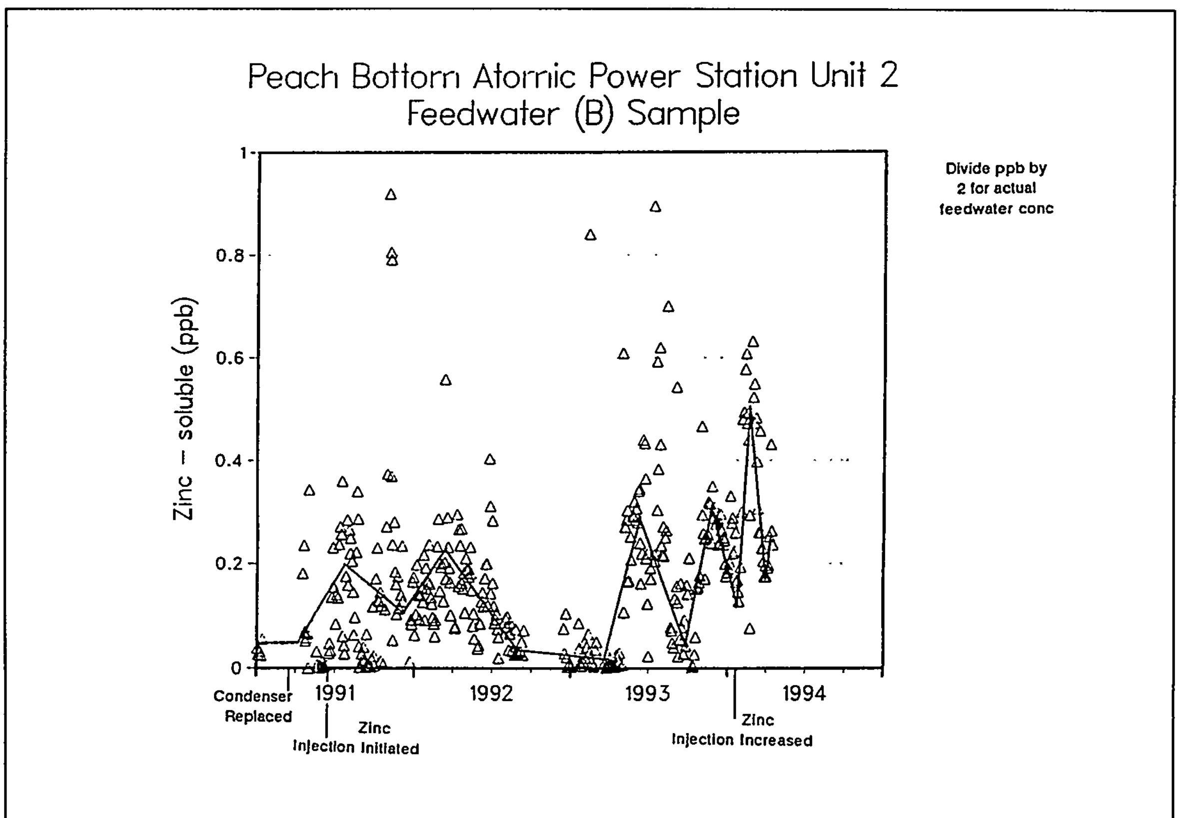


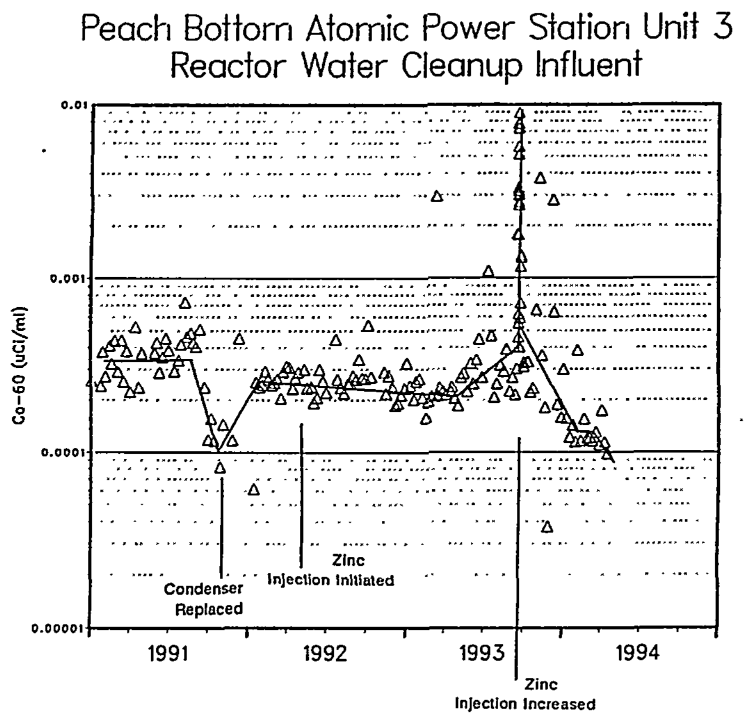


Unit 2 2.4 and 3.5 EFPY
Comparison of Current and Prior Specific Actitity Levels

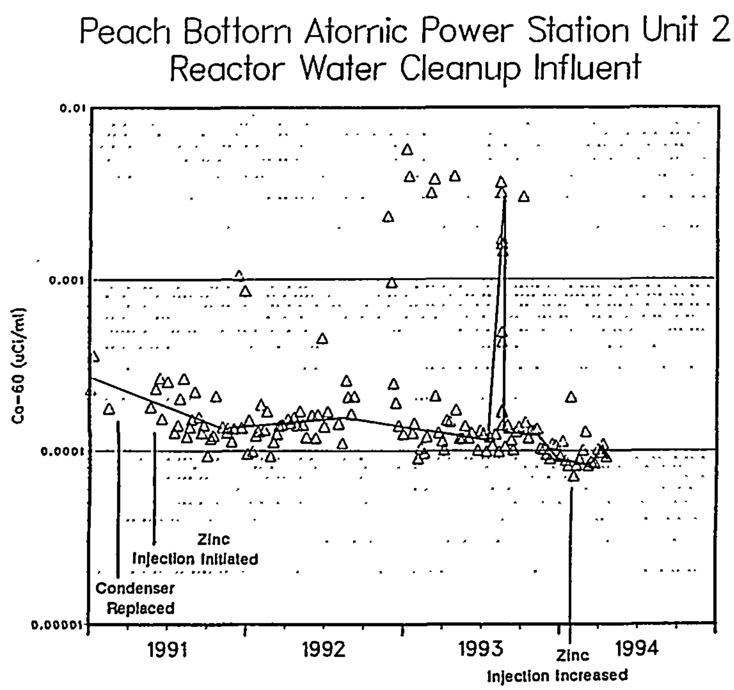
DATE	SPECIFIC ACTIVITY, $\mu Ci/cm^2$				
	Co-58	Mn-54	Zn-65	Co-60	TOTAL
March 1991	1.68	0.49	0.48	4.67	7.32
Nov. 1992	0.46	1.14	0.91	4.91	7.42
Change	-1.22	+0.65	+0.43	+0.24	+0.10
Percent	-73%	+133%	+90%	+5%	+1.4%


Peach Bottom Atomic Power Station Unit 3
Feedwater (B) Sample
Zinc - soluble (ppb)
1
0.8
0.6
0.4
0.2
0
1991
1992
1993
1994
Condenser Replaced
Zinc Injection Initiated
Zinc Injection Increased
Divide ppb by 2 for actual feedwater conc.

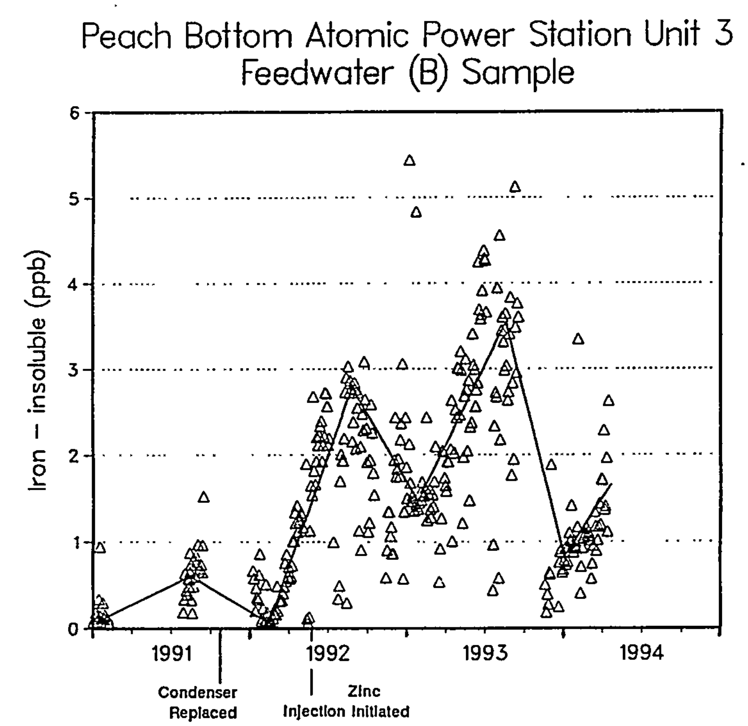

Peach Bottom Atomic Power Station Unit 2
Feedwater (B) Sample
Zinc - soluble (ppb)
1
0.8
0.6
0.4
0.2
0
1991
1992
1993
1994
Condenser Replaced
Zinc Injection Initiated
Zinc Injection Increased
Divide ppb by 2 for actual feedwater conc

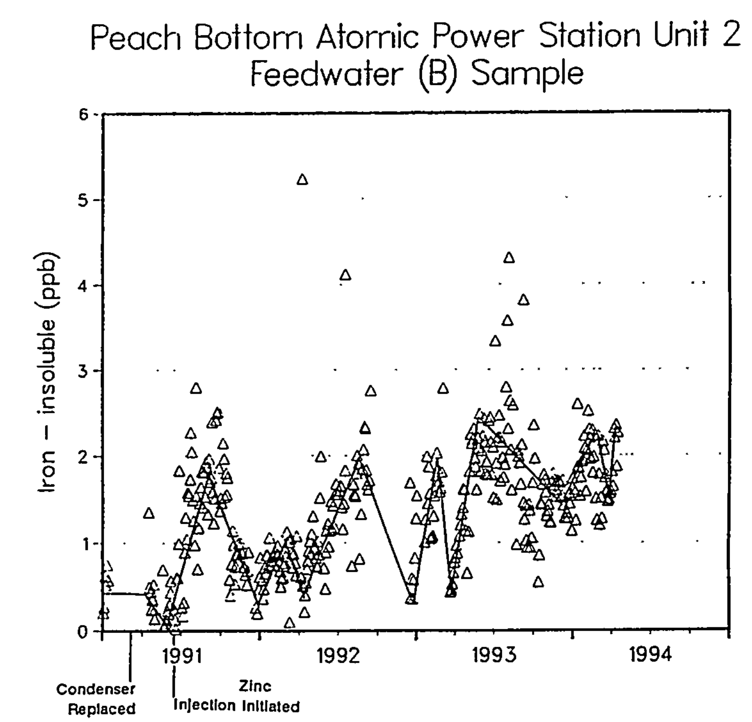


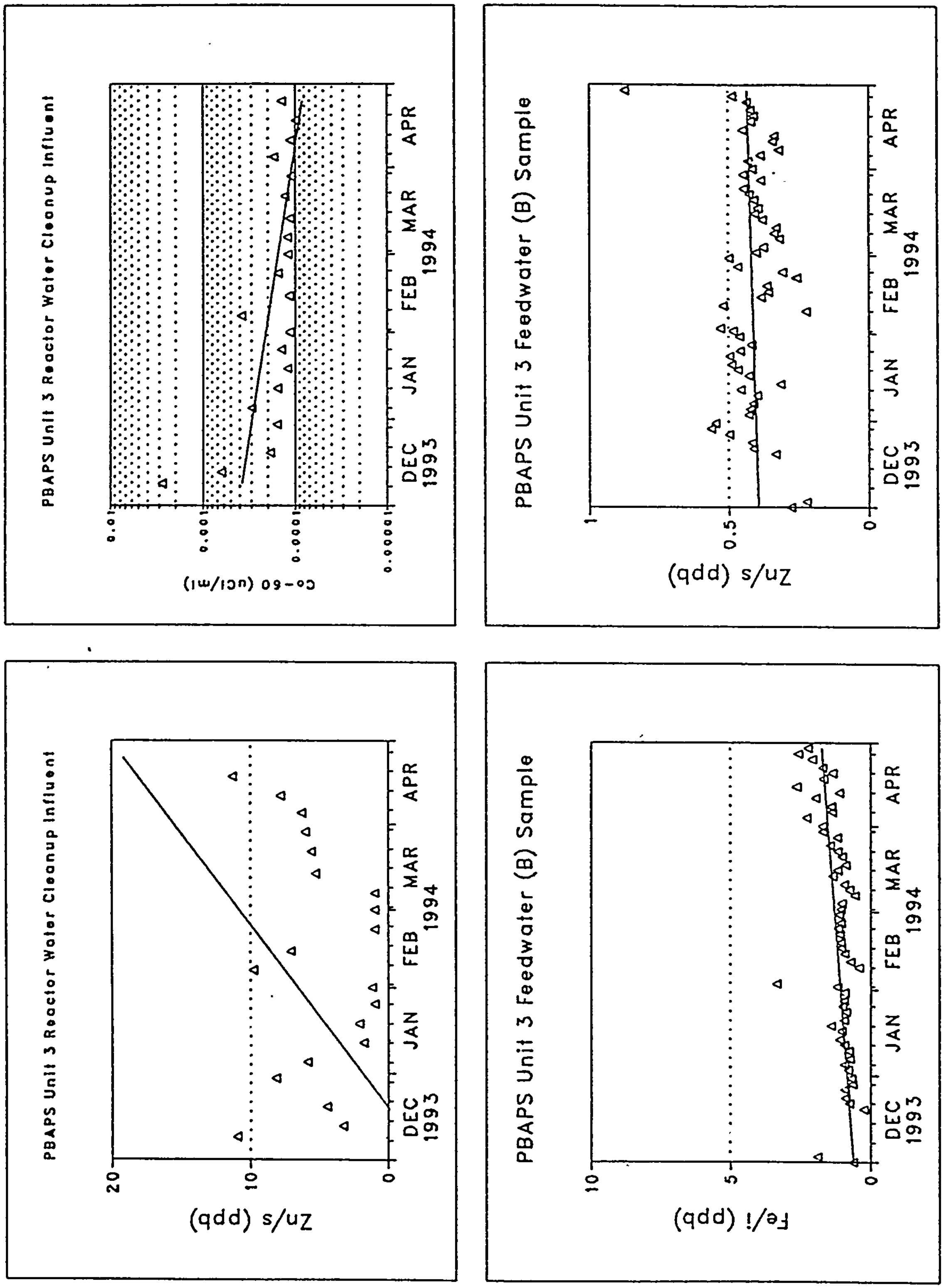
Peach Bottom Atomic Power Station Unit 3
Reactor Water Cleanup Influent
0.01
0.001
0.0001
0.00001
Co-60 (uCi/ml)
Condenser Replaced
Zinc Injection Initiated
Zinc Injection Increased
1991
1992
1993
1994



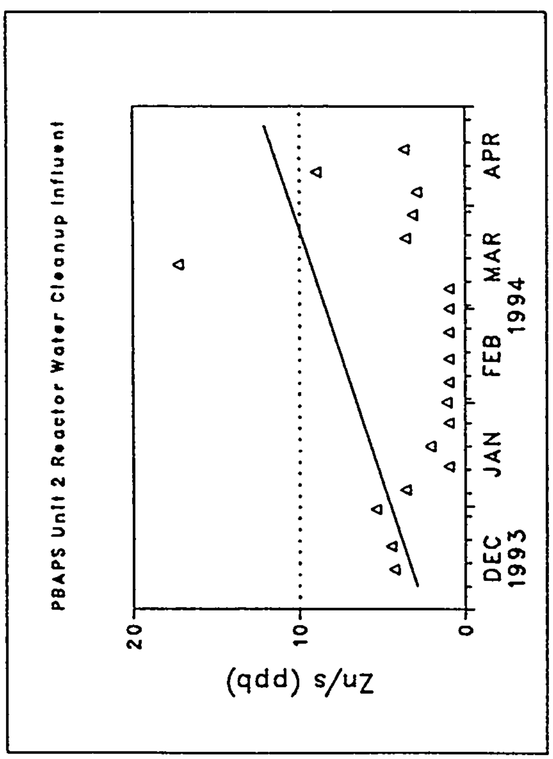
Peach Bottom Atomic Power Station Unit 2
Reactor Water Cleanup Influent
0.01
0.001
0.0001
0.00001
Co-60 (uCi/ml)
Zinc Injection Initiated
Condenser Replaced
Zinc Injection Increased
1991
1992
1993
1994

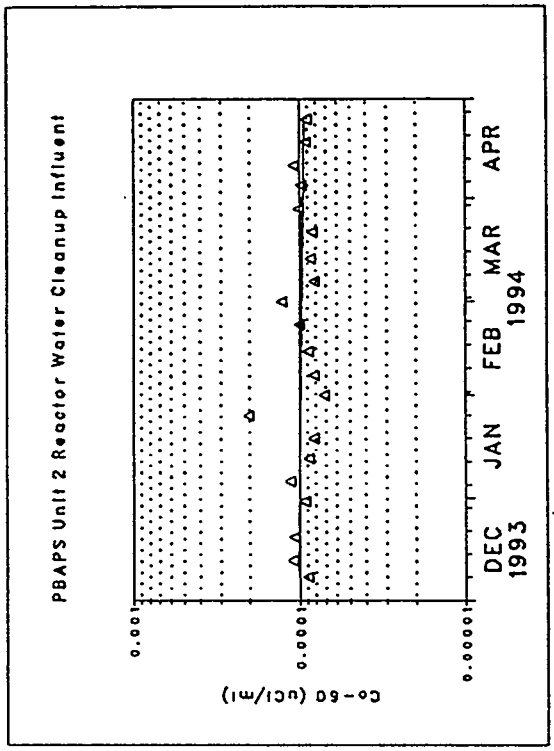

Peach Bottom Atomic Power Station Unit 3
Feedwater (B) Sample
Iron – insoluble (ppb)
0
1
2
3
4
5
6
1991
1992
1993
1994
Condenser Replaced
Zinc Injection Initiated

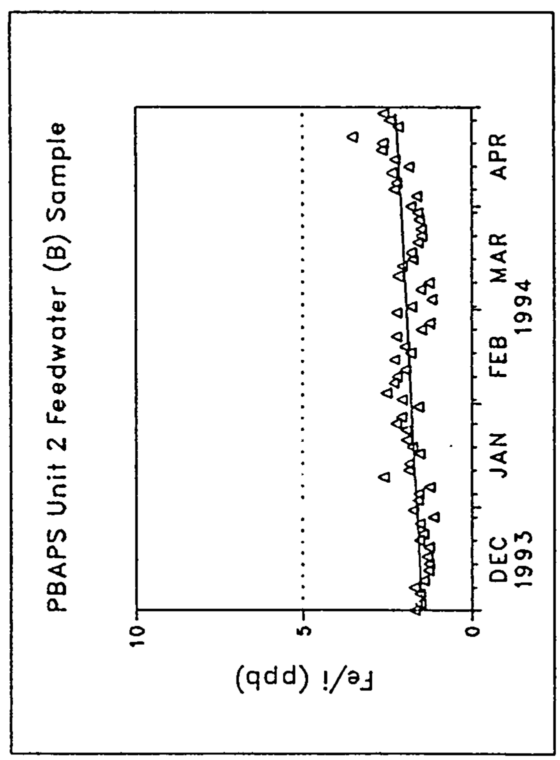

Peach Bottom Atomic Power Station Unit 2
Feedwater (B) Sample
Iron – insoluble (ppb)
0
1
2
3
4
5
6
1991
1992
1993
1994
Condenser Replaced
Zinc Injection Initiated

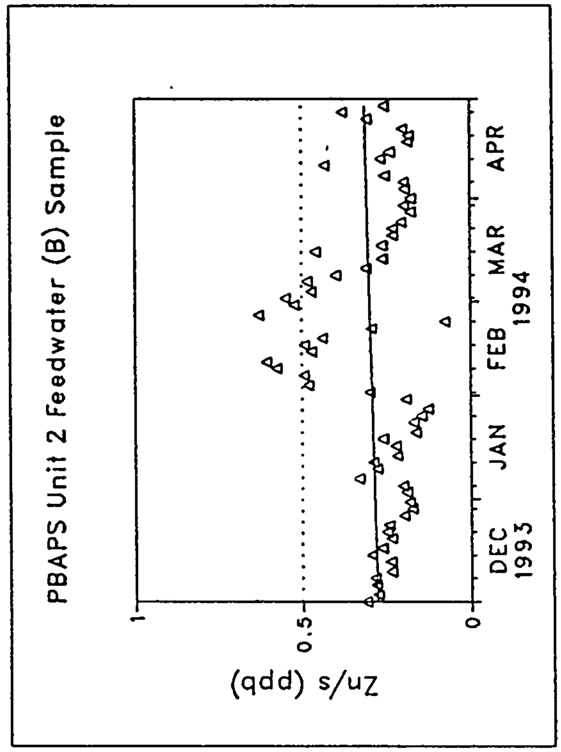


PBAPS Unit 3 Reactor Water Cleanup Influent
Co-60 (uci/ml)
0.01
0.001
0.0001
0.00001
DEC
1993
JAN
FEB
MAR
1994
APR
PBAPS Unit 3 Reactor Water Cleanup Influent
Zn/s (ppb)
20
10
0
DEC
1993
JAN
FEB
MAR
1994
APR
PBAPS Unit 3 Feedwater (B) Sample
Zn/s (ppb)
1
0.5
0
DEC
1993
JAN
FEB
MAR
1994
APR
PBAPS Unit 3 Feedwater (B) Sample
Fe/i (ppb)
10
5
0
DEC
1993
JAN
FEB
MAR
1994
APR


PBAPS Unit 2 Reactor Water Cleanup Influent
Zn/s (ppb)
20
10
0
DEC 1993
JAN
FEB 1994
MAR
APR


PBAPS Unit 2 Reactor Water Cleanup Influent
Co-60 (uci/ml)
0.001
0.0001
0.00001
DEC 1993
JAN
FEB 1994
MAR
APR


PBAPS Unit 2 Feedwater (B) Sample
Fe/i (ppb)
10
5
0
DEC 1993
JAN
FEB 1994
MAR
APR


PBAPS Unit 2 Feedwater (B) Sample
Zn/s (ppb)
1
0.5
0
DEC 1993
JAN
FEB 1994
MAR
APR

ALARA COUNCIL: SHARING OF RESOURCES AND EXPERIENCES TO REDUCE DOSES AT COMMONWEALTH EDISON FACILITIES

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SUMMARY

Commonwealth Edison Company is an investor-owned utility company supplying electricity to over three million customers (eight million people) in Chicago and northern Illinois, USA. The company operates 16 generating stations which have the capacity to produce 22,522 megawatts of electricity. Six of these generating stations, containing 12 nuclear units, supply 51% of this capacity. The 12 nuclear units are comprised of four General Electric boiling water (BWR-3) reactors, two General Electric BWR-5 reactors, and six Westinghouse four-loop pressurized water reactors (PWR).

In August 1993, Commonwealth Edison created an ALARA Council with the responsibility to provide leadership and guidance that results in an effective ALARA Culture within the Nuclear Operations Division. Unlike its predecessor, the Corporate ALARA Committee, the ALARA Council is designed to bring together senior managers from the six nuclear stations and corporate to create a collaborative effort to reduce occupational doses at Commonwealth Edison's stations.

This presentation describes the charter and mission of the ALARA Council, along with its membership. The ALARA Council will provide leadership and involvement in the following critical areas:

- Research and Development of Advanced Technologies.
- Source Term and Cobalt Reduction.
- Robotics Applications and Sharing of Resources.
- Evaluation and Improvement of Engineering, Maintenance and Operation Processes.
- Division Implementation of Cross-Discipline ALARA Initiatives.

The Council's operating principles and measurement standards for success are delineated. Finally, some initial actions of the Council are highlighted.

Author Biography

Frank Rescek is the corporate Radiation Protection Director for Commonwealth Edison and has 18 years experience with Edison. Currently, he serves as the chairperson for the Edison Electric Institute (EEI) Health Physics Committee and is a member of the BNL ALARA Center Advisory Committee and two NUMARC Advisory Committees. He is a past President of the Power Reactor Section of the Health Physics Society and served as a chairperson of the American Board of Health Physics (ABHP) Power Reactor Examination Panel, 1987. Previously, 1989-1990, he served on the Nuclear Energy Agency Expert Group involved with developing the Information System on Occupational Exposure (ISOE). He has a B.S. in Biology from Kent State University, an M.S. in Radiological Health from the University of Michigan, and an M.B.A. from the University of Chicago. Mr. Rescek is certified by the American Board of Health Physics - Power Reactors.

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INTERPRETATION OF ALARA IN THE CANADIAN REGULATORY FRAMEWORK

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The Atomic Energy Control Board (AECB) is responsible for the regulation of all aspects of atomic energy in Canada. This includes the complete nuclear fuel cycle from uranium mining to long-term disposal of nuclear fuel, as well as the medical and industrial utilization of radioisotopes. Clearly, the regulatory approach will differ from practice to practice but, as far as possible, the AECB has attempted to minimize the degree of prescription of regulatory requirements. The traditional modus operandi of the AECB has been to have broad general principles enshrined in regulations with the requirement that licensees submit specific operating policies and procedures to the AECB for approval. In the large nuclear facilities with their sophisticated technical infrastructures, this policy has been largely successful although in a changing legal and political milieu the AECB is finding that a greater degree of proactive regulation is becoming necessary. With the smaller users, the AECB has for a long time found it necessary to have a greater degree of prescription in its regulatory function.

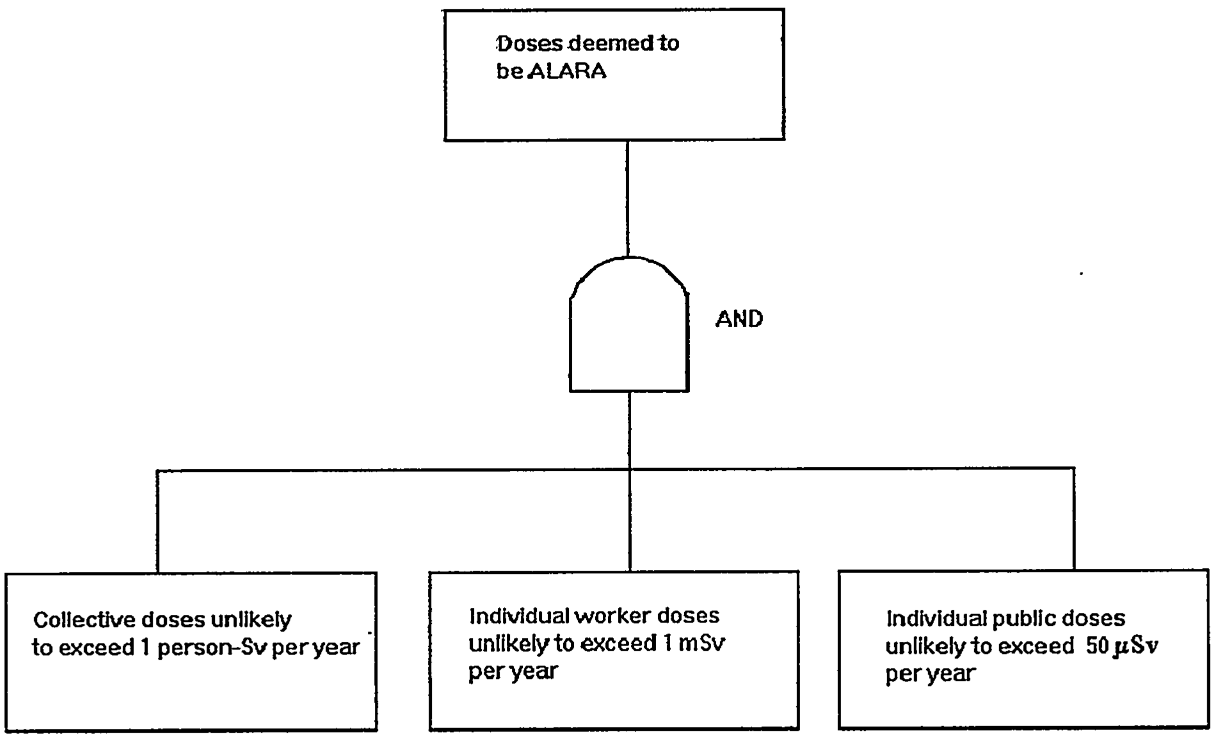
Forthcoming General Amendments to the Atomic Energy Control Regulations will, amongst other things, formally incorporate the concept of ALARA into the Canadian regulatory framework. Within the broad range of practices licensed by the AECB it is not practical to provide detailed guidance on optimisation that will be relevant and appropriate to all licensees, however the following general principles are proposed.

COMMITMENT

It is essential for good radiation protection that all levels of management, including the senior level of the organization, be committed to a policy of safety and good radiation protection. The AECB looks for evidence that senior management takes these commitments seriously and provides the means to carry them out. It is also essential that individual workers have a similar commitment to good radiation protection.

THRESHOLD VALUES BELOW WHICH DOSES ARE DEEMED TO BE ALARA

For many licensees where worker and public doses are already low it may not be reasonable to expect expenditure of resources to further reduce doses. To address such situations, the AECB has proposed the criteria illustrated in the following diagram. This diagram indicates that a)if the annual collective dose (occupational plus public) is unlikely to exceed 1 person-Sv, b) if individual occupational doses are unlikely to exceed 1 mSv per year and c) if doses to individual members of the public are unlikely to exceed 50 μSv per year, then existing exposures will be deemed to be as low as reasonably achievable without further evaluation. The value of one person-Sv is to a certain degree arbitrary but by assuming a value of a few tens of thousand dollars per person-Sv, a simple cost benefit calculation would indicate that one could not justify spending more than this to reduce this collective dose to zero. The cost of professional services to carry out the analysis would probably exceed this value even before any additional radiation protection measures could be implemented. The corresponding criteria regarding individual and public doses serves to highlight those situations where, even though the collective dose may not be large (i.e. less than 1 person-Sv), a limited number of people may still be receiving significant fractions of the individual or public dose limit. In such situations, additional radiation protection measures may still be required.



SYSTEMATIC PROGRAM

When the above criteria cannot be met, the AECB will expect licensees to adopt a systematic and well documented radiation protection program which addresses such issues as organization and management, facilities and equipment, policies and procedures and training programs. Where possible, these items should be reviewed and analyzed to determine if reasonable improvements can result in lower doses. A critical part of ensuring that doses are as low as reasonably achievable is the regular review of doses and other appropriate indicators such as contamination "events" and environmental monitoring results. The objective of these reviews is to identify trends so that the effectiveness of dose reduction efforts may be evaluated. As well as reviewing doses and other appropriate statistics, there should be a constant review of new technologies and procedures that might affect radiation protection.

JUDGEMENT OF REASONABLENESS

Following the above mentioned analyses, it must be determined if the benefit of action is worth the effort of doing it. Some problems may be quantifiable using techniques such as cost benefit analysis or other quantitative techniques. Many others will not and more qualitative judgements should be made. To substantiate the judgement, the licensee should conduct periodic reviews of the radiation protection program including review of such indicators as dose records, effluent releases, number of unplanned exposures etc. Such operational performance indicators can often identify problems that indicate that doses may not be as low as reasonably achievable.

The AECB will shortly publish a consultative document, number C-129, outlining its regulatory proposals on the requirement to keep all exposures as low as reasonably achievable.

Author Biography

Rod Utting is with the Atomic Energy Control Board (AECB) of Canada where he is the Head of the Operational Radiation Protection Section with responsibility for the evaluation of radiation protection programs in all types of licensed operations. During the mid-1980s, he was with the Nuclear Safety Division of the International Atomic Energy Agency (IAEA) in Vienna, where he was responsible for the development of a number of IAEA safety documents on Operational Radiation Protection and Optimization.

Prior to joining the AECB in 1977, Rod was with Ontario Hydro where he was involved with operational radiation protection at a number of different nuclear power plants. Rod started his health physics career at the Berkeley Nuclear Laboratories of the Central Electricity Generating Board in the United Kingdom in the 1960s.

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STATUS OF ZINC INJECTION IN PWRs

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SUMMARY

Based on laboratory and other studies, it was concluded that zinc addition in a PWR primary coolant should result in reduced Alloy 600 PWSCC and general corrosion rates of the materials of construction. Because of these positive results, a Westinghouse Owner's Subgroup, EPRI, and Westinghouse provided funds to continue the development and application of zinc in an operating plant.

As part of the program, Southern Operating Nuclear Company agreed to operate the Farley 2 plant with zinc addition as a demonstration test of the effectiveness of zinc. Since zinc is incorporated in the corrosion oxide film on the primary system surfaces and Farley 2 is a mature plant, it was estimated that about 10 kgs of zinc would be needed to condition the plant before an equilibrium value in the coolant would be reached.

The engineering aspects of a Zinc Addition and Monitoring System (ZAMS) considered such items as the constituents, location, sizing and water supply of the ZAMS. Baseline data such as the PWSCC history of the Alloy 600 steam generator tubing, fuel oxide thickness, fuel crud deposits, radiation levels, and RCP seal leak-off rates were obtained before zinc addition is initiated. This presentation summarizes some of the work performed under the program, and the status of zinc injection in the Farley 2 plant.

Author Biography

Carl A. Bergmann is a Principal Engineer in the Radiation and Engineering Analyses Group in the Nuclear Technology Division of Westinghouse Electric Corporation. He has over thirty years experience in the nuclear field and has been the lead engineer for the research, development and application of dose reduction techniques to PWR nuclear plants for fourteen years. Dose reduction techniques include the application of coolant additives such as zinc and enhanced amounts of lithium to the primary coolant. He also led a study to evaluate sources of cobalt in Westinghouse designed plants. Mr. Bergmann holds a B. S. Degree in Chemical Engineering and a Masters in Business Administration.

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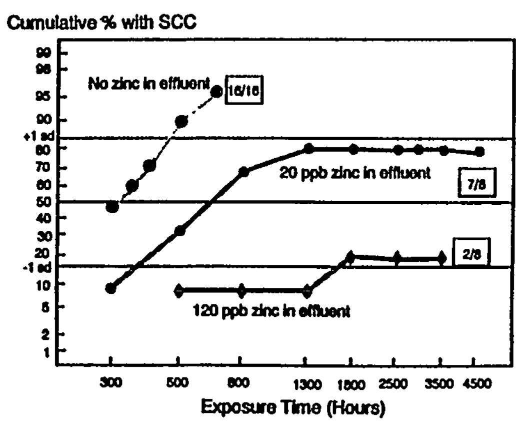
Overview

- Value of zinc addition
- Amount of zinc needed to condition oxide film
- Engineering aspects of Zinc Addition and Monitoring System (ZAMS)
- Demonstration plant status

Cumulative Crack Initiation Data for Alloy 600 (Heat 1019) RUBs

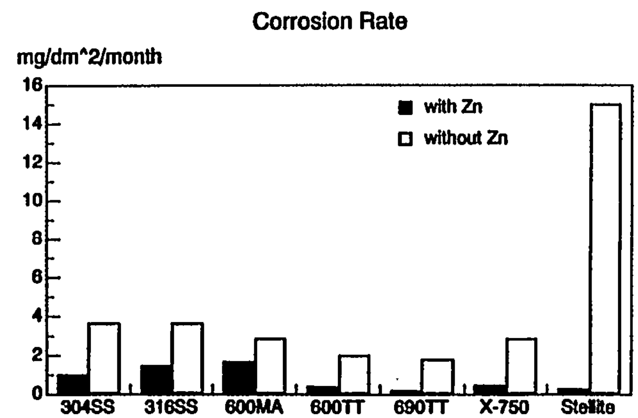
[330°C Exposures in PWR Primary Coolant]



• Time to crack initiation is increased with zinc addition

The Corrosion Rate of Primary Side Materials is Reduced by Zn Additions

[2500 Hrs. at 330°C in BOL PWR Coolant Containing 20 ppb Zinc]



Amount of Zinc to Condition Oxide Film

- Since first application is in a mature plant, an estimate of the amount of zinc to condition oxide film was made to arrive at an expected time for zinc to reach equilibrium in coolant
- Estimates based on plant data and laboratory tests
- Plant data used:
  - Thickness of oxide film on Alloy 600 and stainless steel surfaces about 2 to 6 microns, respectively
  - Amount of corrosion product oxide on fuel and S/G channel head from scrapings and decontamination
- "Equilibrium" concentration of zinc in oxide film, about 2.8% for Alloy 600 and 5.6% for stainless steel, from laboratory tests
- Based on above, about 10 kg of zinc will be needed to condition the oxide film

Constituents of Prototype ZAMS for Demonstration Plant

- RCS sample skid
  - Includes zinc analyzer, valves and pumps to provide automatic sampling and discharge of sampling wastes
- Zinc injection skid
  - Two batching tanks, dry material feeders, mixers and recirculation pump, and injection pump
- Control skid
  - Includes power supplies, fuse blocks, transformers, and computer console
- Note: Arrangement and geometry for the skids is plant dependent

Preliminary Sizing and General Design Criteria

- Using a nominal letdown/charging rate of 100 gpm, the zinc concentration at the regenerative heat exchanger was targeted at 250 ppb and at 50 ppm in the ZAMS feed tanks
- From these values a maximum injection rate of about 0.30 gpm was estimated
- General functional design criteria included:
  - Automatic feed tank batching (two feed tank system)
  - Minimization of plant space, operator attention, maintenance and interfacing with auxiliary systems
  - Zinc analytical method to be on-line, rugged, and generate minimal waste
  - Injection rate automatically controlled by RCS zinc analyses after conditioning

ZAMS Water Supply and Injection Location

- Two options evaluated -- reactor coolant downstream of CVCS mixed bed demineralizers, or demineralized water into the VCT
- Reactor coolant source less desirable due to need for ventilation system to control coolant gases and contamination of batching subsystem
- Demineralized water chosen as better source
  - Can separate injection portion (non-radioactive) from RCS analyzer portion
  - With the low injection rate, no net effect on RCS inventory and on boron dilution

ZAMS Sampling and Injection Locations Criteria

- Sampling is performed downstream of sample heat exchanger, in the RCS hot leg sample line
- Injection location basis was chosen to permit use of non-nuclear tubing, injecting into the VCT or charging pump suction header
- Neither location is safety-related, thus simplifying design

Demonstration Plant (Farley 2) Status

- Objectives:
  - Determine effectiveness of zinc addition in inhibiting Alloy 600 PWSCC and dose rates
  - Provide assurance that zinc addition is not deleterious to the fuel region performance and that there is no unacceptable effect on RCP seals, valves, and CVCS resin beds
- Baseline data taken:
  - Eddy current measurement for fuel clad oxide thickness and sampling of fuel crud
  - Characterization of expansion transition regions of 100% of hot leg tube ends in all steam generators by RCP eddy current
  - Dose rates and gamma spectrometry of selected inspection points

Demonstration Plant (Farley 2) Status (Cont'd)

- Baseline data taken:
  - Maintenance history of certain CVCS valves
  - RCP seal leakoff, seal injection flow and temperature, RCP and motor bearing temperatures, and shaft and frame vibration data
  - Reactor coolant soluble and insoluble radiocobalt concentrations and CVCS resin changeouts
- Monitoring of the areas above as well as the performance of the ZAMS will be continued during the zinc injection cycle
- Zinc injection to start about the second week of May, 1994

PEACH BOTTOM ATOMIC POWER STATION UNIT 3 RADIATION BUILDUP ON THE REACTOR WATER CLEANUP (RWCU) TEST SPOOL - AN UPDATE

David C. DiCello, R. Mark Moore, and Richard P. Farrell
PECO Energy Company
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Delta, PA 17314 USA

SUMMARY

Peach Bottom Atomic Power Station (PBAPS) is located near the town of Delta, Pennsylvania, on the west bank of the Susquehanna river. It is situated approximately 20 miles south of Lancaster, Pennsylvania. The site contains two (2) boiling water reactors of General Electric design and each rated at 3,293 megawatts thermal. The units are BWR 4s and went commercial in 1977. There is also a decommissioned high temperature gas-cooled reactor onsite, Unit 1.

The installation of a RWCU pipe test spool on Peach Bottom Unit 3 was initially sponsored by EPRI and was exposed to reactor water in December 1989. The spool piece had various surface treatments including flex-honing (mechanical polishing), electropolishing, and preoxidation/passivation and was exposed to reactor water prior to filtration under normal BWR chemistry.

Initial contact dose rates and isotopic concentrations in the oxide layer of the pipe were made in 1991 following one fuel cycle. The initial 1991 measurement was made while Unit 3 was a natural zinc plant. In May 1992, zinc injection was started in conjunction with the removal and replacement of the admiralty brass condenser with a Titanium condenser. The latest measurements were made in October 1993 following a second fuel cycle run on the pipe.

The as-received section with no treatment showed the highest contact dose rate of 675 mr/hr. The test section which was flex-honed (mechanically polished) and passivated had the lowest dose rate and measured 400 mR/hr on contact or a 41% reduction from the as-received section. The next best treatment was the electropolished and passivated section which had a 480 mR/hr contact dose rate or a 29% reduction from the as-received section. Surface treatment of as-received pipe resulted in 30 to 40% reductions in pipe dose rates and this data can be used to cost justify pipe surface treatments in the future. Data from other less effective surface treatments are also presented.[1]

---

[1] Data collection and analysis provided by Radiological & Chemical Technology, Inc., 1700 Wyatt Drive, Suite 16, Santa Clara, CA 95054.

Authors' Biographies

<u>David C. DiCello</u> is Manager, Radiological Engineering, at the PECO Energy Company's Peach Bottom Atomic Power Station (PBAPS). He manages the overall implementation of the ALARA program at the station and supervises a staff of six (6) Radiological Engineers. He previously served as the Health Physics Technical Support Supervisor at PBAPS and as an in-plant Radiological Engineer at PECO Energy's Limerick Generating Station. Prior to joining PECO Energy, he worked as a Corporate Radiological Engineer for Long Island Lighting at the Shoreham Nuclear Power Station. Previously he worked at Princeton University as the Assistant University Health Physicist. He has a B.S. in Biological Services from the University of Pittsburgh and an M.S in Radiological Health from the University of Pittsburgh - Graduate School of Public Health. He is both comprehensively and power reactor certified by the American Board of Health Physics (ABHP).

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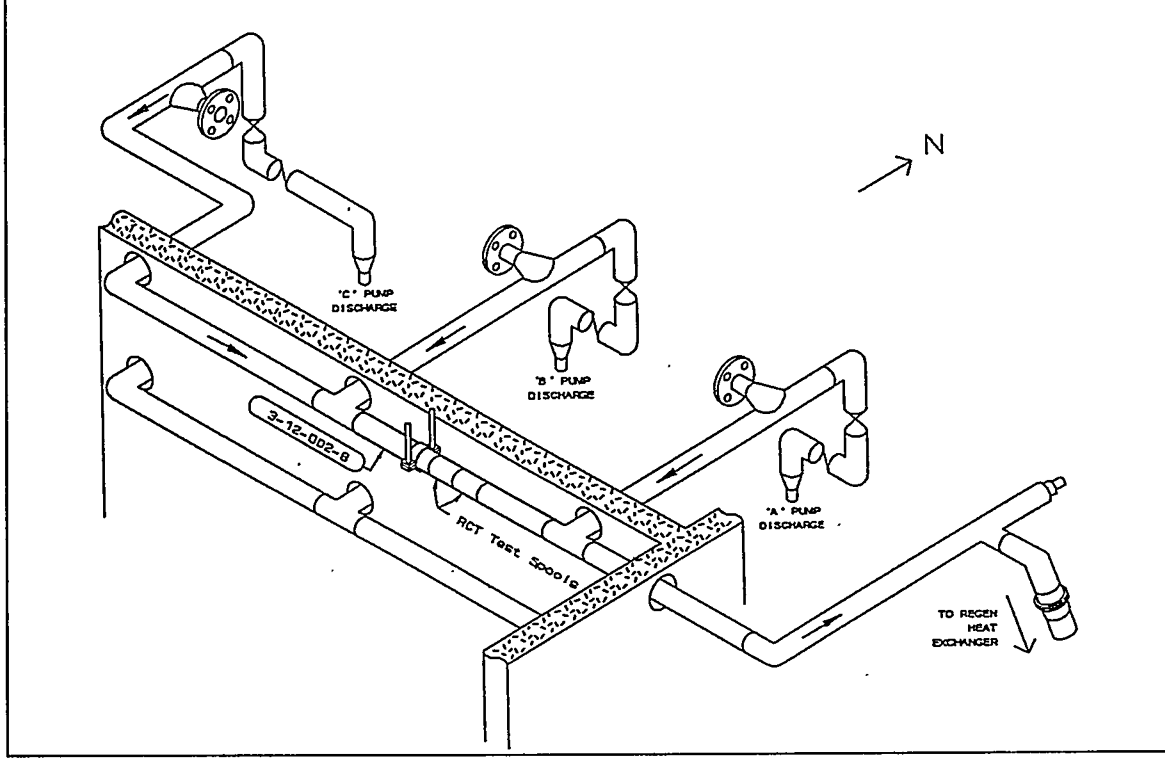
R. Mark Moore is the Manager, Radiation Protection, at the PECO Energy Company's Peach Bottom Atomic Power Station (PBAPS). He manages the overall Health Physics program at Peach Bottom. Previous assignments with PECO Energy included Services Training Manager, Quality Assurance Engineer and Health Physics Technician. He holds a B.S. in Engineering from Widner University.

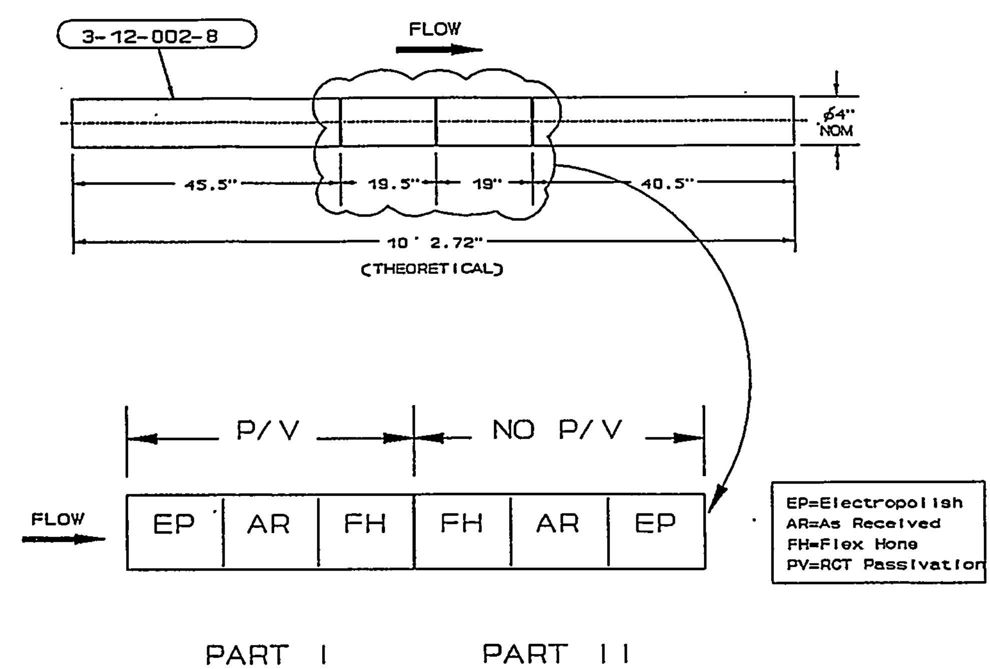
Richard P. Farrell is Manager, Support Health Physics at the PECO Energy Company's Peach Bottom Atomic Power Station (PBAPS). He manages the Technical Support group of Health Physics at the station. The group consists of 5 professional Health Physicists who generate the technical guidance for the station Radiation Protection program, including Dosimetry, Respiratory Protection, Instrumentation and Applied Health Physics procedures. Prior to working at PBAPS, he worked as a Corporate Health Physicist at the PECO Nuclear Group headquarters, Iowa Electric at the Duane Arnold Energy Center, and General Public Utilities at the Oyster Creek Nuclear Generating Center, where he worked as the Dosimetry Supervisor and a Senior Radiological Engineer respectively. He has a BS in Radiation Protection from the Thomas Edison State College and an MS in Health Physics from the Rutgers University Graduate School. He is a registered Radiation Protection Technologist and an Associated Member of the American Academy of Health Physics.

Unit 3 Test Spool

RCT Test Spool Location in Peach Bottom-3 RWCU Discharge Piping



Test Spool Surface Treatment Zones



Surface Roughness of RWCU System Pipe Sections
(Values are in microinches)

Surface Treatment	Part 1 (Passivated)			Part 2 (Non-Passivated)		
	FH Zone	AR Zone	EP Zone	FH Zone	AR Zone	EP Zone
As-Received	20.0	20.0	22.0	26.5	20.0	20.0
After FH	9.2	---	---	8.0	---	---
After EP	---	---	13.2	---	---	10.7
After PV	10.3	20.0	8.8	---	---	---

FH = Flex-Hone Treatment
EP = Electropolishing
PV = Preoxidation

Unit 3 1.4 and 2.8 EFPY

Dose Rates of Peach Bottom-3
RWCU Test Spool Treatment Zones

Surface Treatment Zone	Dose Rate(1), mR/hr	
	(Dec. 1991)(2)	(Oct. 1993)(3)
As-Received Only	550	675
Flex-Honed Only	500	650
Electropolished Only	450	600
As-Received+Passivated	350	550
Flex-Honed+Passivated	300	400
Electropolished+Passivated	275	480

(1) All dose rates measured with E530-N survey meter in contact with pipe (i.e., insulation removed).
(2) 1991 measurements made with pipe water-filled.
(3) 1993 measurements made with pipe empty.

Unit 3 1.4 and 2.8 EFPY

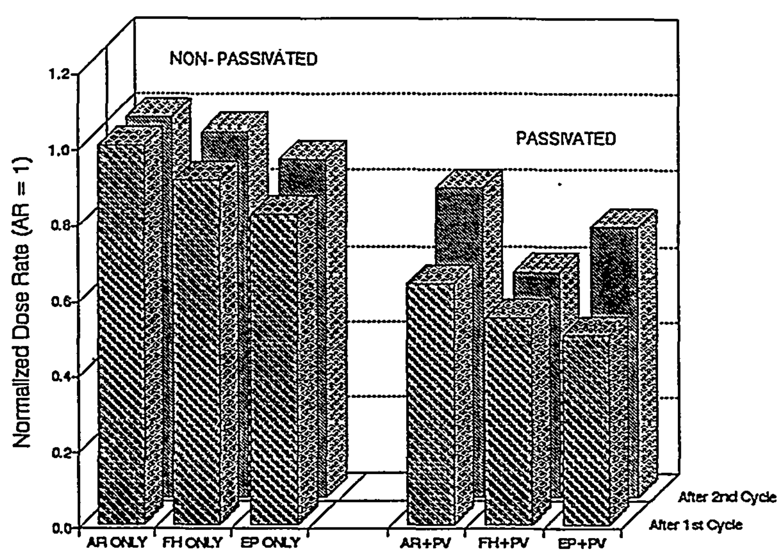
Specific Activity Levels in Peach Bottom-3 RWCU Test Spools Since Recirculation Pipe Replacement

Surface Treatment Zone	Dose Rate (1) (mR/hr)	Specific Activity(1), µCi/cm²				
		Mn-54	Co-58	Co-60	Zn-65	Total
		(December 1991)				
AR Only	550	0.26	0.54	4.50	1.81	7.11
FH Only	500	0.28	0.60	4.03	1.76	6.67
EP Only	450	0.24	0.47	3.88	1.52	6.11
AR+PV	350	0.22	0.60	2.61	0.93	4.36
FH+PV	300	0.24	0.60	2.26	0.72	3.82
EP+PV	275	0.16	0.55	1.95	0.70	3.36
		(October 1993)				
AR Only	675	0.15	0.31	3.90	0.95	5.31
FH Only	650	0.13	0.49	3.91	1.41	5.94
EP Only	600	0.08	0.38	3.80	1.11	5.37
AR+PV	550	0.29	0.75	3.65	1.18	5.87
FH+PV	400	0.05	0.34	2.56	0.82	3.77
EP+PV	480	0.33	0.51	3.44	1.13	5.41

AR - As-Received EP - Electropolished FH - Flex-Honed PV - Passivated

(1) All dose rates measured with E530-N survey meter in contact with pipe. 1991 measurements made with pipe water-filled. 1993 measurements made with pipe empty.

Normalized Dose Rates of WRCU Test Spool Treatment Zones

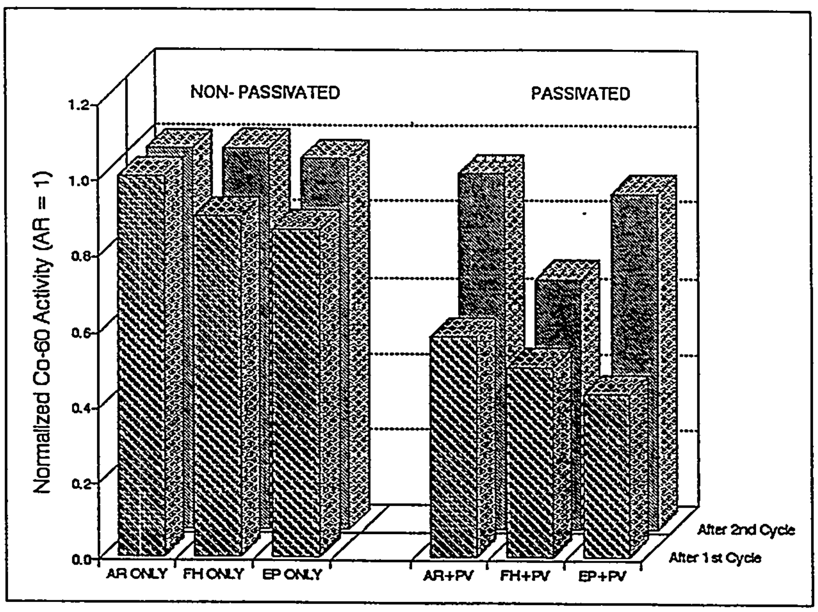


Unit 3 1.4 and 2.8 EFPY

Normalized Activity Levels of RWCU Test Spool Treatment Zones

Surface Treatment Zone	Normalized Activity[1]				
	Mn-54	Co-58	Co-60	Zn-65	Total
	(December 1991)				
As-Received Only	1	1	1	1	1
Flex-Honed Only	1.08	1.11	0.90	0.97	0.94
Electropolished Only	0.92	0.87	0.86	0.84	0.86
As-Received+Passivated	0.85	1.11	0.58	0.51	0.61
Flex-Honed+Passivated	0.92	1.11	0.50	0.40	0.54
Electropolished+Passivated	0.62	1.02	0.43	0.39	0.47
	(October 1993)				
As-Received Only	1	1	1	1	1
Flex-Honed Only	0.87	1.58	1.00	1.48	1.12
Electropolished Only	0.53	1.23	0.97	1.17	1.01
As-Received+Passivated	1.93	2.24	0.94	1.24	1.11
Flex-Honed+Passivated	0.33	1.10	0.66	0.86	0.71
Electropolished+Passivated	2.20	1.65	0.88	1.19	1.02

Normalized Co-60 Activity of RWCU Test Spool Treatment Zones



ALARA AND WORK MANAGEMENT

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ABSTRACT

At the request of Electricité de France (EDF) and Framatome, the Nuclear Protection Evaluation Centre (CEPN) developed a three-year research project, between 1991 and 1993, to evaluate the impact of various work management factors that can influence occupational exposures in nuclear power plants (NPPs) and to assess the effectiveness of protective actions implemented to reduce them.

Three different categories of factors have been delineated: those linked to working conditions (such as ergonomic of work areas and protective suits), those characterizing the operators (qualification, experience level, motivation, etc.), and the factors directly dependent on the operation's organization (tasks planning, general preparation of works, etc.). In order to quantify the impact of these factors, a detailed survey was carried out in five French NPPs, focusing on three types of operations: primary valves maintenance, decontamination of reactor cavity, and specialized maintenance operations on the steam generator. This survey was augmented by a literature review on the influence of "hostile" environment on working conditions. Finally, a specific study was performed in order to quantify the impact of various types of protective suits used in French nuclear installations according to the type of work to be done. All of these factors have been included in a model aiming at quantifying the effectiveness of protection actions, both from dosimetric and economic point of views.

INTRODUCTION

The application of the ALARA principle to the management of occupational exposure implies to adopt an analytical approach in order to identify the relevant factors contributing to the formation of individual and collective exposures. Within these factors, all procedures and actions which can influence the duration of exposure and the number of exposed workers come under the heading of "Work Management."[1]

Within the framework of their ALARA programs, EDF (Electricité de France) and FRAMATOME have initiated a pluri-annual research project conducted by the CEPN (Nuclear Protection Evaluation Centre), in order to delineate the various factors related to work management which can influence occupational exposures and to evaluate the effectiveness of possible protection actions influencing these factors.

Different categories of factors have been delineated. A quantification of the impact of some working conditions have been done. Finally, all of these factors have been included in a model aiming at quantifying the effectiveness of protection actions, both from dosimetric and economic point of views.[2]

RELEVANT WORK MANAGEMENT FACTORS

The objective of applying optimization of radiation protection by the way of work management is mainly to reduce the time spent in radioactive areas and the number of workers exposed. While analyzing the total exposed time resulting from any maintenance operation, it appears that it can be split into two main parts: the "productive" and the "non-productive" time. The productive exposed time can be defined as the time which is technically necessary in order to complete the task, given the state of the technology and the set of working conditions. The non-productive exposed time usually results of mishaps due to a bad training of workers, malfunctioning of tools, etc.

Various factors having direct or indirect impacts on the productive exposed time have been identified. They can be grouped together into three main categories:

1) Working conditions

- Individual protection
- Collective protection
- Noise, light, thermal conditions, etc.
- Dimension of the working area
- Ambient dose rate
- Adaptation of tools
- Video or audio links
- Shift work conditions

2) Worker characteristics

- Qualification
    - Radiation protection education
    - Specific specialty related training
    - Specific task related training
- Experience
    - Individual experience
    - Transfer of experience between teams
- Motivation
    - Individual motivation
    - Management commitment

3) Work organization

- Scheduling of tasks
- Preparation of working areas
- Preparation of equipments
- Level of information of the workers

A modification of one or several working condition factors will have a direct impact on the productivity of workers and then influence directly the productive exposed time. The factors characterizing the operators will modify the productive exposed time and mainly the nonproductive one, which is usually due to a bad knowledge of working areas and tasks to be performed. The general work organization will particularly affect the nonproductive exposed time.

The above factors may be quantified by first creating for each factor a scale of values describing different possible situations. The second step is the quantification of the impact on productivity and doses of moving from one situation to a "better" one, either for one factor or for a combination of factors. This quantification would result in determining time modification coefficients corresponding to the previous scales of values. It is then possible to predict the impact on exposure time of any action improving the situation.

ESTIMATION OF IMPACT OF FACTORS

Impact of Protective Suits

In order to assess the impact of protective suits, a specific study on mock-up was performed.[3] Three different mock-ups were used to take account of the effect of ergonomic parameters like the level of effort, the need for precision or the task duration.

- The first mock-up was a steam generator channel head where a maintenance spider (20 kg) had to be installed and removed. This represented a heavy and precise work of short duration in a very congested area.
- The second mock-up was a "big" valve (12-inch) where the workers had to unscrew, remove, and screw 12 nuts (of 0,9 kg each). It was a heavy work, not very precise, in a less congested area with a long duration.
- The third mock-up was a "small" valve (2-inch) modelizing a long precise work in a congested area. The workers had to remove, place, and adjust two limit switches.

Eight clothing situations have been selected, representing protective suits of both French nuclear power plants and nuclear industry:

Suit 1 : 1 cotton coverall and 1 set of cotton gloves = "Reference"

Suit 2 : 2 cotton coveralls, 2 sets of rubber gloves, 1 respirator, 1 cotton hood

Suit 3 : 2 cotton coveralls, 1 rubber overall suit, 3 sets of rubber gloves, 1 air supplied respirator, 1 cotton hood

Suit 4 : 2 cotton coveralls, 1 rubber coverall, 3 sets of rubber gloves, 1 respirator, 1 cotton hood

Suit 5 : 1 cotton coverall, 1 rubber coverall, 1 set of cotton gloves, 1 set of rubber gloves, 1 air supplied hood

Suit 6 : 1 cotton coverall, 1 air supplied overall suit, 1 set of cotton gloves

Suit 7 : 2 cotton coveralls, 1 air supplied overall suit, 3 sets of rubber gloves, 1 air supplied respirator

Suit 8 : 1 cotton coverall, 1 air supplied overall suit, 1 set of cotton gloves (this suit has only been used for the steam generator mock-up)

Nine workers were timed on each mock-up, with every suits. Then, an average percentage of time difference was calculated for each mock-up and each suit, with the first suit always used as the reference.

The main results of this study are presented in Table 1.

Table 1. Impact of protective suits on exposed time

	Case 1: - Permanent concentration - Precision work - Heavy effort - Duration <2 mns - Very restricted workspace - Uncomfortable posture	Case 2: - Permanent concentration - Precision work - Heavy/light effort - Duration < 10 mns - Restricted workspace - Uncomfortable posture	Case 3: - Nonpermanent concentration - "Non precision" work - Heavy effort - duration <10 mns - Not much work space - Comfortable posture
Non ventilated cotton clothing			
Clothing 2: Cotton coverall + mask	34% (±17)	34% (±14)	19% (±14)
Non ventilated impervious clothing			
Clothing 3: Non ventilated Chadoc + ventilated mask	34% (±19)	65% (±20)	21% (±13)
Clothing 4: Impervious clothing + mask	29% (±8)	46% (±18)	25% (±13)
Clothing 5: Impervious clothing + ventilated hood	28% (±12)	27% (±16)	22% (±10)
Air-fed pressurized clothing			
Clothing 6: Air-fed pressurized Mururoa	30% (±11)	42 (±24)	8% (±4)
Clothing 7: Air-fed pressurized Chadoc + ventilated mask	51% (±12)	57% (±25)	16% (±14)
Clothing 8: Shrunken air-fed pressurized Mururoa	21% (±12)		

Dose Rates as a Working Condition Factor

Usually, the impact of dose rates is taken into account when workers have to perform a job in high ambiences. In this case, it is well known that the stress resulting from the dose rates can influence the productivity of workers. For such operations, the workers should perform a specific training to lower the potential effect of dose rates on the technical performance of the operation.

The analysis of feedback data concerning some specialized maintenance operation has pointed out another effect of dose rates which could be called the "lax" effect: when the same operation is performed in various radiological conditions, the lower is the ambient dose rate, the longer the time is spent to perform the job.

For example, the analysis of the collective exposure associated with the machining of Residual Heat Removal System heat exchangers performed on 17 French units between 1984 and 1988 by nearly the same team, revealed clearly this type of behavior. The trend of the collective dose without any reference to the associated dose rate, shows that an asymptote is reached starting from the eighth operation and the collective dose is nearly equal to 50 man-mSv for the last seven operations. However, the various operations have been performed in different ambient dose rates.. In order to make a true comparison of the exposure associated with the operations, the collective dose has to be related to the same value of ambient dose rate. It can then be seen that the "normalized" total doses of the last seven operations vary widely (see Figure 1).

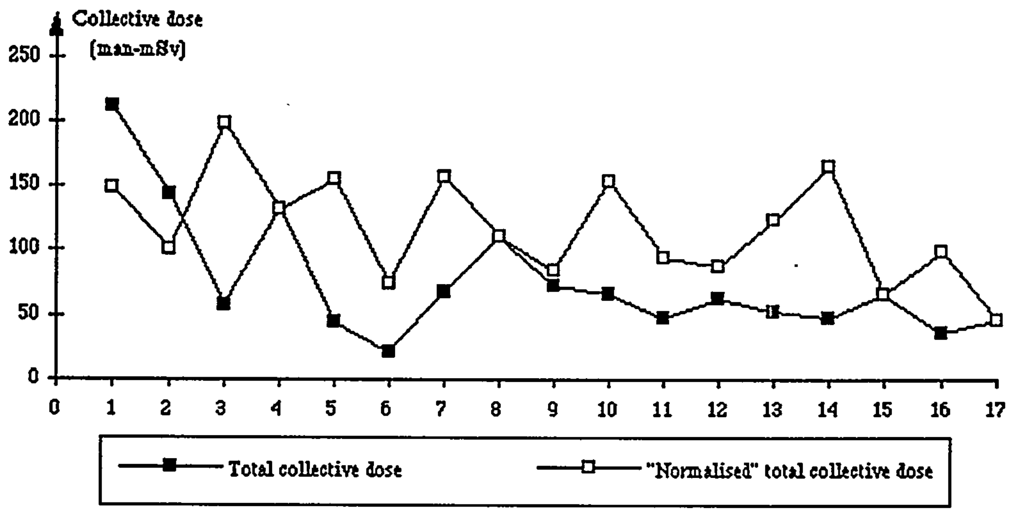

Figure 1. Evolution of the collective dose for the machining of RHR exchangers

The comparison between the "normalized" total collective dose (which in fact represents the level of exposed time) with the level of ambient dose rate reveals an inverse relationship between the level of dose rate and the exposed working time. This is shown on Figure 2, especially for the last operations when the workers are "used" to receive a collective dose of 50 man-mSv. As long as they have not reached the 50 man-mSv level, they are not really concerned by the level of exposure, considering they still have some "dose credit".

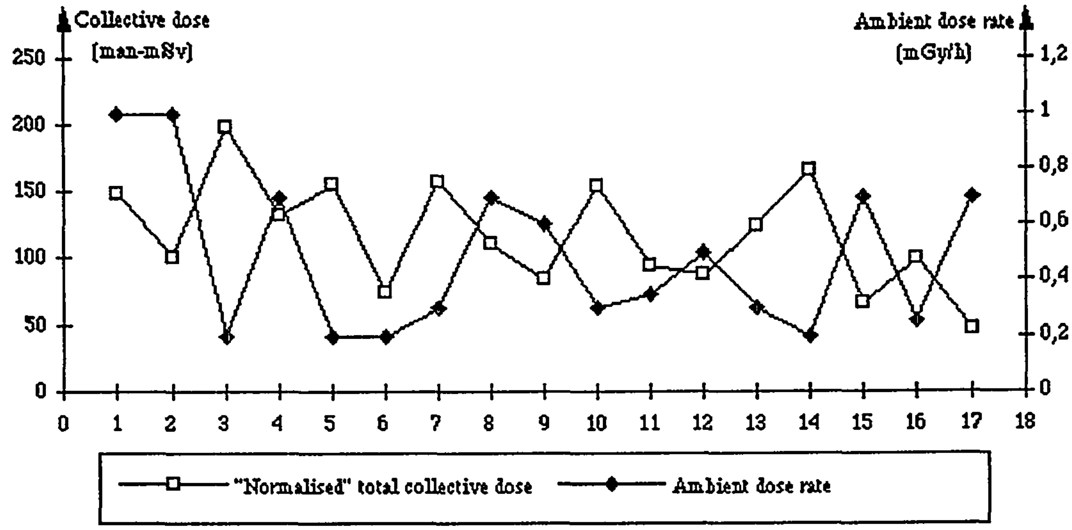

Figure 2. Evolution of the "normalized" total collective dose and the ambient dose rates for the machining of RHR exchangers

This type of result demonstrates the need for adequate estimates of collective doses before each job taking into account the actual ambient dose rate, and for a proper information of workers and health physicists before starting the work.

Other Factors

In order to quantify the impact of the above listed factors, a review of the literature was performed. It allowed to estimate the impact of the modification of some working conditions on exposed time. These results have been complemented by a survey carried out in five French nuclear power plants and focused on three types of operations: primary circuit valves maintenance, decontamination of reactor cavity and some specialized maintenance operations. Eighty persons (workers, foremen, health physicists, planners, etc.) have been interviewed about their perception concerning the impact of working conditions on the exposed work time, and on the main causes of mishaps.

As far as the ergonomic literature is concerned, most studies are focused on the potential "physiological" impact of working condition factors on workers. Very few deal with the impact of these factors on productive time. Table 2 presents a summary of both literature and survey results.

Table 2. Impact of working conditions factors on exposed time

Working conditions	Literature and survey
Training	- Savings between 30 % and 40 % in exposed time
Light	- 20 % of exposed time can be saved by good lighting of working areas.
Audio links	- 20% of exposed time can be saved (decontamination of reactor cavity).
Working space	- Not very congested area: increase up to 20 % of exposed time. - Highly congested area: increase up to 40 % of exposed time.

The "Benefits" of ALARA Programs

After the discovery of cracks on some reactor vessel head penetrations in 1991, it was decided to inspect and, if necessary, repair part of the 900 MW and 1300 MW units' vessel heads in France. Because of the urgency of the situation, and in the absence of feed back experience in this domain, the first operations didn't benefit from a good preparation. This situation leads to an "abnormal" number of mishaps. The application of a specific ALARA program for these post-incidental jobs started by the beginning of 1992. Given the number of involved units, and the great haste of operations, the degree of integration of ALARA procedures differed largely from one operation to another. The analysis of the average percentage of mishap dose for the same operations as a function of the degree of integration of ALARA at the different stages of the preparation, follow up, and feed back experience analysis, shows a direct link between these two factors (see Table 3).

Table 3. Average percentage of mishaps for 22 jobs on reactor vessel heads

Degree of integration of ALARA programs	Average percentage of dose due to mishaps (min-max)
No application of a structured ALARA procedure.	70 % (50 - 80)
No specific ALARA preparation, but application of the ALARA procedure during the operation.	40 % (30 - 50)
ALARA preparation and follow up, but no full technical control of the operation.	30 % (15 - 40)
ALARA preparation and follow up, and use of feed back data from previous operations.	10 % (0 - 30)

At the beginning of 1993, EDF estimated that 5 man-Sv had been saved on the vessel head operations by implementation of ALARA programs.

CONCLUSION

The integration of ALARA within work management procedures is obtained by analyzing precisely the operations from the angle of their associated exposed time. The latter is influenced as much by good general organization of tasks (including planning, preparation of working areas, etc.), as by some specific actions improving working conditions. We have seen here some quantification of factors in terms of their direct impact on time of exposure. The direct impact of general organization is more difficult to quantify. Nevertheless, some studies on causes of mishaps occurring during outage maintenance jobs in French nuclear power plants have shown that up to 30% of mishaps' dose can be attributed to organization problems (planning, scheduling...).[4] A study of outage organization in four different nuclear power plant from various countries has then been performed. It allowed to highlight some "good practices" favoring the implementation of ALARA programs.[5] The main conclusion can be summarized in six points :

- Integration of radiation protection criterion in the overall outage process, from planning stage to feed back experience.
- Management commitment.
- Effective coordination and collaboration of all sections concerned by the outage.
- Important decision making power of health physicists.
- Feed back data bases for jobs, doses, dose rates...
- Motivation and commitment of all actors towards ALARA principles.

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Author Biography

Caroline Schieber has been a researcher for four years in the Nuclear Protection Evaluation Centre (CEPN) in France. she has been working mainly on the implementation of ALARA programs on nuclear power plants through work management actions and in the definition of a system of reference monetary values for the unit of collective dose. She has a masters in economics.

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EXPERIENCE WITH ALARA AND ALARA PROCEDURES IN A NUCLEAR POWER PLANT

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SUMMARY

The nuclear power plant Borssele is a Siemens two-loop Pressurised Water Reactor having a capacity of 480 MWe and in operation since 1973.

The nuclear power plant Borssele is located in the southwest of the Netherlands, near the Westerschelde River.

In the first nine years of operation the radiation level in the primary system increased, reaching a maximum in 1983. The most important reason for this high radiation level was the cobalt content of the grid assemblies of the fuel elements.

After resolving this problem, the radiation level decreased to a level comparable with that of other nuclear power plants.

In the first few years of operation, the annual collective dose was relatively high, 4,000 - 5,000 mSv, but it decreased to 1,200 - 1,500 mSv, the target being a dose of 1,000 mSv in a year of normal operation. These results have been achieved by taking various measures, among them are the following:

- optimization of the water chemistry,
- restricted testing and maintenance management programme,
- dose control and limitation of the individual day and annual dose,
- implementation of an ALARA group (consisting of people with various disciplines),
- implementation of ALARA procedures:
- justification
- optimization.

In the ALARA procedures, recommendations are given on how to estimate the financial value of the received doses.

GENERAL

The Nuclear Power Station Borssele (KCB) was commissioned in 1973. It is a two-loop Pressurized Water Reactor built by KWU.

Right from the start, much attention had to be given to radiation aspects by the station management. After the higher dose values in the beginning of operation, a falling tendency became visible, especially after 1984.

The nuclear power plant Borssele is an older plant with relatively high radiation levels. The collective dose will therefore be higher than in the case of more recent plants.

By means of a strict regime of dose limitation and application of ALARA, it is now achievable to realize a normal year of operation at about 1,200 mSv. The target is to come down to lower than (<) 1,000 mSv.

RADIATION LEVELS

Radiation and Contamination Levels of the Primary System

The radiation levels of the primary system had risen strongly from 1973 to 1982 (see Figure 1). The increase was so disquieting that drastic measures were considered to keep working possible in the primary system, especially in the steam generators.

The nuclide that determines to a large extent the radiation level in the primary system is ^{60}Co. In 1982 it was established that the high ^{60}Co contamination in the primary system was caused by activation of the Co content of the nickel layer present in the grid assemblies of the fuel elements.

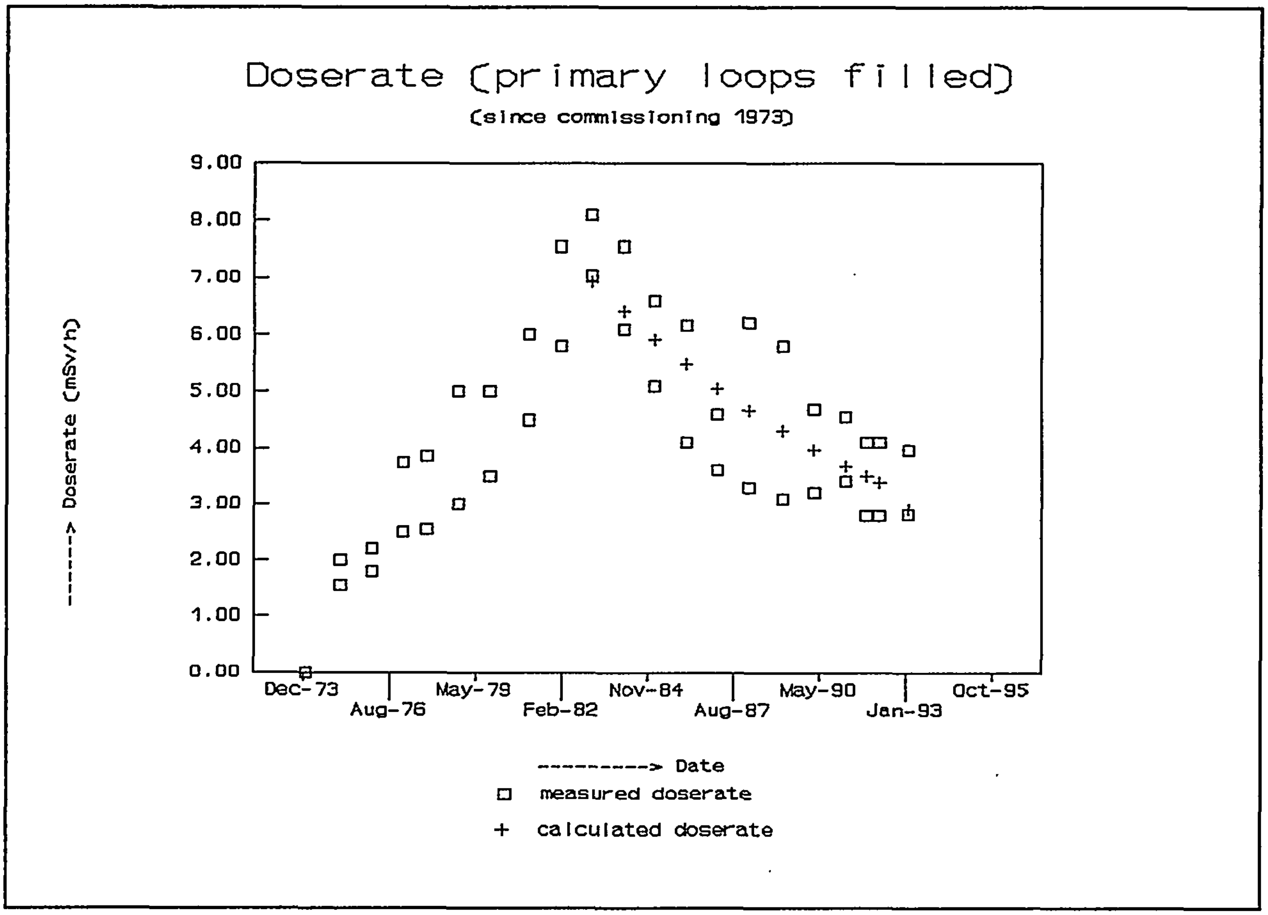
By using grid assemblies with little Co content after 1983, the radiation level of the primary system has strongly dropped since that time. At the moment, the radiation level is again comparable to that of 1977 and a falling tendency can still be perceived.

In addition to changes connected with the fuel elements, measures are also taken regarding operation. Maintaining pH values of the primary coolant between narrowly specified boundaries, together with optimal cleansing, has also contributed to the decrease of the radiation level of the primary system. Also, measures were taken with the intent to make work in strong radiation fields possible.

These measures comprise:

- practicing with dummy equipment, for instance, to train to open and close manholes of the steam generators,
- acquiring of advanced equipment for remote control (e.g., a fingerwalker),
- placing protective walls.

Figure 1: Dose rate primary loops since 1973



In 1987 an investigation was started into the feasibility of a decontamination of the entire primary system. For this purpose the behaviour of many materials under influence of various decontamination fluids was tested and examined by KEMA (a Dutch laboratory for testing of materials and equipment) and KWU.
However a great uncertainty whether such a decontamination would lead to positive results, together with an uncertainty whether the materials would or would not be affected by the decontamination process, lead KCB to the decision not to pursue such a large decontamination further.
This decision was also influenced by the steadily decreasing radiation levels since 1983.

In 1991 gammaspectrometrical measurements were done of the primary system by means of a mobile HpGe-detector.
The measurements were carried out in the hot leg of loop 1 and the cold leg of loop 2.
A summary of the measured surface contaminations is given in table 1.

Table 1: measured surface contaminations in the hot leg of loop 1 (YA001Z001) and the cold leg of loop 2 (YA002Z011).

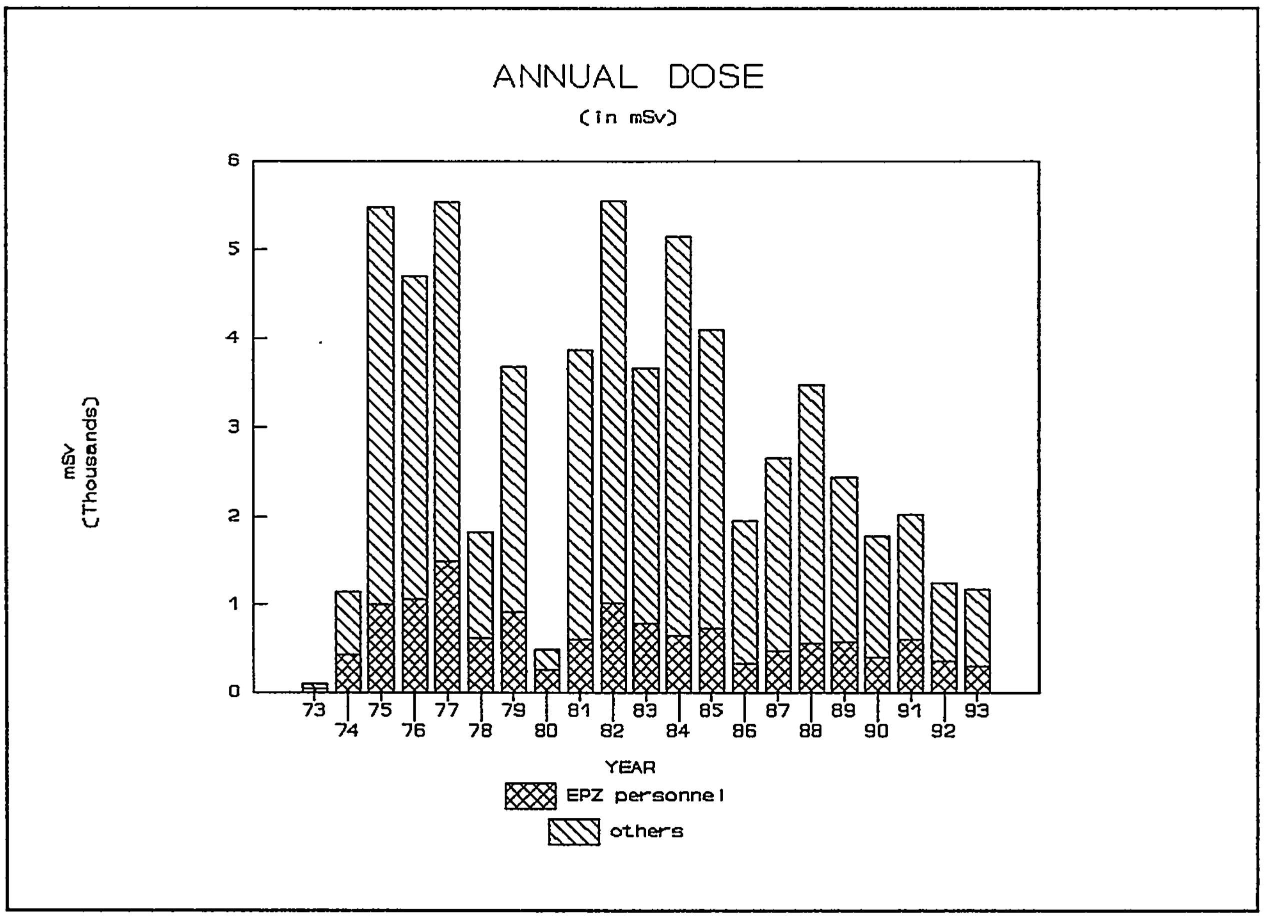
nuclide	surface contaminations			
	YA001Z002		YA002Z011	
	kBq.cm^{-2}	%	kBq.cm^{-2}	%
^{124}Sb	47	4,9	26	2,0
^{58}Co	135	14,1	·108	8,4
^{60}Co	778	81,0	1.152	89,6

From these measurements it can be concluded with some carefulness that in the cold leg the contamination is about 35% higher than in the hot leg.
The above mentioned difference can also be seen from the dose rates at the surface of the pipes which are yearly measured by dose rate measuring devices.

DOSES

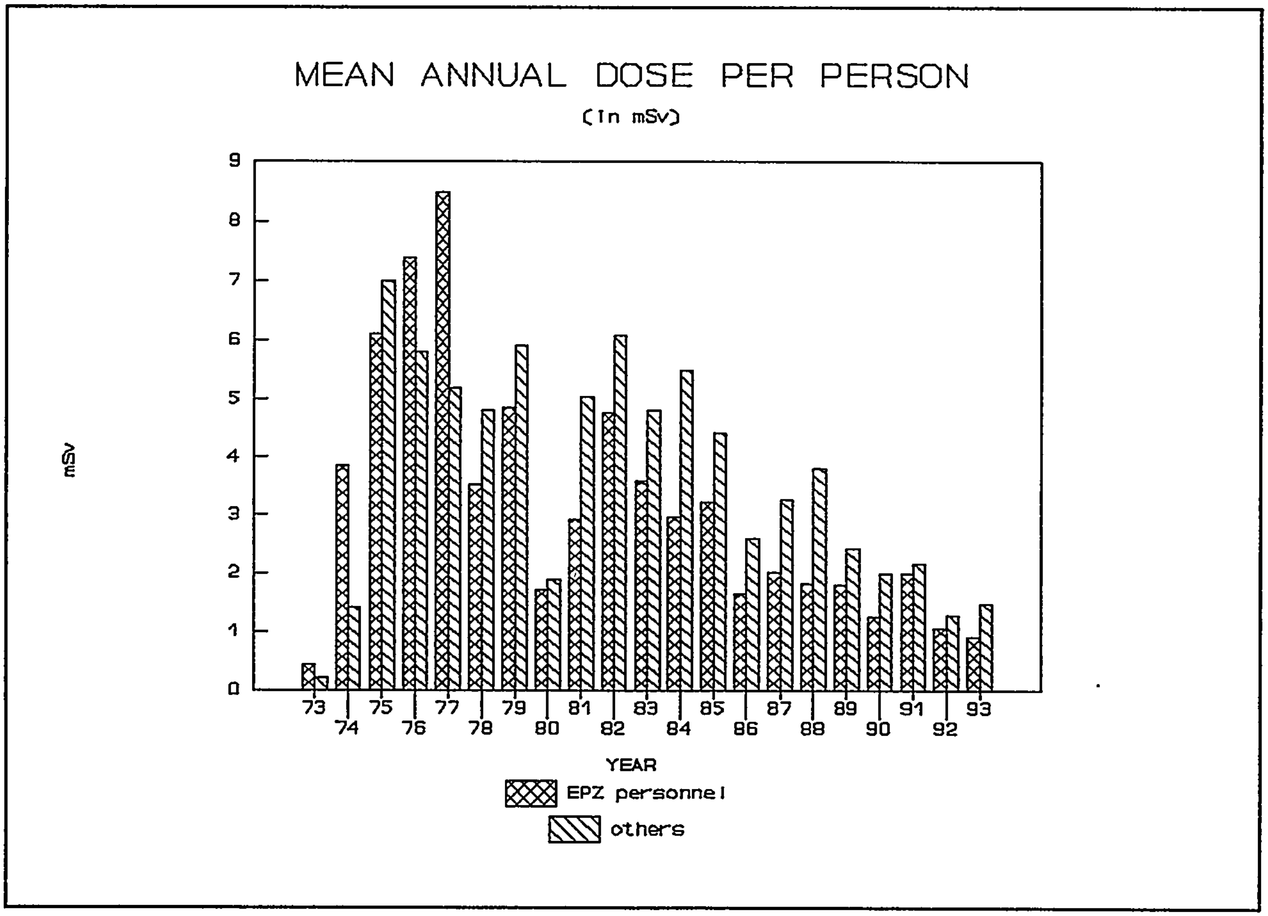
The received collective radiation dose is given below and differs strongly from year to year.
In 1980 the KCB reached the lowest yearly dose (482 mSv) whereas in 1982 the highest yearly dose was registered.

Figure 2: Annual dose



It is of course clear that the amount of work in the installation and the prevailing radiation levels influence directly the collective dose. The dose received during the outage periods amounts as an average over the last 10 years to 80% of the yearly dose.
From the figures it can be seen that in general 80% of the yearly dose is received by contractors (non KCB personnel).
But it cannot be concluded for that matter that the mean individual dose received by others than plant personnel must therefore be higher.
The mean individual yearly dose is given in figure 3.

Figure 3: Mean annual dose per person



From this figure can be derived that over the last years the mean individual yearly dose for KCB personnel is about 2 mSv and for other personnel 2-4 mSv. It should be mentioned that the given dose for outside personnel is only the dose received in Borssele.
The maintenance department and the radwaste department have the highes mean individual yearly dose.

The KCB uses besides the regulatory dose limits the following self-imposed dose limits; these limits are laid down in procedures and are valid for a radiological worker catagory A:

- daily dose : max. 1 mSv
- with permission of the radiation protection department : max. 4 mSv
- in specific situations : max. 10 mSv
- yearly dose (calender year) : max. 15 mSv.

Besides the above mentioned limits KCB has the objective to limit the dose which a radiological worker catagory A receives as an average during a number of years to maximally 10 mSv per year.
As far as the collective yearly dose is concerned the goal is to receive less than 1000 mSv during a normal year of operation.

RELEASES OF RADIOACTIVITY

The discharge of gaseous and liquid radioactive substances has always been far below the licensed limits.
For noble gases and I-131 it is given in figures 4 and 5.

Figure 4: Release of noble gases through ventilation stack

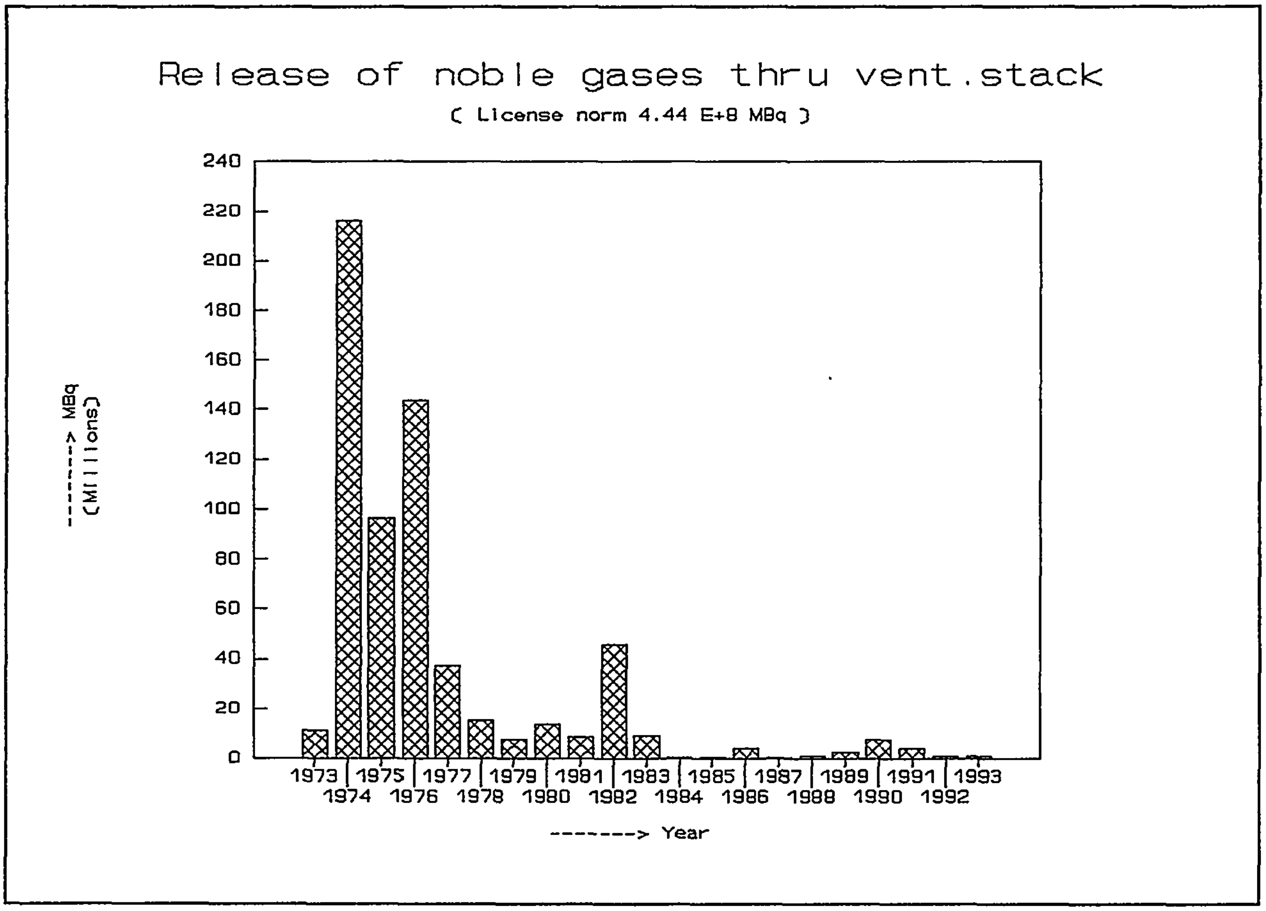
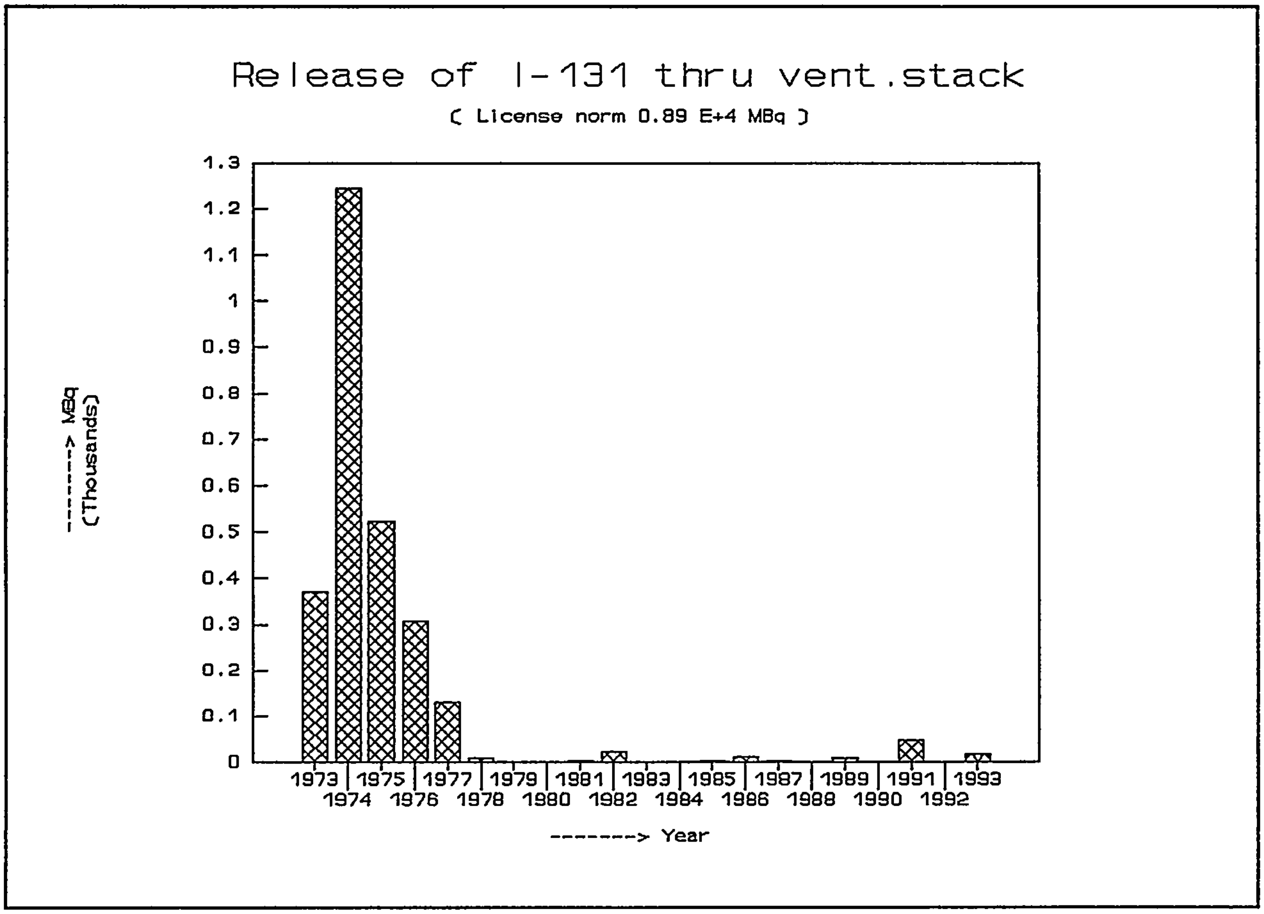


Figure 5: Release of I-131 through ventilation stack



During the first years the release of noble gases and ^{131}I (Iodine) were substantially higher than during the last years.
These higher releases were caused directly by defective fuel elements.

ALARA

In 1977 ICRP 26 came into force, herein the basic principles, justification, optimization and limitation regarding working with radioactivity were clearly established.
At the nuclear power plant Borssele the ALARA principle, also a result of the adoption of ICRP 26, has been put into practice since many years. In 1988 an ALARA committee had been set up. The object of this committee is to advise the management about measures which have to be taken to lower the dose on the basis of the ALARA principle.
The members of the ALARA committee have their own specific expertise and also the most important and involved departments are represented by them.
For a couple of years ALARA procedures are used. In these it is described how one should act to ensure that ALARA is sufficiently applied.
At the moment two ALARA procedures have been developed, one procedure in connection with modification of the installation and one procedure in relation to maintenance.

The following subjects are treated in these procedures:

- justification of the work,
- assessment of alternative solutions,
- influence on the dose
- selection of the optimal solution from the ALARA point of view,
- drawing up an ALARA report.

The implementation of these procedures (within the QA, quality assurance, regime) is laborious and takes a lot of time.

In these procedures a guideline is also given on how to express radiation dose in terms of a financial value. At KCB, a value of 1 million Dutch florin per Sievert is used.

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EDF EXPERIENCE WITH "HOT SPOT" MANAGEMENT

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ABSTRACT

During the past few years, "hot spots" due to the presence of particles of metal activated during their migration through the reactor core, have been detected at several French pressurized water reactor (PWR) units. These "hot spots," which generate very high dose rates (from about 10 Gy/h to 200 Gy/h) are a significant factor in increasing occupational exposures during outages. Of particular concern are the difficult cases which prolong outage duration and increase the volume of radiological waste.

Confronted with this situation, Electricité de France (EDF) has set up a national research group, as part of its ALARA program, to establish procedures and techniques to avoid, detect, and eliminate of hot spots. In particular, specific processes have been developed to eliminate those hot spots which are most costly in terms of occupational exposure due to the need for reactor maintenance.

This paper sets out the general approach adopted at EDF so far to cope with the problem of hot spots, illustrated by experience at Blayais 3 and 4.

INTRODUCTION

Hot spots are very small individual particles which generate high local dose rates. Almost all hot spots in France consist of activated erosion or corrosion products. Seven units are currently affected by this in France:

- Tricastin 1, 1984 - 3rd cycle - 900 MW unit
- Blayais 3, 1984 - 1st cycle- 900 MW unit
- Saint-Laurent B2, 1988 - 5th cycle - 900 MW unit
- Dampierre 3, 1989 - 7th cycle - 900 MW unit
- Cattenom 1, 1989 - 2nd cycle - 1300 MW unit
- Blayais 4, 1990 - 6th cycle - 900 MW unit
- Chinon B2, 1993 - 8th cycle - 900 MW unit

Hot spots have very significant consequences: they affect exposure, increase costs, and lengthen the unavailability of units:

- significant increase in the collective dose for the outage the first year it appears: 50% increase in the collective dose at Tricastin 1, Cattenom 1, and Dampierre 3,

- risks of internal exposure,

- increase in the length of unit outages owing to the need for longer system flushing operations and more delicate maintenance operations,

- increase in maintenance costs, and
- increase in the volume of waste produced.

In order to gain a better understanding of the phenomena involved and to implement the appropriate campaigns, EDF has adopted a dualistic approach to the problem:

- creation of a national group and compilation of good practices from experience feedback ("hot spots" file),
- creation of local groups to apply the strategy to take account of the specific requirements of each site.

The general approach adopted for managing hot spots covers both technical and organisational matters:

- technical, in terms of effective detection and location and using an analysis method to determine the root cause of the problem so that it can be eliminated more easily.
- organisational in terms of making the main players aware of the problem so that effective protective measures can be taken and good practices promoted.

The section below shows how this approach was applied at Blayais Power Plant.

IDENTIFICATION OF THE ROOT CAUSES OF HOT SPOTS: EXPERIENCE AT BLAYAIS 3 AND 4

Blayais 3 and 4 were first linked up to the French national grid in 1983. They are among the small number of French units which, initially loaded with FRAGEMA type fuel, were loaded with ANF assemblies from 1985 onwards in the case of Blayais 3 and from 1986 in the case of Blayais 4. Both units now contain 100% ANF fuel. Since 1984, in the case of Blayais 3, and since 1990, in the case of Blayais 4, Blayais nuclear power plant has been confronted with the problem of hot spots which play a considerable part in increasing exposure during unit outages. In order to remedy these problems, the plant management has, since 1992, adopted an ALARA policy to identify the causes of this high level of exposure and eliminate them, in particular by a policy of dose rate reduction and hot spot elimination.

Detecting and Locating Hot Spots

The analysis of how the hot spots in Blayais 3 and 4 have developed can be summed up as follows:
The first occurrences of cobalt-60 hot spots in Blayais 3 were detected as early as the first unit outage in 1984. They were followed during the fourth unit outage (1988) by large-scale clad spalling, and then in the seventh outage (1991) by the appearance of silver-110 following wear to the clusters and finally by further occurrences of cobalt-60 hot spots in the 1992 outage.

The situation is less complex in the case of Blayais 4. Cobalt-60 hot spots appeared during the sixth and ninth unit outages (1990 and 1993 respectively).

Hot spots are generally propagated in a similar manner in all units. The areas most prone to hot spots are the fuel ponds, reactor coolant system, drainage systems (Reactor Cavity and spent fuel pit cooling and treatment system), systems connected to the drainage systems (blowdown, venting and nuclear drain systems, residual heat removal systems and chemical and volume control systems) and certain systems connected to the reactor coolant system.

Analysis of the Root Cause of Hot Spots

The national working group recommended the following four-stage procedure for discovering the initial metallurgical composition of the hot spots and determine their cause:

1) Analysis of hot spot composition:

   - gamma spectrometry associated with measuring the dose rate upon contact with the hot spot,
   - analysis of dimensions,
   - chemical analysis in the laboratory once the hot spot has been isolated (if it can be isolated).

2) Estimation of the active life and radioactive half-life after activation.

3) Determination of the initial metallurgical composition of the hot spots (using previous results or by consulting a table of ratios of activation products).

4) Search for past events and comparison with analysis results.

At Blayais Power Plant, radiochemical analyses were carried out with the unit in various states: in operation and in outage.

With the unit in operation, peaks appear for the activities of corrosion products in the reactor coolant system, and vary with operating conditions during load following. Gamma spectrometry carried out in recent years has shown a high amplitude in the cobalt-58 peaks (in the region of 100 MBq/m^3) with a cobalt-58: cobalt-60 ratio of less than 5:1.

One other matter worth noting is the simultaneity of the peaks for cobalt, niobium-95, and zirconium-95, which are the main components of the cladding.

It would seem that with the unit in operation, the contamination, which only weakly adheres to the fuel cladding, tends to come away in the coolant flow, taking with it fragments of the cladding.

Several sorts of analyses were carried out with the unit shut down.

Mention must first be made of the contamination measurements taken in the reactor coolant system by means of g spectrometry on the U-tubes, the hot and cold legs and the primary and secondary sides of the steam generators. These measurements indicated a considerable amount of activity deposited in Blayais 4 in the form of cobalt-60, compared with the average for the population of plants; cobalt contamination is presumed to occur between cycles 1 and 7, in view of the evolution curve. Measurements on Blayais 3 indicate a normal quantity of activity deposited in the form of cobalt-60.

Furthermore, specific g spectrometry measurements were carried out during the 93 outage on hot spots consisting of cobalt-60 only (nuclear sampling system, residual heat removal system and reactor cavity and spent fuel pit cooling and treatment system). These analyses showed that contamination propagated through the systems after fuel handling operations during unloading. This meant that contamination in the systems adjoining the reactor coolant system would then probably be spread by movements of water. As a matter of fact, it was found that radioisotopes such as cobalt-60, zirconium-95, niobium-95, and chromium-51 were only found at the bottom of the fuel pond after unloading operations, showing that contamination is caused by flux. Again, it would seem that the corrosion products fixed to the fuel come loose during handling operations, carrying with them fragments of cladding, explaining why hot spots of zirconium-95 and niobium -95 have been observed.

This clearly shows that fuel is a vector in contamination by flux.

Finally, samples were scraped from the cladding of three fuel pins in the reactor building and analysed. In addition to the major components of the cladding (zirconium-95 and niobium-95), no representative quantities could be found of other radioisotopes. It would seem that deposits of chromium and cobalt oxide all come loose during handling operations.

We can therefore conclude that in the lack of other sources of contamination, the situation should improve with time, if reactor building ponds are carefully cleaned. This was observed in Blayais 3, where no more chromium-51 can be found and where there are no new hot spots, just old hot spots carried by the movements of water.

Movements of water are therefore the contamination vector in the absence of flux.

The root causes of contamination are doubtless linked to the degradation of stellite parts which make up certain items of equipment in the reactor coolant system and associated systems:

- wear of the mating surface of certain valves (charging pumps discharge),
- degradation of the mating surface of the self-aligning bearings and anti-rotation pins of reactor coolant pump bearings,
- metal pick-up at the radial keys of reactor vessel bottom internals.

The results of these analyses show that one possible scenario for the cause of contamination at Units 3 and 4 could be:

1. **Stellite contamination,**
2. **Migration of contamination from fuel cladding,**
3. **Loosening of deposits during handling operations (even during operation),**
4. **Spread of contamination by movements of water.**

APPROACH ADOPTED AT THE BLAYAIS SITE TO REDUCE EXPOSURE

System Flushing

System flushing by shift teams and decontamination by the General Service Departments under the co-ordination of the Industrial Safety and Radiological Protection Section are unarguably the major factors in reducing dose rates.

The drains of the nuclear island vent and drain system are flushed at the start of the outage. Further clear improvements in results can be made, where this is possible, by fitting out the low points to optimise venting and by using mobile filtration equipment.

Prior to unloading, the pond drain line is flushed through a filter at the bottom of the pond, depending on the dose rates in the lines. This is a delicate operation; should the filter leak, contamination would spread into the reactor cavity and spent fuel pit cooling and treatment system and the nuclear island vent and drain system.

Still depending on the dose rates, the high-pressure safety injection system, U-tubes and medium-pressure safety injection systems are flushed. One of the sensitive issues is that of flushing the pressuriser surge leg; the pressuriser spray is used to perform this task at the dissimilar-metal weld when draining the reactor coolant system to bring the residual heat removal system to mid-loop operation. Flushing is indirect, and therefore of limited effectiveness.

After unloading, following large-scale contamination of a nuclear island vent and drain system header in the containment annulus (200 mSv/h) for collecting the drains from the U-tubes and safety injection system accumulator tanks, flushing was carried out but proved to be of limited effectiveness since much of the contamination had already become fixed.

Finally, the residual heat removal system was flushed, thereby reducing the ambient dose rate around the heat exchangers by a factor of ten.

Decontamination

As shown above, the bottom of the reactor building pond must be decontaminated after unloading, so as not to spread contamination in the systems.

Other decontamination operations can be carried out. In particular:

- decontamination of the steam generator water boxes using a high-pressure water lance at the beginning of the outage; the exposure cost of this operation is high, and so should only be carried out if the steam generators are highly contaminated or as part of a large-scale maintenance program,

- decontamination of the sump at the bottom of the reactor building with its associated tank; the exposure cost of this operation outweighs the few advantages it might have. A water filtration/circulation system is used for the decontamination, and should be replaced by a mechanical process,

- ultrasonic decontamination of pipework valves; this process gives good results.

Scheduling Maintenance Operations

A considerable reduction in exposure can be achieved by the constant concern during the outage of ensuring that the systems are not dewatered.

In the case of the secondary side steam generators, a schedule is drawn up and distributed to all the relevant persons in charge of maintenance operations to ensure that the operations are carried out with the steam generators full. This good practice will be developed during future outages.

Coordination of Lead Shielding Work

The use of shields can bring about significant dose savings, but care must be taken to ensure that their installation does not entail additional dosimetric costs. Lead shielding work is therefore coordinated during the unit outage in the light of:

- the work planned,
- the various system flushing operations,
- the water levels.

Such co-ordination allows the amount of lead used to be quantified, the points where lead shielding work is systematically carried out to be located, and suggestions for improvement to be made. At the end of the outage, an end-of-job meeting is held with the contractor managers to analyse results and suggest improvements.

Training and Motivating Workers

Doses can be reduced by making the players aware of how they can modify their behaviour on the worksite. When contractors start working at the site they are made aware of the specific radiological protection problems at the unit, and the maintenance workers are taught simple actions for reducing their exposure. A dossier is drawn up for this, containing the following major elements :

- reason for the maintenance work,
- unit background,
- action undertaken by EDF (search for a root cause and a remedy),
- what is expected of the workers,
- actions which will promote experience feedback.

Exposure targets are set and maintenance work is planned in the light of dose rates as early as the joint plant/contractor manager preparation meetings. During the maintenance work, regular meetings are held with the work managers to solve the problems in real time.

An excellent example of this co-ordination involves the services. The services consists of four sections: scaffolding, heat lagging, pond decontamination and cleaning assistance/various decontamination operations. Analysis of the average dose for this area indicated a reduction of almost 35% between 1992 and 1993 through the policy of making the players aware of the problems and as a result of work by the Industrial Safety and Radiological Protection Section in preparing, monitoring and organising experience feedback for worksites.

CONCLUSION

Considerable progress was made with regard to reducing collective exposure during unit outages in 1993. Brainstorming organised by the power plant management, and carried out in close collaboration with the site Operations, Maintenance, Chemistry, General Services and Radiological Protection Departments, reduced the collective dose for Blayais 3 by 20 % and for Blayais 4 by 40 %. Nothing is ever completely certain in this area, and efforts have still to be made.

The program of actions at Blayais Power Plant is centered around two main areas:

- Identification of root causes: it is planned to inspect the radial keys of reactor vessel bottom internals at the next unit outage in order to validate hypotheses. Comparisons of radiochemistry will be carried out in operation and during shutdown on units using different fuel and on units without hot spots. Finally, in order to increase statistical sampling, the cladding of certain types of fuel elements will continue to be scraped and the samples will be analysed.

- Reduction of exposure: the awareness program will be extended to include services and operations workers.

  Furthermore, the possibility of a large-scale decontamination program for Blayais 3 during its ten-yearly overhaul is currently being considered as a means of improving the effectiveness of flushing, and to replace sections of pipe to which contamination has become fixed, and to develop a tool for decontaminating and inspecting the dissimilar-metal weld in the pressuriser surge leg.

Generally speaking, radiological protection is beginning to be recognised in the field as one of the technical components of the problems encountered. This approach should be promoted by adopting a global approach to problems, combining the various specialisms, professions and hence preoccupations, without being detrimental to the quality of interactions between the plants and head office. This would make the most of experience feedback.

Author Biography

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THE OPTIMISATION OF OCCUPATIONAL POTENTIAL EXPOSURES
Preliminary considerations

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ABSTRACT

One of the major innovation brought about by the ICRP 60 recommendations and emphasised by the ICRP 64 publication, is the introduction of the concept of potential exposures into the system of radiological protection. Potential exposures are characterised by "probability of occurrence lesser than unity" and "radiological risks exceeding normal levels" where normal must be interpreted as not exceeding the planned routine exposures. It is then necessary to develop consensual methods to look for and choose the optimum scenarios (i.e. those for which probability of events and possible consequences have been reduced as low as reasonably achievable). Moreover, the boundaries for the unacceptable levels of risks for workers should be defined, as well as reasonable risk indicators.

The aim of this paper is to discuss the actual changes in the field of occupational radiological protection, induced by the potential exposure concept with particular emphasise on the optimisation of protection.

INTRODUCTION

One of the major advancements brought about by the ICRP 60 recommendations [1] and emphasised by the ICRP 64 publication [2], is the introduction of the concept of potential exposures into the system of radiological protection. Much of the discussion that has followed, focused on the impact and appropriateness for public protection and nuclear safety approaches, especially relating to the prevention of accidental situations and the mitigation of their consequences. This concept has needlessly thrown people involved in nuclear safety and in radiological protection into a real confusion, essentially centering on the usefulness or appropriateness of such a concept in their respective domains. One may hope that ICRP 64 or the future INSAG publication on potential exposures will definitively close the discussions.

The aim of this paper is to discuss the actual changes in the field of occupational radiological protection, induced by the potential exposure concept. The potential exposure concept most certainly calls for new methods in the practical application of occupational radiological protection, in particular for the optimisation of protection.

BASIC CONCEPTS

Potential Exposures

From the ICRP point of view, a potential exposure *is an exposure that, while not certain to occur, can be anticipated as a result of introducing or modifying a practice ("human activities that increase the overall existing radiation risk"), and to which a probability of occurrence can be assigned. Such exposures involve considerations of risk which fall outside the general boundaries considered for normal exposures, being recognised that, if these exposures effectively occur, they may lead to interventions ("human activities intended to decrease the already existing radiation risk").* The potential exposures are then characterised by "probability of occurrence lesser than unity" and "radiological risks exceeding normal levels" where normal levels have to be interpreted as planned routine exposures.

Risk

The word "risk" has been debated for a long time because of its different definitions and interpretations. In a common way, risk is understood as the "possibility of a harmful effect". Probability is the most common indicator to express this possibility. However, the methodology of "effect" and "probability" assessment have to be adapted case by case depending on the size of the problem considered: the risk appraisal of the practice itself, an operation inside a given practice, or specific tasks included in a given operation that will not lead to investigations in the same ranges of probabilities and consequences. Even if the general objective of any risk assessment is a matter of establishing a quantified framework to help decision-makers in their final choices, it is clear that one specific model of risk assessment ("mathematical expectancy of fatal cancers", for example) could be well-adapted to one situation and totally unsuitable to another due to its possible multidimensionality.

THE MANAGEMENT OF POTENTIAL EXPOSURES

The Optimisation Of Uncertain Risk

The optimisation of radiological protection allows to consider the best use of resources in reducing the radiation risks to individuals. The ICRP 60 recommendations expressed that its broad aim should be to ensure that *the magnitude of the individual doses, the number of people exposed, and the likelihood of incurring exposures where these are not certain to be received, are all kept as low as reasonably achievable, economic and social factors being taken into account.*

The likelihood that a person will suffer a given detrimental effect is quantitatively expressed by:

$$P = P_s . P_e(D_s)$$

where P_s is the scenario probability (where scenario has to be understood as an unique combination of events, sequences, processes and procedures), and P_e is the probability of severe radiological effect related to the individual dose D_s arising from the given scenario and defined by the dose-response relationship model.

In the case of doses staying below deterministic thresholds, risk may be expressed by

$$R = P_s . c_R . D_s$$

where:

- $P_s = \left(1 - e^{-T.\frac{dp}{dt}}\right)$ is the likelihood of the event knowing that the value of P_s is not far from the probability rate $\frac{dp}{dt}$ (per year) only for practice whose duration T is in the order of one year.

- c_R is the nominal probability coefficient for dose ranges leading to only stochastic effects (fatal cancer or fatal cancer equivalent).

But, it must be underlined that in the case of doses which may lead both to stochastic and deterministic effects, the proportionality between risk (expected fatal effects) and doses is no longer an acceptable estimate. The definition of the harm indicator is no longer straightforward and the problem of the summation of probabilities of health effects of different nature must be added. Moreover, trade-offs between potential and actual exposures should be addressed.

Even if these problems are solved, optimisation aims to demonstrate that everything possible has been done to reduce as low as reasonably achievable (ALARA) both probability of events leading to potential exposures, as well as the individual (and collective) doses themselves, economic and social factors being taken into account. Thus, the main goal of optimisation is not to respect a tolerable level of risk but clearly to ensure that an acceptable residual risk level has been reached by reducing both the probability, P_s (action of prevention), and exposures, D_s (action of mitigation), independently or simultaneously.

The problem lies in the fact that it will be generally difficult to balance prevention vs mitigation (see Figure 1), especially when situations do not directly implicate installation safety (for which prevention is a priority). The optimisation techniques will have to be adapted to take into account this bi-dimensionality, emphasizes the cases where mitigation would be more cost-effective, or more useful, than prevention and vice versa. In other terms, optimisation should permit the most satisfactory allocation of resources and protection efforts towards reduction of potential consequences or towards reduction of the probability of events.

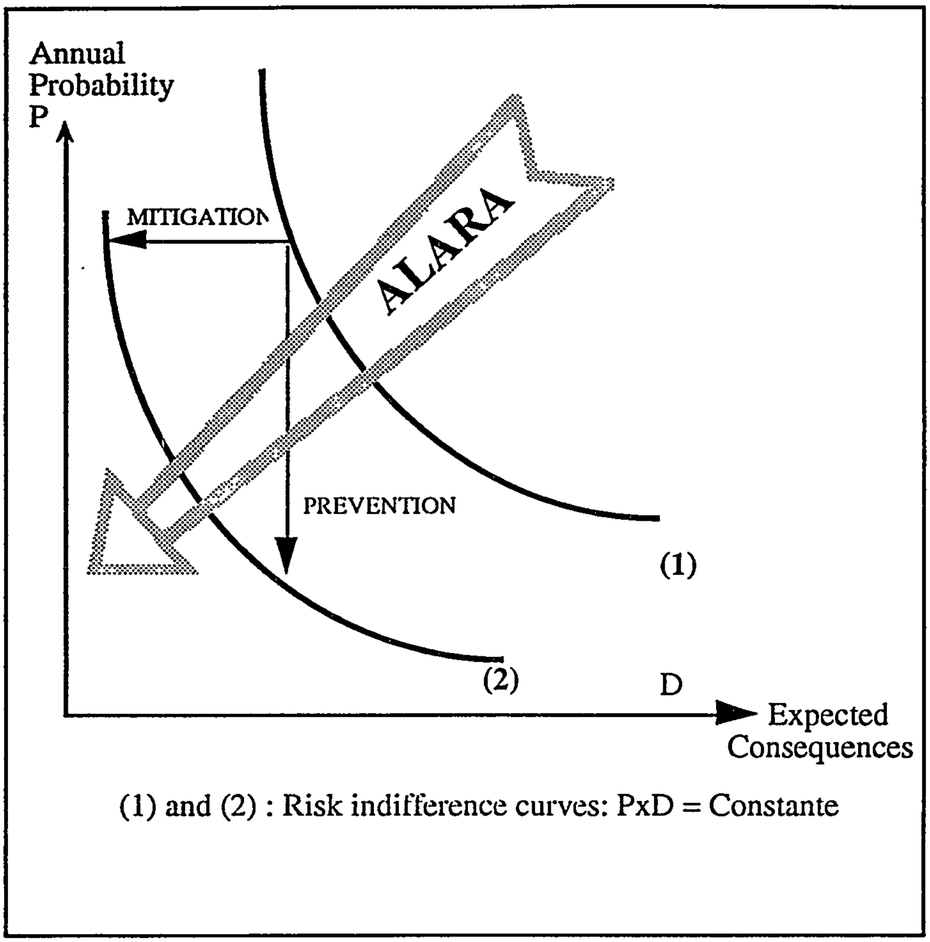

Figure 1. Reduction of potential occupational exposures ALARA

Risk Aversion

The acceptable level of risk for a given practice (i.e. the ALARA level of risk) depends, in fact, not only on the estimated level of the maximum individual and collective levels of risk but also on social, economic and other factors conveying with the individual perception of risks and reflecting the collective attitudes towards potential consequences at a given time. This last aspect is probably the most important point in the assessment of the acceptable risk and can be formalized by:

$$R_{acceptable} = f\left(R, A(D_s, P_s)\right)$$

where A, is a risk-acceptability function defining the individual aversion, according not only to the potential level of exposures but also, the probability of the event giving the dose. It will lead to considerations for aversion to high potential exposures or expected exposures dispersion, which are sometimes taken into account in cost-benefit analysis. But, it will also lead to considerations for aversion to allocate resources to the reduction of probabilities of potential events, with clearly uncertain results. This last point is probably a very novel aspect which could shed light on the decision making process.

In the case of aversion to exposures in a given group of individuals or between different groups, methods of optimisation should reveal preferences in the distribution of potential doses, especially since doses in the deterministic range are possible. Account must also be taken for the dispersion of potential individual doses and the possible risk transfers (for example, from workers to other workers, workers to public or, even the present to future).

A next step is to introduce methods to assess aversion for making decisions under uncertain conditions and to include monetary and non-monetary attributes. Techniques (based on maximisation of expected utility, stochastic dominance...) already developed by financial risk theorists [3] could be adapted in order to help decision-makers in their choices and judgments on the acceptability of practices or scenarios with occupational potential exposures.

In the case of the optimisation of occupational practices or operations (i.e. large operations for example, steam generator replacement, installations dismantling..., involving many workers, and smaller more specific and repetitive tasks involving smaller staff of specialists during shorter periods) the acceptability of the individual levels of risk is certainly easier to check, because it is dealing with better known and well-followed populations and with time- and space-limited potential consequences (i.e. with more limited risk systems of reference).

However, special considerations like aversion towards possible high exposures, risk dispersion in the worker population (considering their other possible tasks), possible risk transfers between them still exist. These aspects, which are not yet really taken into account for the normal exposure management, will have to be more carefully assessed or weighted in the case of potential exposures.

The Tolerability of Risks

As the individual dose limits restrict the field of the optimisation for normal exposures, risk constraints could be used to ensure that the level of potential exposures of a given practice is effectively under a tolerable level (i.e. to ensure that the practice is "safe enough"). In this context, it is necessary to define bounds above which the level of risk becomes unacceptable. Symetrically, it can be noted that lower bounds below which the risk level may be considered as negligible regarding both probability and exposures will be also useful to decision makers.

Tolerable upper risk bounds for accident consequences have already been proposed [4] to define probabilistic safety objectives.
Analogously, ICRP Publication 64 defined a range of annual probabilities vs individual whole body exposures, whereby constraints for potential individual whole body exposures could be selected (see Figure 2). The proposed ranges are large enough to fit with the case of public risks as well as the workers case. It shows that potential occupational exposures with annual probabilities greater than 10^{-2} should be considered as normal and constrained by the actual limits for normal occupational exposures.

More debatable is the fact that the system should *a priori* authorize potential exposures leading to possible deterministic effects in the range of annual probability from 10^{-6} to 10^{-5}. Anyway, the tolerable risk boundary scheme to be applied for incidental and accidental occupational situations, should take into account greater probabilities of events leading to higher potential exposures (i.e. above regulatory occupational limits). The reason to define the occupational risk boundaries is to verify *a priori* the tolerability of a given practice or scenario, even if it does not entail very important societal or collective consequences, but because it may lead to important individual harm to workers.

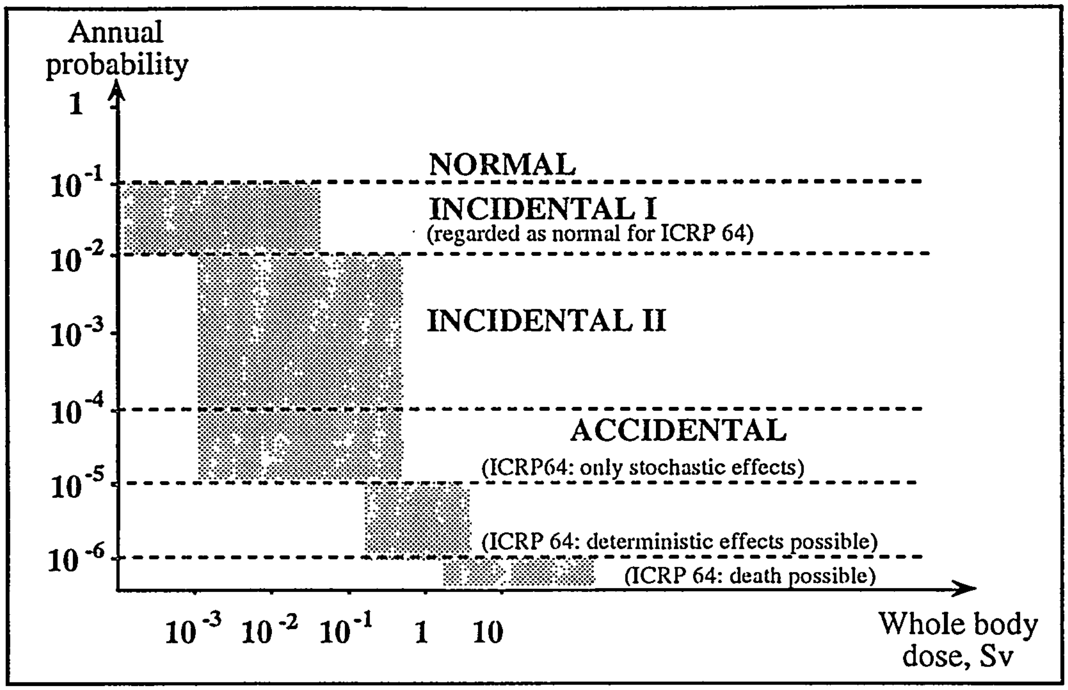

Figure 2. ICRP 64 domain of risk constraints for whole body potential exposures (workers and public)

In conclusion, the definition of what could be the boundaries for tolerable occupational risk of exposure is an important step in the potential management of exposures . But, they must be discussed and chosen taking into account different points of view of employers, authorities and regulators.

PRACTICAL CONSEQUENCES

Despite the difficulties related to the definition of acceptable and tolerable risks of exposures, the structuring and dissemination of potential exposure assessment through the radiation protection framework could effectively enhance the protection of workers. Some practical results and consequences from this assessment could then be included at the operational level.

Analysis of Occupational Incidents and Mishaps

In many countries, the control of occupational exposure and the statutory recording of occupational doses are based mainly upon the results provided by dosimeters. But, the increasing emphasis on optimisation of protection (ALARA) will gradually constitute adequate dosimetric data banks to relate the potential exposures of individuals to specific tasks. More specifically, the systematic recording of incidents and dosimetric mishaps during specific tasks would form a sound basis of feedback experience to identify where efforts must be given in priority. In most cases, better work management (organisation, training, motivation...) will provide high improvements [5], but sometimes and especially for well-managed scenarios, it could illustrate that radiological mishaps may be better prevented (by improvements in the reliability of system and materials) or mitigated (by improvements in the protection). Having such an approach for the detection of the most probable incidental causes of workers exposures and generalising it at the national -even international- level leads to the reduction of the collective and individual doses. Up to eighty percent of the doses during NPPs French outages are due to unplanned events [6], more often brought about technical mishaps and/or human errors; these mishaps must be well known, understood and quantified and therefore can be early prevented and mitigated.

Probabilistic Occupational Radiation Assessment

Techniques of identification of potential scenarios leading to unplanned exposures have to be developed. These techniques are well known in the safety field, but essentially focus on the reduction of risk related to accidents with off-site consequences or incidents with high consequences for the installation safety. They could probably be adapted to the radiological protection framework by defining dominant sequences of incidents and dose-event trees.

Resulting from deterministic ("postulation of initiating events" like human error or material failures) or from a probabilistic approach ("Probabilistic Occupational Radiation Assessment"), the methodologies should predict with high confidence the practices or scenarios which are leading to clearly untolerable radiological occupational risk. To perform such studies at the design stage of nuclear cycle installations, will obviously reduce the possibilities of incidental scenarios (prevention), as well as potential doses in case of their occurrence (mitigation).

CONCLUSION

The need to take into account potential exposures into the radiological occupational protection framework is not to demonstrate. Nevertheless, it will be necessary to develop consensual methods to look for and choose the optimum scenarios (i.e. those for which probability of events and possible consequences have been reduced as low as reasonably achievable). Moreover, the boundaries for the unacceptable levels of risks for workers must be defined, as well as reasonable risk indicators. This approach will turn radiological protection into actual risk management by controlling and limiting the scale of the exposures presented by the nuclear practices, and reducing the probability of incidents that might occur. At last, it must be keep in mind that even if it will lead to new occupational arrangements, the final aim of that process is to enhance both radiological protection and safety of workers.

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DEPARTMENT OF ENERGY ALARA IMPLEMENTATION GUIDE

RESPONSE TO THE HEALTH PHYSICS SOCIETY

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SUMMARY

In the August 1993 Health Physics Society (HPS) newsletter, the HPS Scientific and Public Issues Committee published a Position Statement entitled "Radiation Protection of the Public and the Environment." In this article, this HPS committee made the statement that they were deeply concerned by the trend for agencies to incorporate the ALARA concept as a regulatory requirement, without providing specific guidance as to what it means and how to implement it consistently. The HPS position paper was in response to the DOE notice on proposed rulemaking for Title 10 Code of Federal Regulations Part 834, "Radiation Protection of the Public and the Environment" (10 CFR 834). In the notice of proposed rulemaking for 10 CFR 834, the Department of Energy (DOE) defined ALARA as follows: "As used in this part, ALARA is not a dose limit, but rather a process which has the objective of attaining doses as far below the applicable limit of this part as is reasonably achievable" (10 CFR 834.2, p. 16283 of the Federal Register). The HPS position paper continues, "The section goes on to elaborate on what is meant by a process without providing sufficient guidance to assure uniform applicability of the process." Although this concern is directed towards the ALARA process as it relates to the environment, the Office of Health, which is responsible for occupational workers, shares the same definition for ALARA.

On March 14, 1991 the Office of Environmental Guidance (EH-23) issued a document to distribution within DOE "Guidance for Implementation of ALARA Requirements for Compliance with DOE 5400 series Orders: For Interim Use and Comment." This provided guidance to the field for the environmental aspects of ALARA contained in the environmental orders DOE Orders 5400.1 and 5400.5. It is expected that when 10 CFR 834 is published as a Final Rule, that an appropriate Implementation Guide will be issued.

On December 14, 1993 DOE's rulemaking on Occupational Radiation Protection was published as a Final Rule in Title 10 Code of Federal Regulations Part 835 (10 CFR 835), "Occupational Radiation Protection." This rule contains the same definition of ALARA as does the draft 10 CFR 834. When this Final Rule (10 CFR 835) was transmitted to the DOE sites, it was sent with 12 Implementation Guides (IGs) to provide guidance and discuss methods that are acceptable to the headquarters staff. Additional IGs will be sent as they are completed, to assist the contractors with compliance. One of the IGs provided was "Occupational ALARA Program", G-10 CFR 835/B2 - Rev. 0. This guidance document provides sufficient guidance to assure uniform applicability of the ALARA process.

This IG had originally been issued to the DOE complex for comment, in 1991, as draft ALARA IG "Occupational ALARA Program". 5.XXX, Rev. 1. Over 200 comments were received evaluated and incorporated where applicable. This guide was restructured to the new IG format, the Secretary's policy statement added and other changes made to update it. As an example, the key references are now to 10 CFR 835 and how to implement it. DOE Order 5480.11, the predecessor to 10 CFR 835, is also referenced since it still applies to a few installations. The Radiation Control (RadCon) Manual requirements are also provided. Therefore, the requirements and guidance are integrated in this one document to make it easier for the ALARA personnel to understand and implement in a reasonably consistent manner.

This does not imply that all the programs are going to be the same, because the degree of risk and the potential levels of exposure are different at the many different sites. A large diverse site, such as Hanford or Los Alamos, with many different sources of radiation, would have a large ALARA program whereas a small laboratory using only small amounts of radioactive material would have a correspondingly small program. It would not be cost effective or ALARA to have an elaborate program at sites where there is currently very little exposure and there is little likelihood that it will increase.

Author Biography

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SESSION 7A

PWR AND CANDU PRESENTATIONS

Co-chairs:

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PRIMARY WATER CHEMISTRY IMPROVEMENT FOR RADIATION EXPOSURE REDUCTION AT JAPANESE PWR PLANTS

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ABSTRACT

Radiation exposure during the refuelling outages at Japanese Pressurized Water Reactor (PWR) Plants has been gradually decreased through continuous efforts keeping the radiation dose rates at relatively low level. The improvement of primary water chemistry in respect to reduction of the radiation sources appears as one of the most important contributions to the achieved results and can be classified by the plant operation condition as follows.

- Hot Functional Test (HFT):	H_2 + LiOH added chemistry Dissolved Hydrogen (DH_2): 30 cc-STP/kg-H_2O Lithium (Li): 0.5 ppm
- Power operation:	pH control of 7.3 ± 0.1 at 285°C (Li_{max} = 2.2 ppm)
- Shutdown:	Low DH_2 control (≈ 0.5 cc-STP/kg-H_2O)

The effectiveness of the above improvements was verified and radiation levels at Japanese PWRs are expected to decrease further with the elapsed operation time.

INTRODUCTION

The purposes of PWR primary water chemistry are to assure the integrity of the component materials and fuel cladding, together with minimizing the out-of-core radiation field.
Fortunately, in the past there have been no such integrity problems in the primary side. That is why, the main concern of the primary chemistry has been focused on the reduction of the radiation field. Particularly, pH control has been considered as an important measure to suppress the transfer of the corrosion products (CP) and as a result, to reduce the radiation sources. Several investigations on the pH control improvement have been done in Japan, so far. Some of these results are described below. This paper touches also the state of other chemical control improvements at Japanese PWR.
The following three points are important to reduce the radiation sources:

(1) to suppress CP generation by minimizing the corrosion rate;
(2) to suppress CP activation by controlling CP transfer;
(3) to reduce CP inventory by removing CP from the primary system.

Further on, each item is described in relation to the corresponding plant operational condition.

PRIMARY WATER CHEMISTRY IMPROVEMENTS FOR RADIATION EXPOSURE REDUCTION

Radiation Sources Reduction

Improvement of Chemical Control during HFT (Suppression of CP Generation)

Generally, the initial corrosion rate is rather high. Therefore, in order to suppress both the CP generation and release from the primary component metal surfaces, it is very effective to form more stable oxide film on the component surfaces during the first heat up, prior to the power operation. To fulfil this, the chemistry control during HFT, performed at the end stage of the plant construction, was developed[1] as follows.
Steam Generator (SG) tubes are considered as a main source of the CP generation in PWR. Alloy 600 and Alloy 690 are the materials used respectively in the conventional and the new designed plants. Corrosion studies with these test materials were carried out under the chemical condition, shown in Table 1.

Table 1. Test conditions of HFT simulation

	Deaerated water	added LiOH	added H_2	added H_2+LiOH
Water quality	DO_2<10ppb	Li : 0.5ppm DO_2<10ppb	DH_2 : 30cc/kg-H_2O DO_2<10ppb	Li : 0.5ppm DH_2 : 30cc/kg-H_2O DO_2<10ppb
Temperature	286℃ (TT600 As Received) 292℃ (TT690 As Received)			
Testing time	600 hr			

Ni is a key element, because it is a major constituent of each Alloy and a parent element of ^{58}Co. The corrosion data for Alloy 600 are shown in Figure 1. From these data it was found that "H_2 + LiOH added chemistry" provides the best condition to suppress the CP generation from the Alloys.
The oxide composition formed under this chemistry was rich of Cr. In addition, from the comparison between the depth composition profiles of the oxide film on the each test specimen, it was found that the oxide film thickness in this case is the thinnest one. As a result of the above examination, "H_2 + LiOH added chemistry" was considered as the most effective water chemistry for HFT. For the first time this chemistry was applied to Tomari No.1 unit.
Figure 2 shows the comparison between the dose rates of the major components measured at the first refuelling outage of Tomari No.1 and another plant with the similar design but without H_2 +LiOH added chemistry during HFT. Almost all the measured dose rates at Tomari No.1 were lower than those of the other plant, as dose reduction accounted for ~ 40%. The H_2 +LiOH added chemistry contributed to about a half of this dose reduction. Note, that Fe and Ni concentrations in the primary coolant measured during the HFT at Tomari No.1 were also at relatively low level.
Based on the Tomari No.1 experience, the "H_2 +LiOH added chemistry" has been recognized as the most beneficial chemistry for HFT and applied to the new units as follows.

DH_2: 30 cc–STP/kg–H_2O , Li: 0.5 ppm

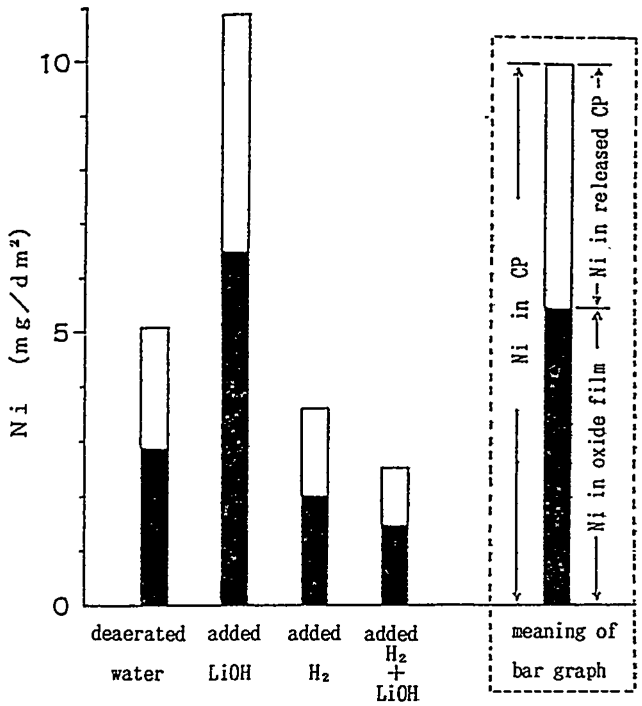

Figure 1. Relations between water quality and amount of Ni in CP (Alloy TT600)

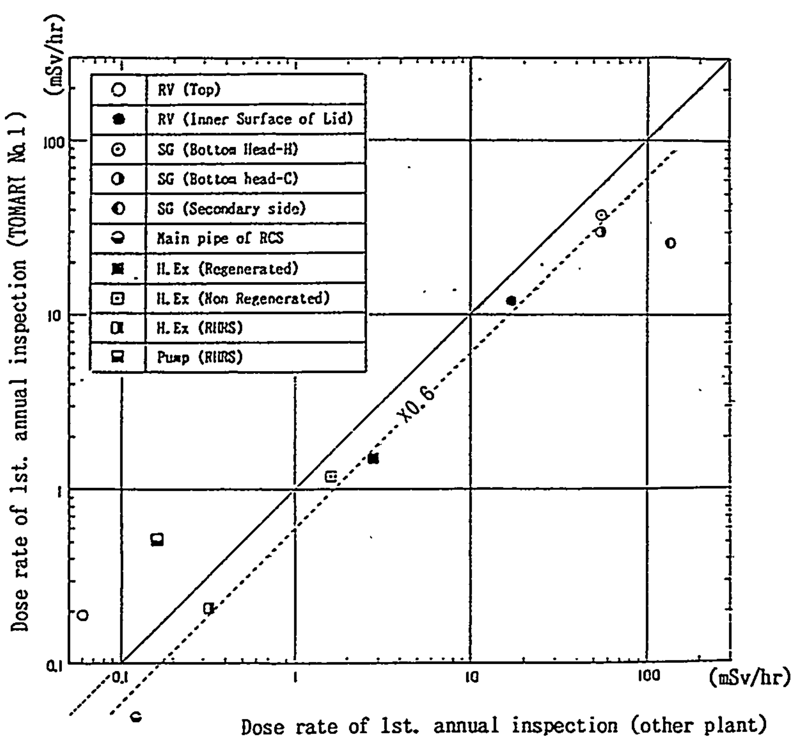

Figure 2. Comparison of dose rates between TOMARI No.1 and other plant

Improvement of pH Control during Power Operation (Suppression of CP Activation)

To suppress the CP activation it is necessary to minimize the crud deposits on the fuel rods and mainly to prevent the precipitation of ionic matter. Many investigations have been carried out worldwide to measure the CP solubility. Abe et al. measured the Fe, Ni, Co solubilities from a model substance (nickel–cobalt–ferrite) simulated the composition of the crud analyzed in the Japanese plants[2].
Based on the measured data, a solubility formula for each element was derived as a function of pH and temperature. Figure 3–5 represent each solubility curves calculated by those formulas.

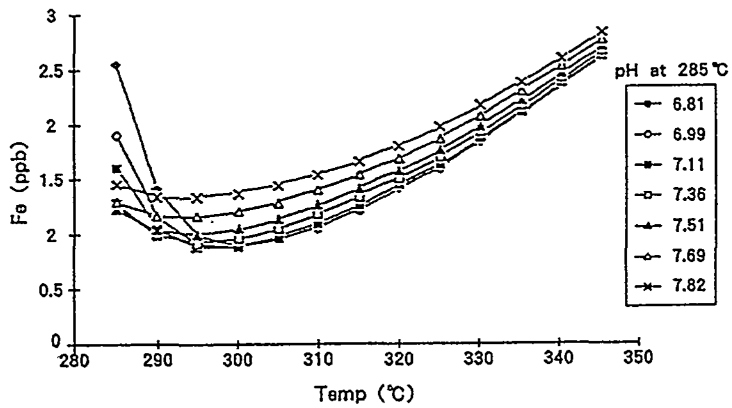

Figure 3. pH and Temp. Dependency of Fe Solubility

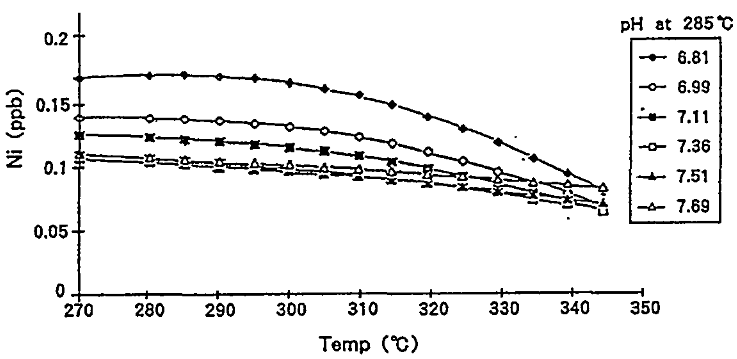

Figure 4. pH and Temp. Dependency of Ni Solubility

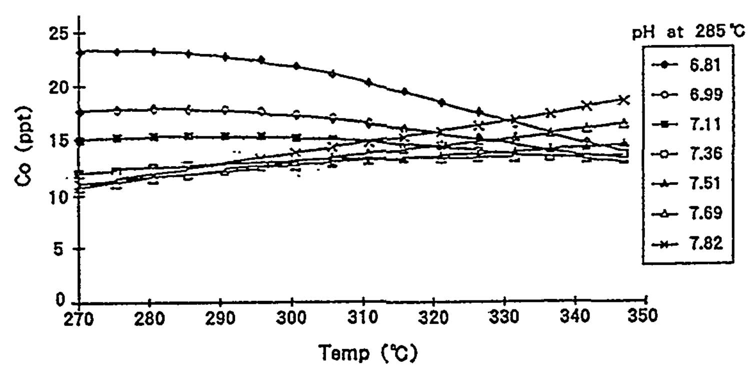

Figure 5. pH and Temp. Dependency of Co Solubility

The optimum pH which minimizes the out-of-core radiation field was estimated by CRSEC code[3], where the solubility relations were included. The optimum pH was evaluated of around pH 7.3 at 285°C. Considering the limitation for further rising of Li concentration, three typical cases were selected (Table 2 and Figure 6) to evaluate the actual effectiveness on the out-of-core radiation field reduction.

Table 2. Evaluation Results by CRSEC Code

Case	Target pH at 285	B at BOC (ppm)	B at BOC (ppm)	Upper limit of Li (ppm)	1st cycle	Average
0	6.8	1100	50	2.2	base	
1	7.0	1100	50		−3.1	−8.3
2	7.3	1100	50		−3.8	−11.1

Based on Domestic Data

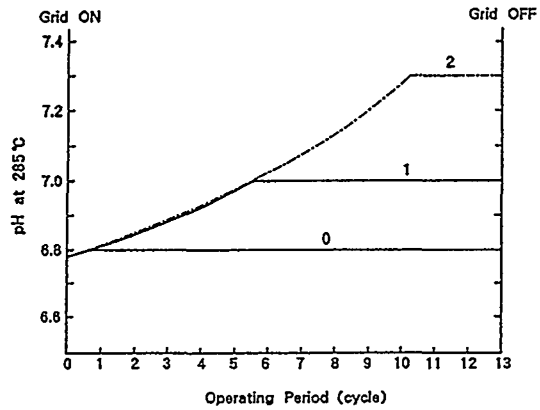

Figure 6. Case study of pH Control

Mitsubishi had recommended pH of 7.0 ± 0.2 at 285°C (Li_{max} = 2.2 ppm) as the optimum pH control during the power operation until recently.
At present, a shift to the higher pH control of 7.3 ± 0.1 at 285°C (Li_{max} = 2.2 ppm) based on the above results is proposed by Mitsubishi (Figure 7). Now in Japan, many plants operate following this pH control.

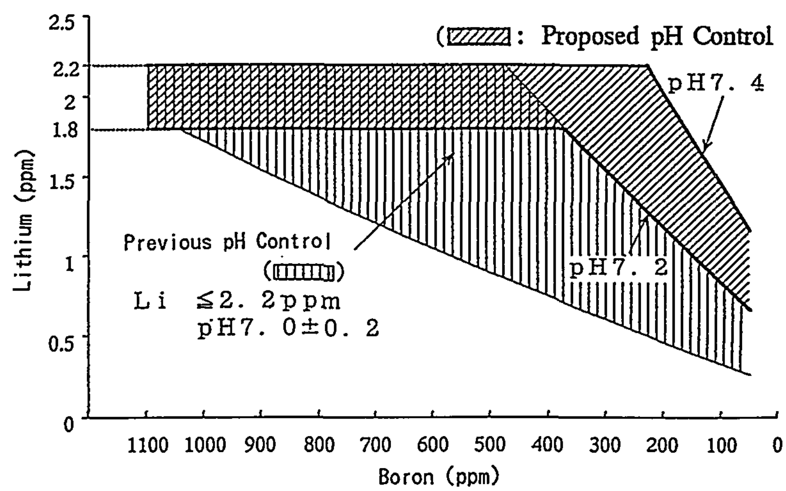

Figure 7. pH Control proposed by Mitsubishi

Removal of CP during Plant Shutdown (Reduction of CP Inventory)

During the plant shutdown, after the plant off-line, Ni and radioactive Co concentrations increase drastically because of the changes in the coolant temperature and chemical condition. Consequently, the ionic CP (Ni, ^{58}Co) levels become more than 10^3 times higher than the observed ones at steady-state power operation. Therefore, the removal of CP by purification of the coolant during the shutdown period is considered as a effective measure to reduce the radiation sources.
A chemical condition which enables the crud to dissolve more effectively was investigated in order to promote the crud inventory reduction. As a result of this investigation, "Low DH_2 control" was developed[4].
Figure 8 displays our understanding of the crud characteristics and its dissolution reactions in PWR plant during the plant shutdown.

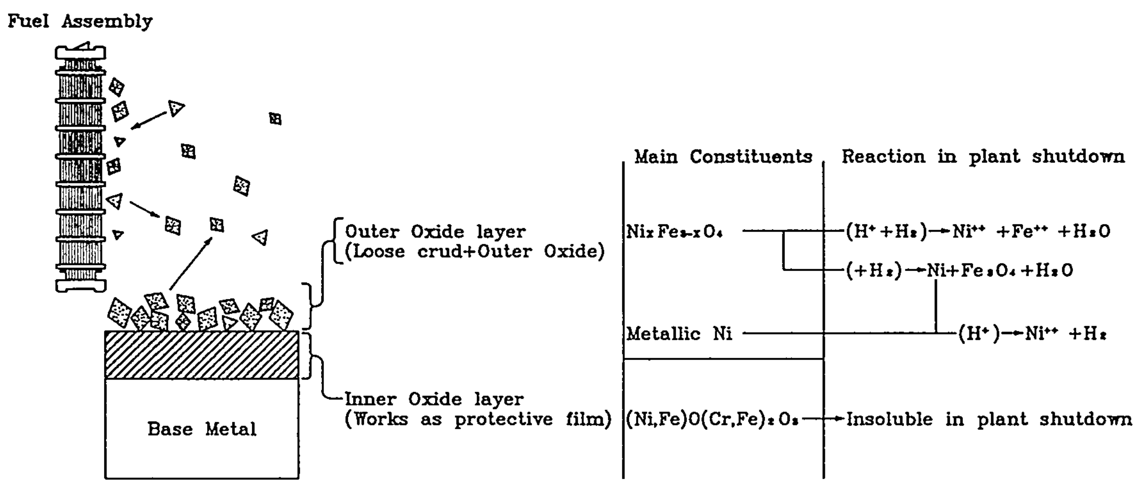

Figure 8. General feature of crud formed in PWR and its dissolution during the plant shutdown process

Using an autoclave test apparatus, Tawaki et al. investigated the dissolution properties of $Ni_{0.8}Fe_{2.2}O_4$ and the metallic Ni under chemical condition which simulates the plant shutdown.
Figure 9 shows the test result. From these data it was found that the trend of Ni dissolution from metallic Ni is very similar to that observed at the actual plants. It was verified that because of the Ni-metal dissolution the concentration of Ni ions increases under the lower DH_2 condition during the simulated plant shutdown. On the other hand, Fe dissolution from $Ni_{0.8}Fe_{2.2}O_4$ exhibits similar trend as the actual plant data. Consequently, it was confirmed that the main source of dissolved Ni during the shutdown condition is metallic Ni, but not nickel ferrite.
Metallic Ni dissolves by reaction of oxidation. Thus, Ni dissolution could be promoted by changing the redox potential to the higher level. To establish such condition, the "Low DH_2 control" (keeping DH_2 concentration about 0.5 cc-STP/kg-H_2O) was developed.
Usual degassing operation takes too long time to reach such a low DH_2 concentration level. Therefore, a hydrogen peroxide (H_2O_2) is injected to the reactor coolant to promote hydrogen degassing.
Figure 10 shows the concentration pattern of ^{58}Co, Ni and Fe measured during "Low DH_2 control". Predicted concentrations correspond to the concentration levels observed under the usual shutdown condition. Data analysis shows that this new control leads to 1.7 times higher Ni dissolution rate than the conventional shutdown chemistry.
The "Low DH_2 control" has recently been applied to many Japanese PWR plants as an effective crud removal operation promoting the crud dissolution during the outage shutdown.

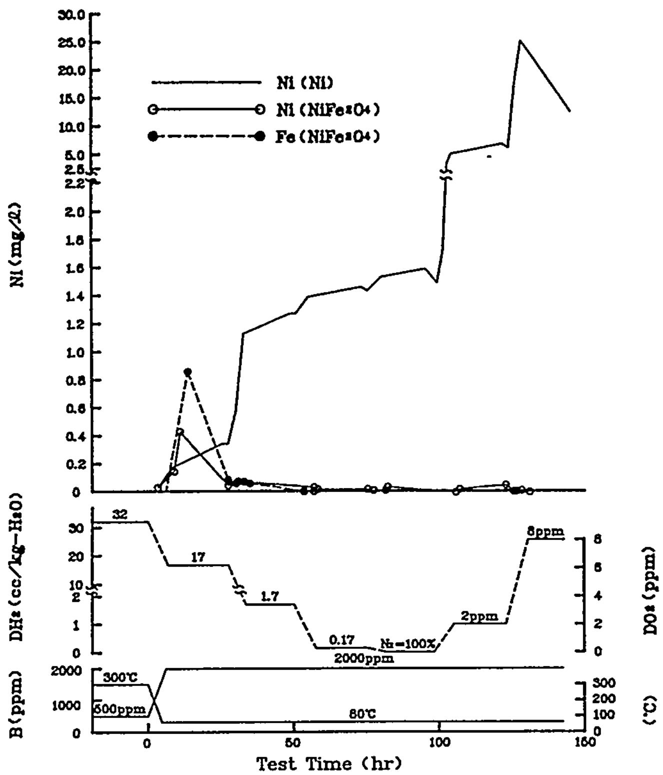

Figure 9. Dissolution of Ni and $Ni_{0.8}Fe_{2.2}O_4$ in boric acid solution simulating plant shutdown

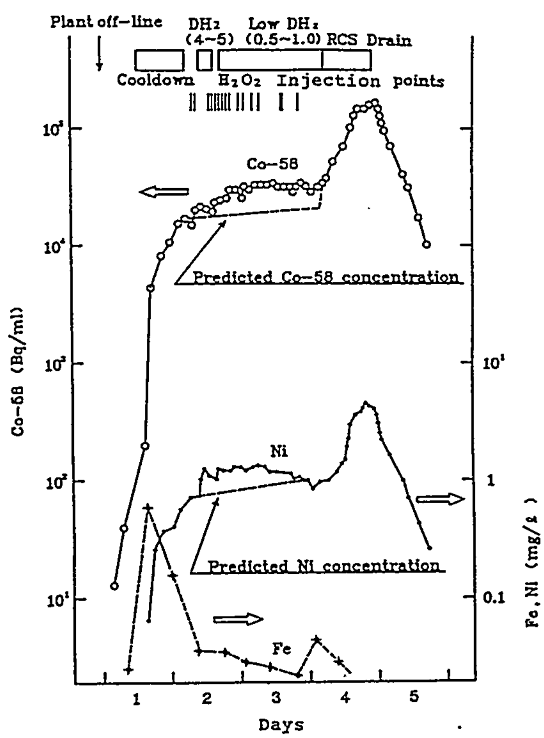

Figure 10. Co-58 and Ni concentration changes during the plant shutdown with low dissolved hydrogen control

Other Improvements

Now, under investigation are several new methods for radiation sources reduction as follows.

- Zn injection
- Other additives (NH_3 etc.)

On the other hand, we have developed the advanced code, called "ACE" for evaluation of crud behavior[5]. This code is capable to model CP transport and has the following range of application:

CP:	^{58}Co, ^{60}Co, Ni and Co
Layer:	3 (Loose crud, Outer Oxide and Inner Oxide)
Region:	14 (Figure 11)

1 – Hot leg
2 – SG channel head (Hot)
3 – SG tubes (Hot Side)
4 – SG tubes (Cold Side)
5 – SG channel head (Cold)
6 – Cold leg
7 – Fuel Assemblies (Lower spans)
8 – Fuel Assemblies (Middle spans)
9 – Fuel Assemblies (Upper spans)
10 – Fuel grids & stainless steel inventory
11 – Non–irradiated area
12 – RTD (Hot)
13 – RTD (Cold)
14 – RTD (Return)

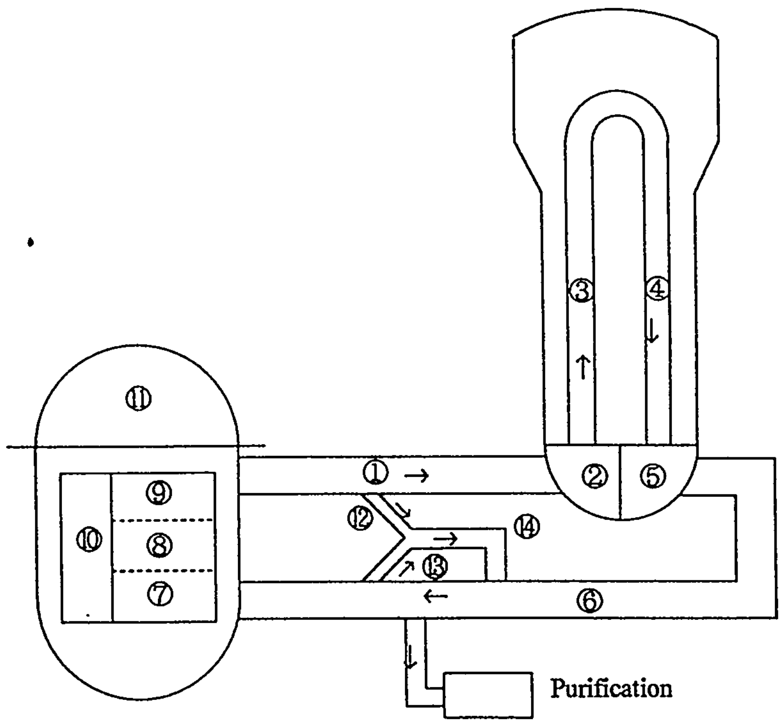

Figure 11. ACE code regions of PWR primary system

Radiation Exposure at Japanese PWRs

As mentioned above, primary water chemistry has been gradually improved to promote the radiation reduction. As a result of these efforts, radiation sources have been considerably reduced.

Figure 12 shows the radiation exposure at the first refuelling outage in Japanese PWRs. From this trend, the effectiveness of plant design and the above chemical control improvements on radiation exposure reduction were verified. Figure 13 displays the trend of SG channel head dose rates and the effectiveness of the improvements for two plants[6].

It is believed that future primary water chemistry improvements would further decrease the radiation sources and associated with them radiation exposure rates.

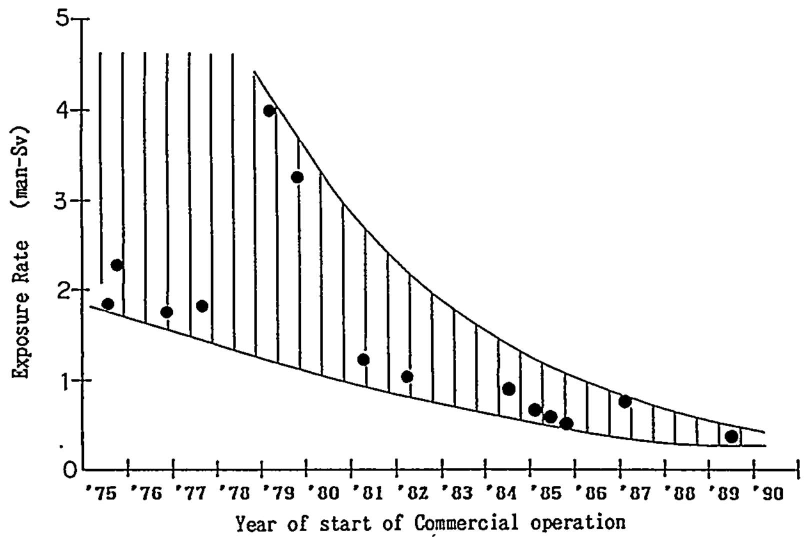

Figure 12. Radiation Exposure during 1st Refuelling Outage

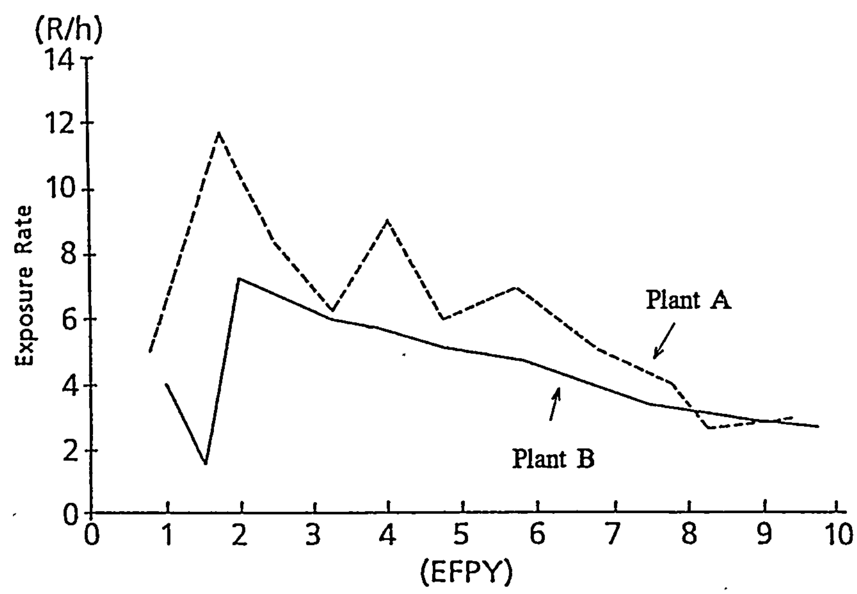

Figure 13. SG Channel Head Exposure Rate

CONCLUSIONS

The following improvements of primary water chemistry in respect to the radiation sources reduction in Japanese PWRs have been done.

- Improvement of the Chemical Control during HFT (Suppression of CP Generation)
- Improvement of pH Control during Power Operation (Suppression of CP Activation)
- Removal of CP during Plant Shutdown (Reduction of CP Inventory)

The effectiveness of above improvements was confirmed by the exposure trends.
Now, under investigation are several new methods for radiation sources reduction, such as Zn and other additives injection.

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Author Biography

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UPDATE TO MILLSTONE 3 ELEVATED pH TEST

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INTRODUCTION AND BACKGROUND

In view of the potential radiological benefits of elevated coolant pH operation, Northeast Utilities (NU), in support of an EPRI-Westinghouse program, agreed to operate the Millstone 3 plant at the start of its second fuel cycle as a demonstration of the effect of elevated coolant pH on out-of-core radiation fields. Operating with an elevated pH is defined as operating with an average lithium concentration of 3.35 ppm, until reaching an end of cycle pH of 7.2 or 7.4. The plant operated during its second and third cycles with an elevated coolant pH. The end of cycle pH during the second cycle was 7.4, and 7.2 during the third cycle. (During the first cycle, operation was with a coordinated pH of 7.0.)

Evaluation of the dose rate trends in Millstone 3 after two cycles of elevated coolant pH operation concluded that an elevated coolant pH resulted in a 15 percent lower component dose rate compared to other plants that operated with coordinated pH 6.9. However, due to a possible increase in fuel clad corrosion, operation during cycle 4 was restricted to pH 6.9 coordinated chemistry, with the exception of the last two months during which the pH increased to 7.35. At the end of cycle 4 (EOC4), there was a greater increase in component and crud trap dose rates than expected. This paper reviews the radiological trends in the plant and discusses the potential causes for the increase in the dose rates at EOC4.

PRESENTATION OF RADIOLOGICAL DATA

Several types of radiological data were taken during the Millstone 3 test. These included: coolant radiocobalt activity data; dose rates at various Electric Power Research Institute-Standard Radiation Monitoring Program (EPRI-SRMP) locations; and nuclide concentrations from two EPRI-SRMP locations representative of ex-core components.

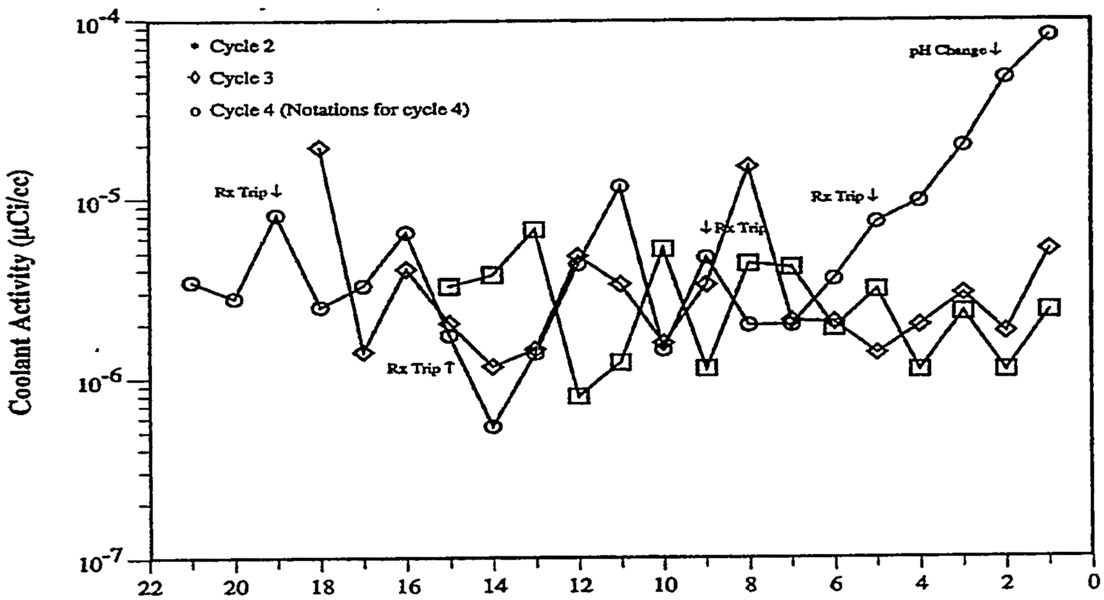
Coolant Radiocobalt Activity

The Millstone 3 primary coolant radiocobalt activities for cycles 2, 3, and 4 of the soluble and insoluble components were averaged on a monthly basis (similar data are not available for cycle 1). Figure 1 shows the trends for the insolubles plotted in months prior to the EOC to discern any effects of the final pH on the activity trends. Observations regarding the trends include:

- A considerable overlap in the insolubles among the three cycles, except for the last five months of cycle 4 operation.

- A reactor trip in the fifth month before the EOC4 (as noted in the figure). After the trip, the average activity in the insolubles started to increase. Three other reactor trips also occurred in previous months. During the month of the trip, the insoluble activity was generally higher compared to prior or subsequent months, however, an increasing trend did not occur after these trips, as it did following the trip in the fifth month before the EOC.

Monthly Average Co-60 Insoluble Coolant Activity
Millstone Unit 3



Monthly Average Co-58 Insoluble Coolant Activity
Millstone Unit 3

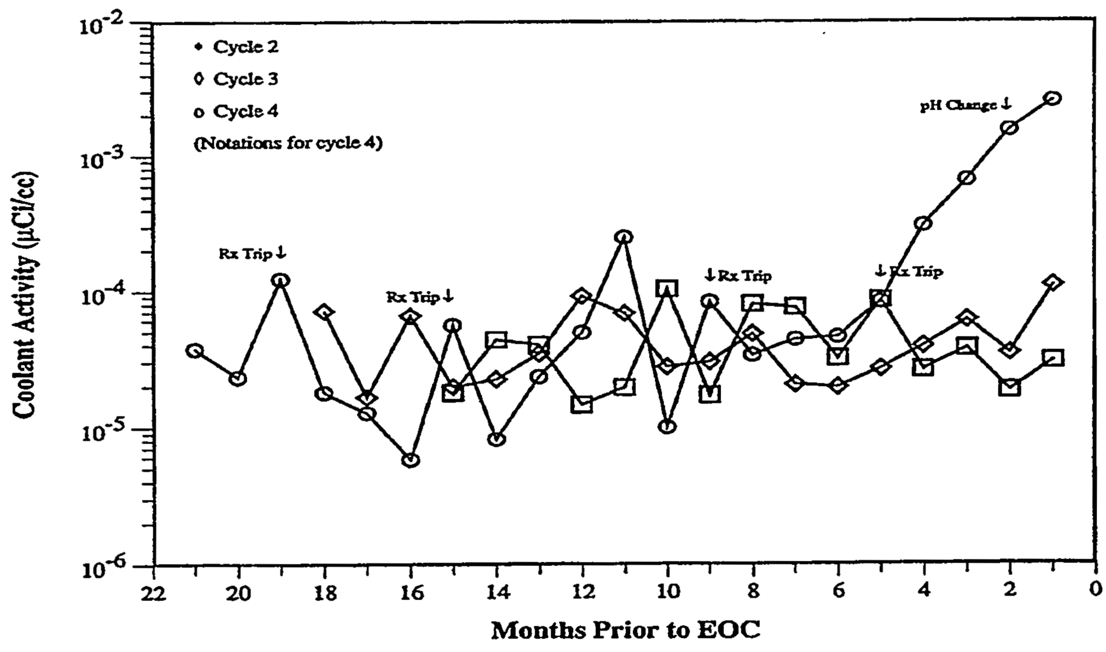


Figure 1. Insoluble Coolant Radiocobalt Activity Trends in Millstone 3 for Cycles 2, 3 and 4

- After the change in pH in the second month before the EOC, the average activity in the insolubles increased again. The absolute activity increase was about a factor of 3 to 5 times greater compared to that observed during the months after the reactor trip.

Measurable total suspended solids were found in the coolant (values of 10 to 25 ppb) during the last two months when the activity in the insolubles was increasing. Prior to this time, the suspended solids were always less than 10 ppb (except for a few times during or soon after a plant shutdown). However, the insoluble activity changes were not entirely consistent with the suspended solids values, thus suggesting that the overall increase in the insoluble activity was at least partially due to an increase in the specific activity of the insolubles, rather than solely an increase in the suspended solids concentration.

No similar increasing trend for the soluble radiocobalts was noted for the last five months before the EOC, thus suggesting that the reactor trip may have loosened insoluble crud from the core or caused it to be more mobile during the ensuing months.

Dose Rates

Figure 2 shows the dose rates of three EPRI-SRMP locations monitored (steam generator channel head, exterior to the steam generator shell, and on the crossover piping) for the Millstone 3, Ringhals 3 and 4, and the comparison group of plants. Ringhals 3 and 4 also operated with an elevated coolant pH. Observations regarding the trends are:

- A considerable increase in the dose rates in Millstone 3, by an average factor of 1.68, from EOC3 to EOC4.

- A similarity between the trends for Millstone 3 and Ringhals 3 and 4 prior to EOC4: an increase up to 1-2 EFPY and then a leveling off. This is in contrast to the trends in the comparison group of plants (which operated with a pH 6.9 coolant chemistry). In these plants, the dose rate trend did not level off until about 3-4 EFPY. In addition, the absolute value of the dose rate in the comparison plants are generally greater than in the other three plants. These trends suggest the radiological benefit of the elevated pH operation.

In addition to the above increase in the component dose rates, increases in the dose rates at other areas shown in Table 1 were noted from EOC3 to EOC4. It was estimated that the increases in dose rates shown in the table contributed to an extra 175 man-rem to the total dose during the cycle 4 outage.

Plant Health Physics personnel also noted an increase in dose rates at certain areas routinely monitored after the reactor trip that occurred five months before the EOC4 (March 31, 1993). Table 2 lists the changes in the average general area dose rates at several locations monitored during the five-month period. The locations monitored are representative of the components of sections of the chemical and volume control system (CVCS) letdown line. The locations near the letdown heat exchanger could be typical of component dose rates, whereas the dose rates near the letdown line could be more typical of component and crud trap dose rates. Note that the dose rates several days after the reactor trip did not change. However, after about one to two weeks, the dose rates increased about a factor of 1.5 to 2. They then continued to increase until they were about another factor of two above the initial increase until near the EOC4. The absolute value of these general area dose rates are subject to some uncertainty due to the nature of taking general area dose rates compared to contact dose rates.

In addition to the above changes, a large increase in dose rates was noted at the EOC4 in the pressurizer cubicle during the shutdown process. During this same period, a particulate crud burst of 1.4 ppm occurred, and the dose rates near the top of the ladder in the pressurizer cubicle increased from 50 mR/hr to 800 mR/hr. The increase was believed to be due primarily to an increase in the dose rates from hot spots in the pressurizer spray line and valves in the pressurizer system.

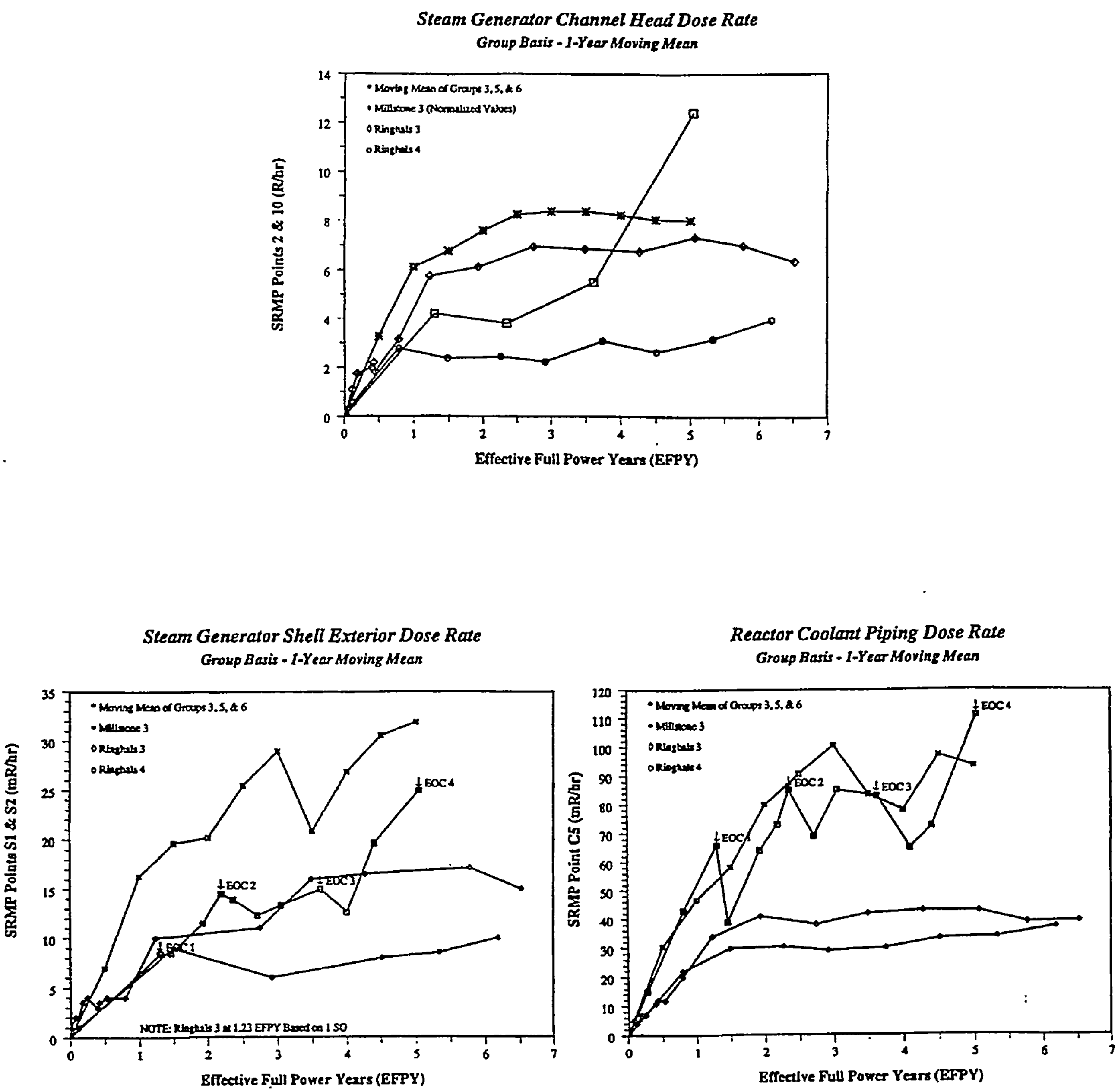


Figure 2. Dose Trends in Millstone 3 and Other Comparable Plants

Table 1 Increases in Dose Rates (mR/hr unless indicated) from Before EOC4 to EOC4			
Component	**1986-1993 Values**	**EOC4 Values**	**Maximum Factor Increase**
Pressurizer	30 - 500	100 - 8,000	16
Head Vent Valves	10 - 50	200 - 10,000	200
Refueling Cavity Water	1 - 5	24 - 90	18
Reactor Head/Pit Seal	10 - 30	30 - 300	10
Loop Hot Spots	300 - 500	10 - 200 R/hr	40
Regenerative HX	2 - 3 R/hr	4 - 5 R/hr	1.8
Tri-Nuclear Filters	300 R/hr	1,800 R/hr	6
SG Secondary Side	5 - 7 R/hr	15 - 17 R/hr	2.5

Inspection of average piping and steam generator shell component dose rate trends shown in Figure 3 during the EOC4 shutdown process shows essentially no change during the crud burst, indicating minimal crud/activity deposition on component surfaces. In addition, the amount of activity released in the crud burst is a small fraction of that deposited in the components. Thus, while dose rate increases at some crud traps happened gradually after the reactor trip on March 31, 1993 to the EOC4, increases in the pressurizer cubicle occurred quickly after the crud burst at the EOC4.

Nuclide Surface Concentrations

The nuclide surface concentration on the steam generator tubing and crossover piping was measured in the plant at the end of each cycle using in situ gamma ray spectroscopy techniques. Using these data, the total ex-core activity was calculated and Figure 4 gives the results for the two radiocobalts. Due to the relatively short half-life of cobalt-58, one would expect the activity value to reach equilibrium after a cycle of operation. Note that the amounts of cobalt-58 deposited for each cycle appears to vary depending on the pH (or ending pH) of the coolant, and that there is a minimum at pH 7.4 and a maximum for pH 6.9. For cobalt-60, the rate of activity buildup appears to decrease with time of operation.

PRELIMINARY EVALUATION OF DATA TRENDS

Based on CORA code calculations, an overall increase in dose rates of about a factor of 1.40 would have been expected with operation at pH 6.9, and taking into account a benefit of about a 5 percent reduction in dose rate due to conversion of the fuel to Zircaloy grids. (The CORA-calculated change due to elevated pH operation in lieu of pH 6.9 operation for the fourth cycle was only a few percent less.) The fact that the dose rates increased by about a factor of 1.7 indicates that other factors not accounted for by CORA were in effect. One such factor is changes due to crud bursts, since CORA does not take into account dose rate changes due to these events.

The large increase in cobalt-58 deposited activity at EOC4 compared to a leveling of cobalt-60 activity suggests that the transport of nickel is more influenced by pH operational changes as compared to that of cobalt.

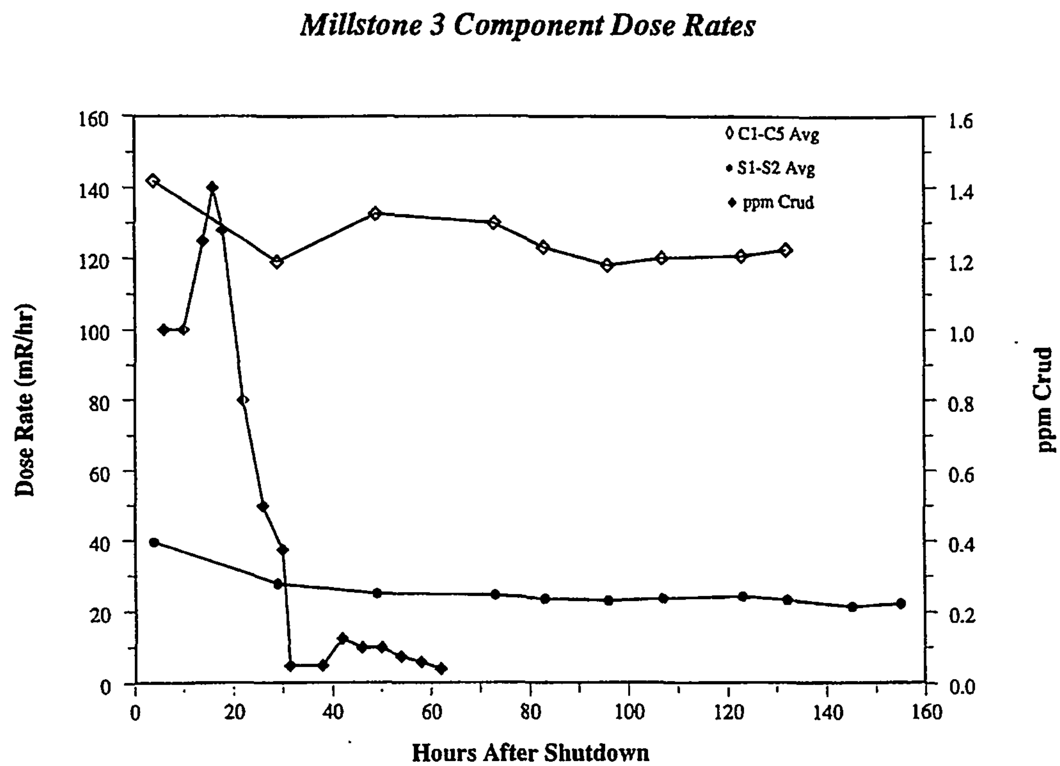


Figure 3. Component Dose Rate Changes During EOC4 Shutdown

As shown in Figure 3, the increase in dose rates at the pressurizer cubicle noted during the crud burst at the EOC4 shutdown did not cause an increase in the component dose rates during this same time period. This suggests that the increase from cycle 3 to 4 for the components usually monitored was due to a long-term change in the plant activity levels and not the short-term change noted at the pressurizer cubicle. The cause of the crud burst during the shutdown period is not known. Based on an evaluation of coolant chemistry and temperature changes during the shutdown process, it was concluded that the cause was not due to coolant chemistry evolutions during the shutdown.

The behavior of the trends in Table 2 also indicate an increase in crud trap and component dose rates during a longer time period than during the shutdown period. As noted in Figure 1, the last reactor trip could have initiated a continuing release of crud from the fuel deposits. The change in the coolant pH during the last two months could have caused a continuation of this process. This fuel crud would have a higher specific activity than that of the normal circulating coolant crud, thus contributing to the overall increase in the insoluble activity. The increase in activity in turn contributed to the increase in the crud trap and component dose rates from the EOC3 to the EOC4.

Based on evaluation of the radiation data to date, the following preliminary conclusions regarding the causes of the unexpected dose rate increase at the EOC4 have been made:

- An increase in the activity and/or concentration level of the coolant particulates during the last five months of operation appeared to be related to a reactor trip that occurred five months before the EOC4 and the pH increase that occurred two months before EOC4. After this increase, the dose rates at certain components and crud traps increased by several factors.

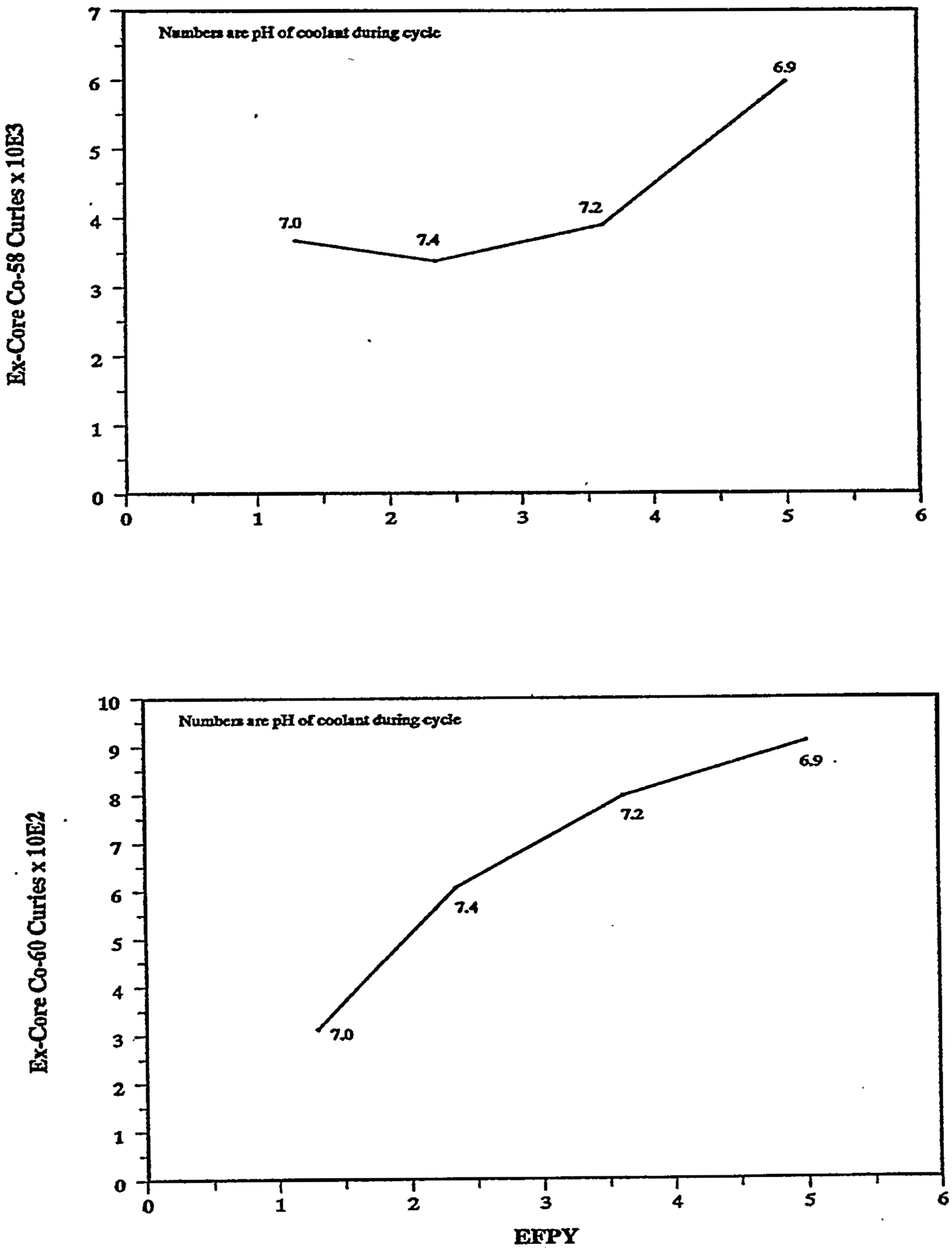


Figure 4. Total Ex-Core Activity

Table 2 Changes in Certain Average General Area Dose Rates (mR/hr) from Mid-March, 1993 to EOC4		
	Letdown Heat Exchanger Area	
Date	Second Level	4-foot Lower Level
3/22	55	54
3/29	59	46
4/5	44	30
4/9	140	81
4/12	130	101
4/19	183	85
4/26	193	119
6/24	213	123
7/5	153	75
7/19	288	182
8/1	--	221
Date	Near Letdown Line in Auxiliary Building	
3/23	17	
4/6	20	
4/13	26	
5/4	27	
5/11	33	
6/15	73	
6/22	61	

- A greater proportion of the dose rate increase was due to an increase in cobalt-58 activity rather than increase in cobalt-60 activity.

- The large particulate crud burst that occurred several hours after shutdown did not contribute to the EOC 3 to EOC 4 component dose rate increase. However, it could have contributed to increases at certain crud traps. The cause of the crud burst is not known but was not due to coolant chemistry evolutions during the shutdown.

Additional data evaluation is continuing to further define the causes of the increase in dose rates.

Author Biography

Carl A. Bergmann is a Principal Engineer in the Radiation and Engineering Analyses Group in the Nuclear Technology Division of Westinghouse Electric Corporation. He has over thirty years of experience in the nuclear field and has been the lead engineer for the research, development and application of dose-reduction techniques to PWR nuclear plants for fourteen years. Dose-reduction techniques include the application of coolant additives such as zinc and enhanced amounts of lithium to the primary coolant. He also led a study to evaluate sources of cobalt in Westinghouse-designed plants. Mr. Bergmann holds a B.S. Degree in Chemical Engineering and a Masters in Business Administration.

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PAPER 7A-2
DISCUSSION

Wood: So your recommendation, Carl, will be that plants then go back to pH 6.9?

Bergmann: We are still trying to sort out whether or not the pH 6.9 operation caused the whole problem, but you can certainly speculate that pH 6.9 caused more crud to be generated and deposited on the core, thus being available for further activation. On this basis, it looks like you shouldn't go back to pH 6.9 after operating at an elevated or modified pH.

Wood: It seems pretty obvious to me that you've got a lot more crud.due to the going back to pH 6.9 and that's being released from the core for whatever reason. The key point is that you are obviously forming a lot more crud in the core, which is a bad thing.

Bergmann: Yes.

Wood: Perhaps the co-chair, Krister Egnér, would like to comment.

Egnér: No.

Riess: Could you explain a little about the fuel cladding and the reason why you went to pH 6.9?

Bergmann: There were a number of people concerned from the fuel viewpoint, that when you operated with an elevated or modified pH you have more exposure to lithium. If you can plot the lithium days, the days of certain lithium concentration vs. the time period during the cycle, you have more lithium days operating at a modified or elevated pH than you would have if you operated at a coordinated pH of 6.9. Since lithium is the culprit in terms of fuel cladding corrosion, it is desirable to minimize the lithium days.

Riess: Yesterday I think I heard someone say that there was no evidence of additional fuel cladding corrosion.

Bergmann: That analysis was done after the decision was made to operate at pH 6.9. For this cycle, the plant is planning to operate with modified chemistry. I guess one reason is because most of the fuel is Zirlo fuel, which has less susceptibility to lithium corrosion.

Wood: We did a comparison between Millstone 3 and North Anna, which had operated on pH 6.9. The Millstone 3 data suggested that zircaloy corrosion could be about 14% higher. That whole cycle was the reason for the concern. Then we went back and looked at more detail. Just looking at the Millstone 3 data, we actually thought that the elevated lithium had reduced zircaloy corrosion, comparing cycle 1 with the fuel that went into cycle 2 onwards. So at the moment, we don't believe that there is an adverse lithium effect. We are looking now at the cycle 4 data, which is going back to 6.9, and that will give us a direct comparison. So we can directly compare the pH 6.9 data with or without elevated lithium. Hopefully, in the next few months we will have a better picture on the zircaloy data.

Bergmann: Millstone cycles are 18 months. So in addition to having the higher lithium levels, the higher burnup was also a factor.

Wood: I should have made the point that the first cycle was only 12 months, and we had a problem comparing that with the later ones. The cycle 4, of course, was 18 months, the same as the elevated lithium cycle, so that should help our evaluation.

A USER FRIENDLY DATABASE FOR USE IN ALARA JOB DOSE ASSESSMENT[1]

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ABSTRACT

The pressurized water reactor (PWR) design chosen for adoption by Nuclear Electric plc was based on the Westinghouse Standard Nuclear Unit Power Plant (SNUPPS). This design was developed to meet the United Kingdom requirements and these improvements are embodied in the Sizewell B plant which will start commercial operation in 1994.

A user-friendly database was developed to assist the station in the dose and ALARP assessments of the work expected to be carried out during station operation and outage. The database stores the information in an easily accessible form and enables updating, editing, retrieval, and searches of the information.

The database contains job-related information such as job locations, number of workers required, job times, and the expected plant doserates. It also contains the means to flag job requirements such as requirements for temporary shielding, flushing, scaffolding, etc.

Typical uses of the database are envisaged to be in the prediction of occupational doses, the identification of high collective and individual dose jobs, use in ALARP assessments, setting of dose targets, monitoring of dose control performance, and others.

INTRODUCTION

The PWR design chosen for adoption by Nuclear Electric plc was based on the Westinghouse Standard Nuclear Unit Power Plant (SNUPPS). This design was developed to meet the United Kingdom requirements and these improvements are embodied in the Sizewell B plant which is expected to start commercial operation in 1994.

The Health Physics Department of the station is responsible for the radiological protection of all workers on the station. To assist the Health Physics Department in the dose and ALARP assessments of the work expected to be carried out during station operation and outage, a simple user-friendly database was developed. The database was called "The Task Dose Database."

Such a computer database is required to store the information in an easily accessible form in order to fulfill the following requirements:

- enable maintenance of the information; i.e, update/edit

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- retrieve information and carry out searches; i.e., on:
    - activities
    - collective/individual doses
    - parameters such as temporary shielding, scaffolding, etc.

- enable incorporation of operational data

The database contains job-related information such as job locations, number of workers required, job times, and the expected plant doserates. It also contains the means to flag job requirements such as requirements for temporary shielding, flashing, scaffolding, etc.

Typical uses of the database are envisaged to be:

- prediction of occupational doses,

- identification of high collective and individual dose jobs

- use in ALARP assessments

- setting of dose targets

- monitoring of dose control performance

- others

This paper describes the structure of the database, the sources of the initial data and the use of the database.

The Database Software and Hardware

The decision was taken from the beginning of the work to make the database easily transportable and to base its development on a commercially available database package.

The software used for the development of the database was chosen to be the DBASE III Plus software. The database is run on personal computers (IBM/IBM-compatible).

THE DATA SOURCE: THE ACTIVE-TASK DOSE ASSESSMENT DATA

A dose assessment of the tasks that are likely to be carried out within the station's Radiologically Controlled Area (RCA) during outage and normal power operation, the active-task dose assessment, was carried out in support of the station's Pre Operational Safety Case (POSR) and was included in the POSR.

This task dose assessment considers the locations, manpower requirements, times, and dose rates of the tasks in order to calculate the collective and individual doses associated with the tasks.

In presenting the information of the task dose assessment, various formats were found to be necessary. All formats are based on the same concept which gives the information on the details of the task, the location, the manpower, the dose rates, and the doses. The most typical example of presenting a task is shown in Table 1.

Table 1. Typical Dose Assessment of a Task/"Activity"

ISI of Flux Thimble Guide Tubes

Location: RB Flux Mapping Room - Seal Table
Test: VT (Visual Testing)

Frequency: 1/10 years

Step	Task Details	No. of Men	Time hr	Location	Dose Rate mSv/h	Indiv. Dose mSv	Collec. Dose mSv	Dose-Rate Point
1. Prepare Area	Set up inspection control area	1	0.5	Seal tab	0.08	0.04[2]	0.04	20
2. Equipment Preparation	a) Bring equipment in to RB	2	1.0	RB rooms	0.01	0.01	0.02	3456
	b) Set up equipment	1	1.0	Seal tab	0.08	0.08[1]	0.08	20
3. Inspection	Semi-remote inspection	1	16.0	Seal tab	0.08	1.28[1]	1.28	20
4. Equipment Removal	a) Equipment disassembly	1	1.0	Seal tab	0.08	0.08[1]	0.08	20
	b) Remove equipment	1	1.0	RB rooms	0.01	0.01	0.02	3456
5. Re-instate Area		1	0.5	Seal tab	0.08	0.04[2]	0.04	20

Individual Dose
1. Operator 1 = 1.44 mSv
2. Operator 2 = 0.08 mSv

task collective dose = 1.56 mSv
including 10% Health Physics dose = 1.72 mSv

x frequency 1/10

annual average coll. dose = 0.172 mSv

The station work was grouped according to its nature in nine groups called 'Work Functions' consistent with the USNRC grouping. These are:

1. In service Inspection (ISI)
2. Refuelling
3. Scheduled Maintenance - mechanical
4. Scheduled Maintenance - electrical
5. Scheduled Maintenance - In Service Testing (IST)
6. Waste Processing
7. Unscheduled Maintenance
8. Operations and Surveillance (Ops. & Surv.)
9. Health Physics (HP)

The tasks in the Work Functions are further classified into "Classes" and '"Categories." For example, the ISI work has the following Classes: ISI of Reactor Pressure Vessel (RPV), ISI of Class 1 components, ISI of Class 2/3 components; the tasks in the Class 1 components ISI are further subdivided in Categories such as the Steam Generator ISI, the Pressuriser ISI, etc.

In the POSR, detailed task dose assessment was carried out for the first six work functions. The dose assessment of the last three was based on operational experience. In particular:

- Unscheduled Maintenance: this was set equal to the scheduled maintenance dose,
- Operations & Surveillance: this was expressed as a percentage of the total collective dose (12%),
- Health Physics: this was added as a dose allowance on the other work (10%).

Out of these three, the Health Physics dose is added as a dose allowance on each work activity and the dose from the other two is added to the total collective dose.

The database incorporates the data from the detailed dose assessment of the six work functions. No work activities have been included in the database for Unscheduled Maintenance and Operations & Surveillance. These can be added by the user (the station) at a later time as the work requirements become better defined.

THE STRUCTURE OF THE DATABASE

The Task Dose Database was set up initially to contain the POSR information and to enable the updating of this information as station operating experience develops. It was recognised that the data should be stored in a manner which minimizes the possibility of data inconsistencies during its updating and enables the inclusion of features to search and retrieve the data.

The "database" as such is a set of interrelated databases which are manipulated by a central program which also provides the interface with the user. In order to define the structure of the databases to be used, the following three steps were followed: (a) definition of the data to be included, (b) normalisation of the data-fields, and (c) definition of the databases.

The form to be used as the basis of the database is shown in Table 1. This is called an "**ACTIVITY.**"

The information included in an Activity is (see Fig.1):

(a) Activity Identifier: an activity is identified by a Unique Number and its title.

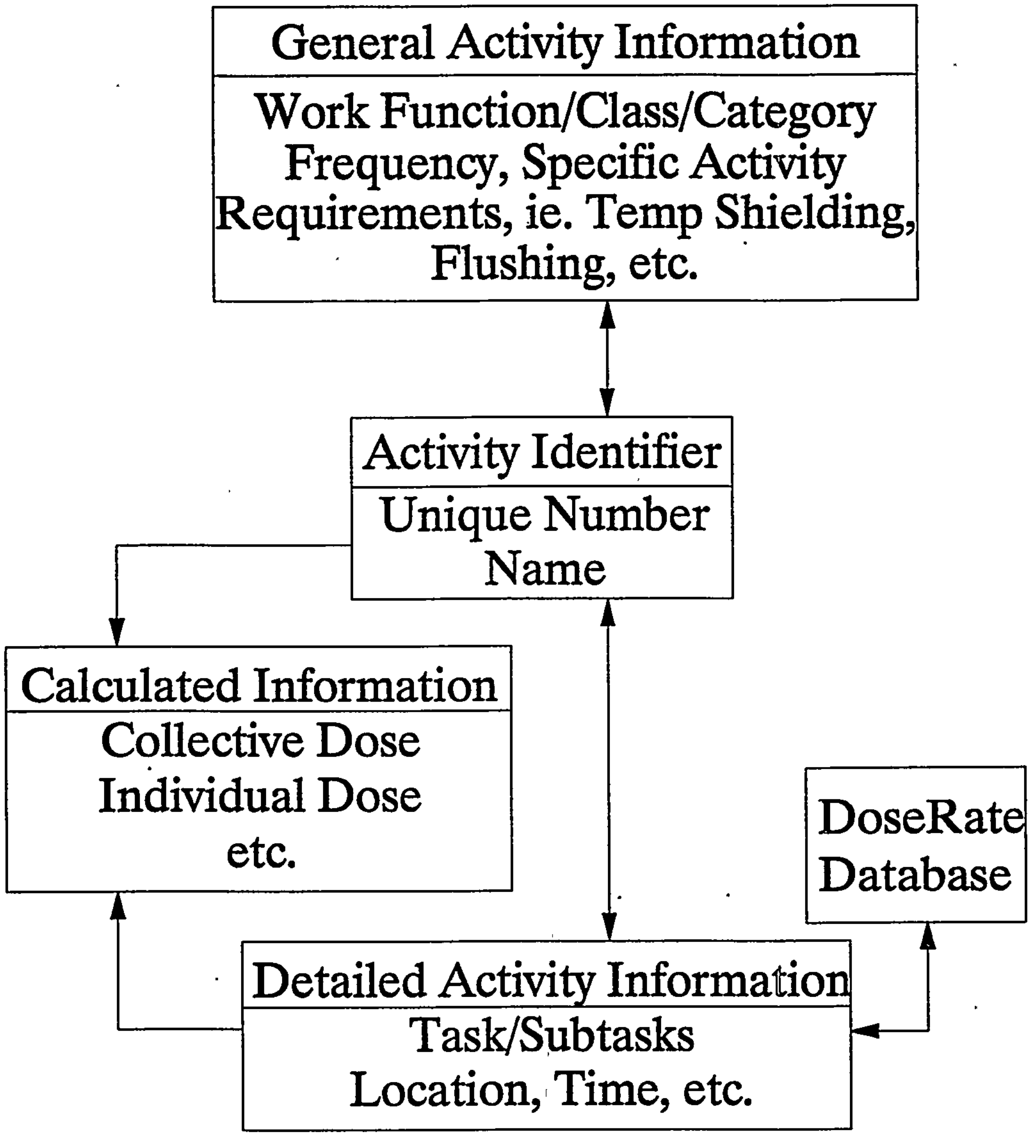

Fig 1: Information to be included in the Database

(b) General Information: this gives general information about the activity such as the Work-Function to which the Activity is grouped under, the frequency of operation, etc.

(c) Detailed Information: this details the work being carried out, the number of men, times, locations, etc.

(d) Dose rate Information: this is information about the doserates being used in the calculation of the doses.

(e) Calculated Information: this is information which is calculated from other defined data such as the individual and collective doses, etc.

In order to decide on the structure for the database, a procedure called Normalisation was used. This procedure groups attributes (or fields) into well structured relations and results in a database (or set of databases) which contains the minimum amount of data redundancy. The user is, therefore, allowed to insert, delete, and modify the data without errors or inconsistencies resulting from these operations.

The resulting database structure includes six separate databases as follows:

ACTIVITIES	(Unique_No, Activity, WorkF_Code, Class_Code, Categ_Code, Frequency, Scaff_Reqd, TShld_Reqd, Flush_Reqd, Other_1, Other_2, Comments)
TASKS	(Unique_No, Task_No, Task_Name, Subtask_ID, Subtask, Manpower, Time, Location, Worker_ID, Drt_Ref, Rel_Activ)
WORKFUNCTIONS	(WorkF_Code, Full_WorkF)
CLASSES	(Class_Code, Parent_WFn, Full_Class)
CATEGORIES	(Categ_Code, Parent_Class, Full_Categ)
DOSERATES	(Drt_Ref, Drt_Code, Drt_Location, Dose_Rate, Dep1, Dep2, Dep3, Dep4)

The Underlined fields represent the key field (primary search field) of the database that it is contained in. The Dotted Underlined fields are the fields which are the Key field in another of the databases.

Two databases (ACTIVITIES, TASKS) contain the information relevant to the Activities and are linked through the Activity identifier (an Activity unique number).

Three databases (WORKFUNCTION, CLASSES, CATEGORIES) contain the information on the grouping of the Activities into Work Functions and classifications within the Work Functions. These three databases are linked with the general activity database through abbreviated codes.

The sixth database (DOSERATES) contains the information on the doserates and is linked to the database containing the detailed activity data through the doserate reference.

A seventh database (a Configuration database) is also used to hold other general data that is required for various purposes such as the percentage contribution of the Health Physics work.

FACILITIES PROVIDED TO THE USER BY THE DATABASE

The purpose of setting up the task dose database is to enable the user to manipulate the data and extract information with ease.

The software package used to set up the database provides some basic facilities for updating the database and extracting information. It also provides a programming language which enables the development of further facilities for more complex data operations.

As part of the development of the database, a set of user options have been specified and developed as user tools. The approach adopted in the development of the options is to use a hierarchy of menus to guide the user to the required facility/option. The principle of the use of menus is illustrated in Figure 2. The user options are discussed below.

Data Operations

This group of options provides the user with the means to add, delete and modify the following data in the database files:

- Activities
- Tasks / Subtasks
- Work Functions / Classes / Categories.

View Information

This group of options provides the user with the means of extracting and presenting the raw data contained in the database files as follows:

- General information on: Work Functions / Classes / Categories
Primary data of a group or range of activities

- Detailed task/subtask information of a group or range of activities

Searches

This group of options provides the user with the means of retrieving specific information from the database based on criteria specified by the user as follows:

- Miscellaneous searches for activities requiring scaffolding, temporary shielding, etc.

- Individual Dose: search activities or subtasks for specified individual doses.

- Collective Dose: search activities or subtasks for specified collective doses.

Collective Dose Calculations

This group of options provides the user with the means of calculating the collective dose of groups or ranges of activities specified by the user. The database provides the means for storing two task frequencies. One frequency is used to store the overall frequency of the task and the second frequency is used to store the frequency of the task for the period of interest.

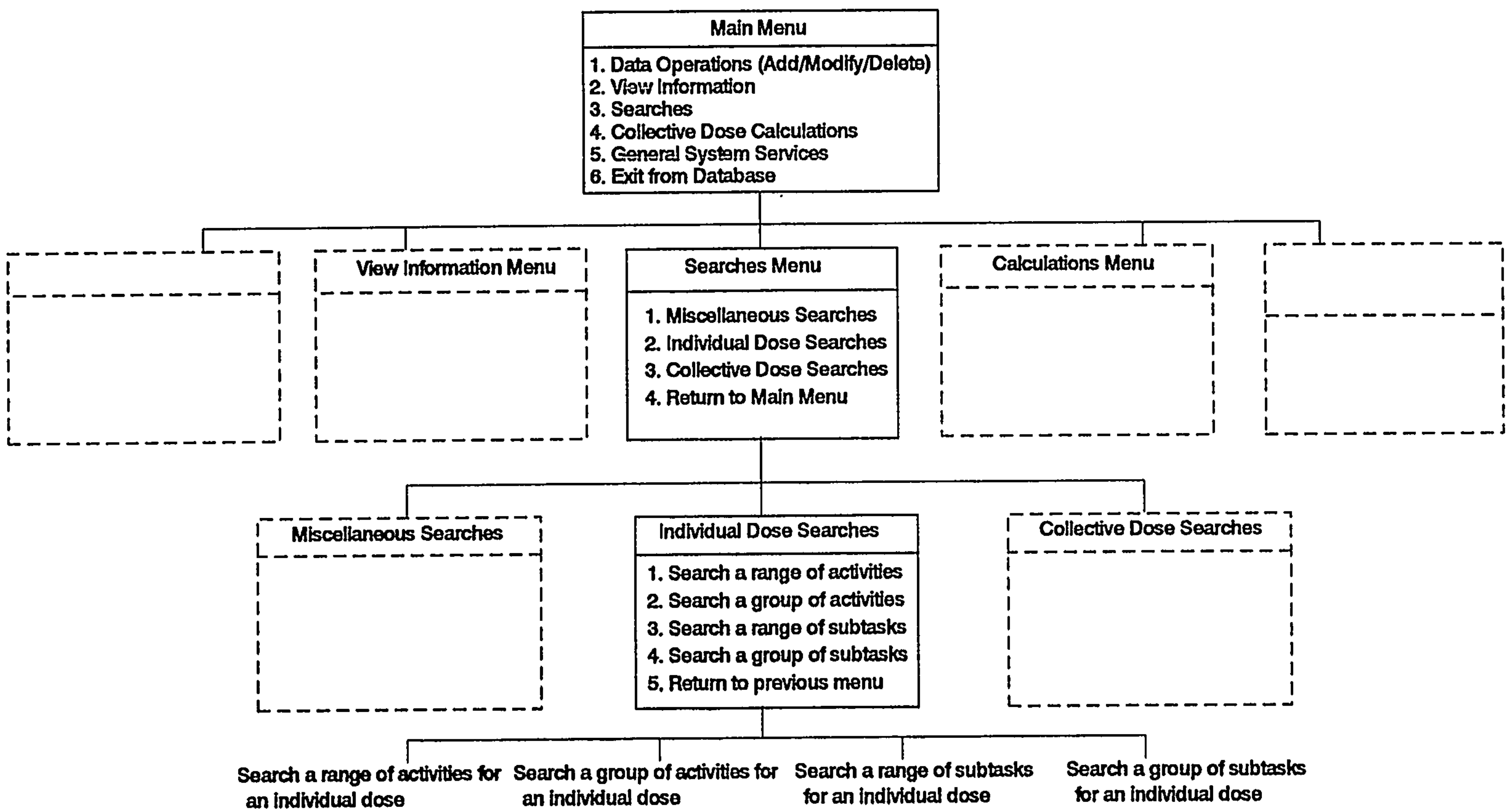

Fig 2: Illustration of Database Menu Structure

General System Services

This group of options provides the user with the means of changing the output device (print or screen), the Health Physics dose factor, to back-up and restore the database to and from a floppy disk, and to provide database diagnostic information (number of records and data/time of last changes).

DATABASE QUALITY ASSURANCE

The quality assurance of the database addressed the data input to the database and the software developed for the database.

The data input to the database originated from the Active Task Dose Assessment report. The contents of the data files were checked against the data presented in the report.

A test program was developed for the testing of the database software and the software was tested by extensive trial runs on a test database.

CONCLUSIONS

The requirement for the means to store, update, and retrieve information to assist in the performance of the station Health Physics duties was identified. The task dose database described in this paper was developed to fulfil this function and will be used at the Sizewell B Station.

ACKNOWLEDGMENTS

The author would like to thank the Sizewell B Health Physics department for their interest in the database and their cooperation in the preparation of this paper.

Author Biography

Anastasios M Zodiates is a Radiological Protection (Operator) Engineer with the PWR Project Group, Nuclear Electic plc., England. Dr. Zodiates is involved with the development of the radiological protection aspects of the design and licensing of the Sizewell B PWR. Before joining the Sizewell B team, he worked as a Reactor Physicist at the company's Heysham 2 Advanced Gas Cooled Reactor. He was born in the island of Aprodite, Cyprus, and attended university in England. He has a B.Sc. in Electrical Engineering from Southampton University, an M.Sc. in Nuclear Reactor Physics and Technology, and a Ph.D. in Physics from Birmingham University. He is a UK registered Charterd Physicist and Engineer.

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PAPER 7A-3
DISCUSSION

Lau: Will the QA manual fit on a diskette?

Zodiates: Yes, but you will not have the signature for the test.

PWR UPPER/LOWER INTERNALS SHIELD

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ABSTRACT

During refueling of a nuclear power plant, the reactor upper internals must be removed from the reactor vessel to permit transfer of the fuel. The upper internals are stored in the flooded reactor cavity. Refueling personnel working in containment at a number of nuclear stations typically receive radiation exposure from a portion of the highly contaminated upper internals package which extends above the normal water level of the refueling pool. This same issue exists with reactor lower internals withdrawn for inservice inspection activities. One solution to this problem is to provide adequate shielding of the unimmersed portion.

The use of lead sheets or blankets for shielding of the protruding components would be time consuming and require more effort for installation since the shielding mass would need to be transported to a support structure over the refueling pool. A preferable approach is to use the existing shielding mass of the refueling pool water. A method of shielding was devised which would use a vacuum pump to draw refueling pool water into an inverted canister suspended over the upper internals to provide shielding from the normally exposed components.

During the Spring 1993 refueling of Indian Point 2 (IP2), a prototype shield device was demonstrated. This shield consists of a cylindrical tank open at the bottom that is suspended over the refueling pool with I-beams. The lower lip of the tank is two feet below normal pool level. After installation, the air within the tank is evacuated, thereby drawing water up into the shield. This extends the height and width of the natural shielding provided by the existing pool water. This paper describes the design, development, testing and demonstration of the prototype device.

INTRODUCTION

The goal of this research and development program was to design, develop, test, and demonstrate a shielding system which would use the existing mass of the refueling pool water to provide shielding from the protruding components of the upper internals in order to reduce the radiation exposure of refueling personnel in containment. Figure 1 is a conceptual sketch of the device. The design had to meet the following objectives:

- Provide sufficient radiation shielding.
- Require minimum exposure for installation.
- Minimize critical path involvement for installation
- Allow similar use as a shield for the lower internals package during a future outage.

The project plan included the following major tasks: field inspection and measurement of the proposed installation site, design, fabrication, load testing, mock-up training and site installation/removal. The project was accomplished in a time frame of approximately ten months.

Figure 1

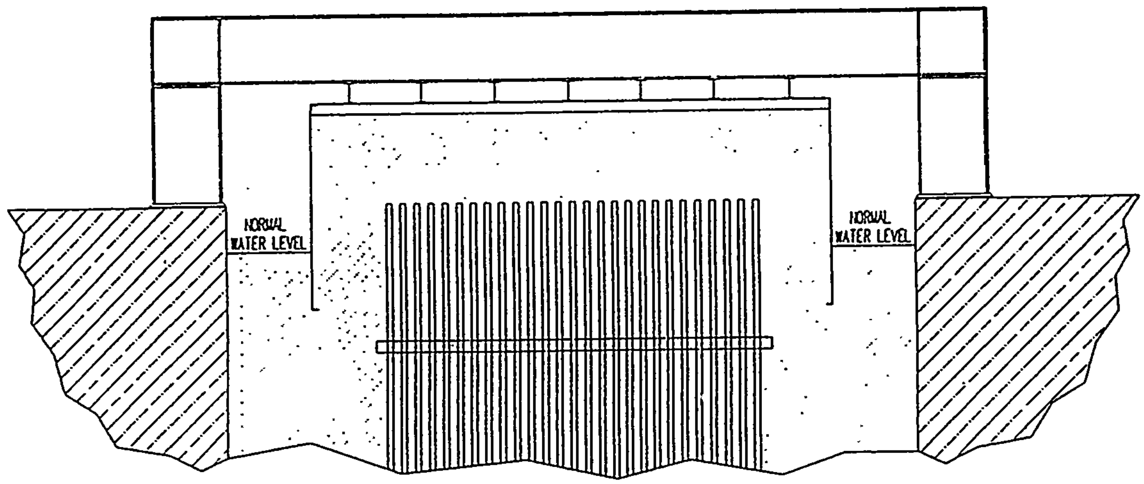
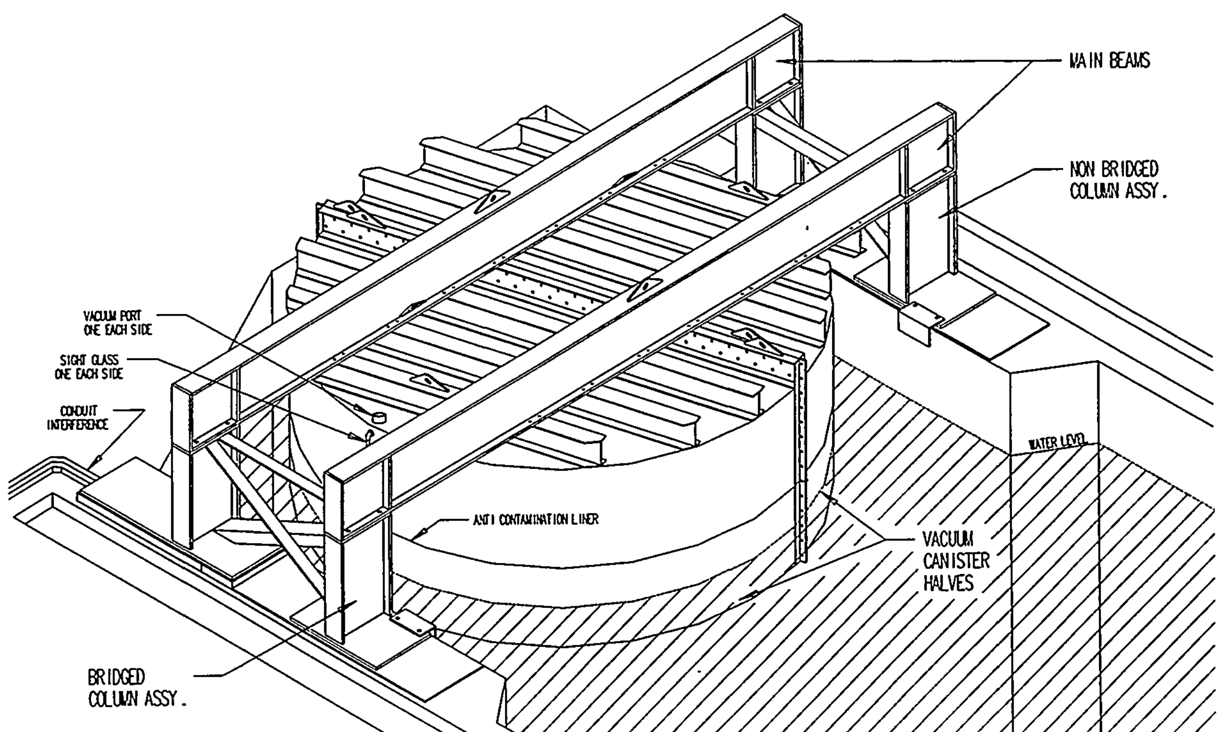


Figure 2



SYSTEM DESIGN

The upper internals vacuum radiation shield system shown in Figures 2 and 3 consists of a cylindrical tank with an open bottom that is suspended from outside the cavity by I-beams. The tank is positioned to provide 18" of immersion in the existing pool water. After installation most of the air trapped in the upper 54" of the tank is evacuated and the vacuum draws water from the refueling pool to fill the tank above the pool level.

The tank is fabricated in two halves which are bolted together and to its I-beams. The design eases transport through the containment building equipment hatch and allows for more compact storage between outages. Portions of the shield which will contact the refueling pool water are fabricated from stainless steel and all remaining components are carbon steel.

A stress analysis was performed to determine that the design is adequate for static loading. Static loads are approximately 20,300 lb empty and 72,150 lb when full.

A vacuum pump skid unit was designed and fabricated which consists of two liquid ring pumps, a vacuum accumulator tank, a seal water reservoir, control panel, and connections for float switches as well as vacuum inlet and vapor outlet. Vacuum-activated switches were originally installed but were later abandoned in favor of float switches which provide more accurate water level control.

CONCLUSIONS

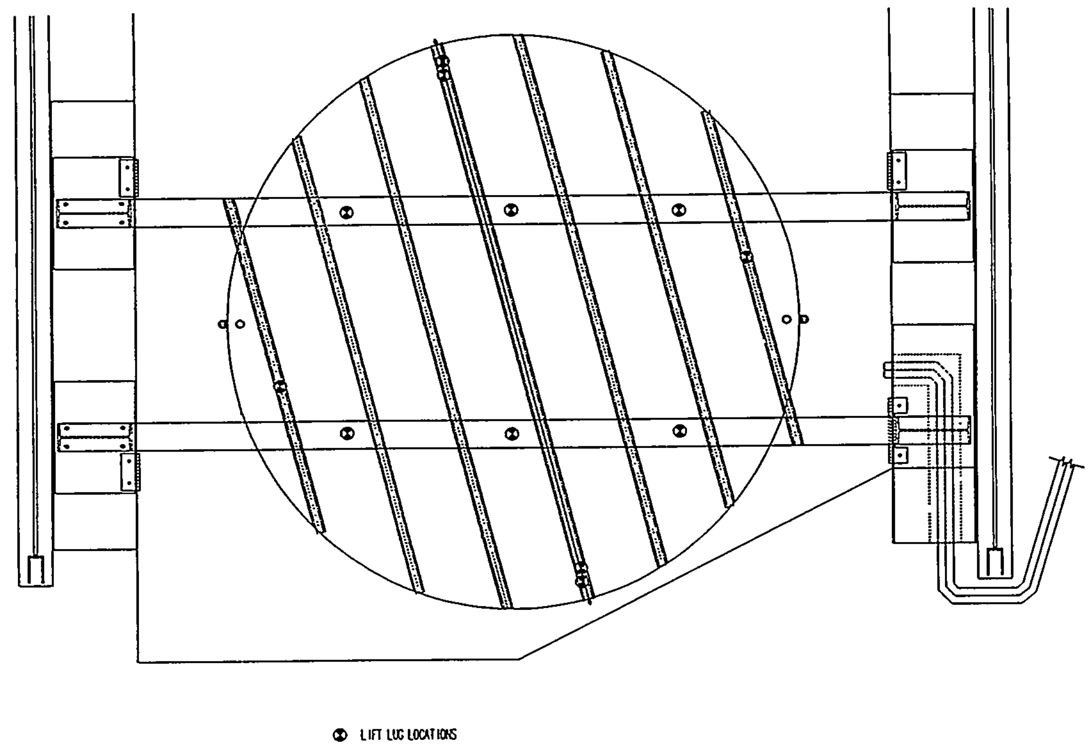
The development of the vacuum radiation shielding system resulted in significantly reduced dose rates to personnel. General area dose rates to refueling bridge personnel were reduced from 154 mR/hr to 25 mR/hr (see Figure 4). Fourteen person-rem of exposure were saved as compared to the 1991 refueling outage. At 10,000/person-rem, the net savings for Con Edison is approximately $140,000 per use.

The shield is extremely efficient in that it places the existing pool water in a geometric configuration that surrounds the source of the radiation. Any other shielding method would be less efficient and require more floor space adjacent to the pool. The smaller tank diameter is important to provide clearance for the fuel manipulator bridge to have access to refuel the unit. The use of pool water as the shielding medium minimizes the cost of water treatment and handling, and avoids concerns about inadvertent boron dilution for PWRs.

RECOMMENDATIONS

- Fabricate the shield with a smoother finish to facilitate decontamination
- Apply a strippable submersible coating to the shield prior to use.
- Improve venting to prevent possible splashing as air trapped under shield escapes.
- Consider use of alignment pins or rods between beams and pedestals.
- Install a float in the sight glass for easier reading.
- Simplify the vacuum system via use of vane type pumps (similar to those used in load test) mounted on bell with all switches and relays required for automatic level control mounted to the tank. The only connection needed would be a 115 VAC power cord.

Figure 3



Survey Location	Dose Rate (mR/hr) No Shield	Dose Rate (mR/hr) With Shield	Reduction Factor
2	160	20	8.0
3	100	22	4.5
4	120	28	4.3
7	180	28	6.4
8	140	30	4.7
9	100	22	4.5
11	240	20	12.0
12	180	22	8.2
13	220	28	7.9
14	100	30	3.3
Average	154	25	6.2

Figure 4. Indian Point 2 Hydro-vac® Radiation Survey Data

Author Biography

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REPORT ON THE PWR-RADIATION PROTECTION/ ALARA COMMITTEE

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ABSTRACT

In 1992, representatives from several utilities with operational Pressurized Water Reactors (PWR) formed the PWR-Radiation Protection/ALARA Committee. The mission of the Committee is to facilitate open communications between member utilities relative to radiation protection and ALARA issues such that cost effective dose reduction and radiation protection measures may be instituted. While industry deregulation appears inevitable and inter-utility competition is on the rise, Committee members are fully committed to sharing both positive and negative experiences for the benefit of the health and safety of the radiation worker. Committee meetings provide current operational experiences through members providing Plant status reports, and information relative to programmatic improvements through member presentations and topic specific workshops. The most recent Committee workshop was facilitated to provide members with defined experiences that provide cost effective ALARA performance.

INTRODUCTION

Although there are many forums for information exchange amid the nuclear utility industry, these forums typically take the form of written responses to specific questions, symposiums with a variety of topics on a defined subject or generic high level seminars. While all of these forums provide a specific benefit, none of them provided for a face-to-face verbal exchange dictated by the specific interests of the parties involved. The inception of such a forum was derived from a presentation by the U.S. Boiling Water Reactor (BWR) Owners' Group at the Radiation Exposure Management Seminar sponsored by Westinghouse in 1992. This presentation detailed the formation of a working group devoted to information exchange by utility representatives from BWRs on the subject of maintaining personnel exposure ALARA. Following this seminar, staff members of several utilities with operational PWRs gathered to form the PWR-Radiation Protection/ALARA Committee.

ADMINISTRATION OF THE PWR-RADIATION PROTECTION/ALARA COMMITTEE

Formation of the Committee

The need for a working group with the expressed mission of facilitating inter-utility information exchange relative to the promotion of maintaining the exposure of utility workers ALARA was validated by utility staff of PWR in the fall of 1922. Once the need was validated, an ad-hoc committee set out to identify the mission, principles, scope, and administration of the Committee.

The mission of the Committee was defined as:

"The PWR-Radiation Protection/ALARA Committee is committed to continual improvement in radiation protection standards and performance at our utilities".

The principles of the Committee were defined as:

a. The reduction of radiation dose to the workers of our plants, both utility employees and contractor personnel is a key measure of our success as a committee.

b. The free exchange of pertinent information, data, and lessons learned will be pursued in a constructive dialogue and atmosphere of mutual respect.

c. The Committee strives to provide a high quality product in the most cost-effective manner.

d. The Committee will develop and implement an integrated and consistent information exchange process by which issues are effectively identified, prioritized, analyzed, and communicated in a timely manner.

e. The Committee supports effective outage work planning, develops information exchange, and communicates lessons learned in support of short-term (outage-to-outage) dose reduction.

f. The Committee strives to identify, evaluate, and endorse recommendations for long-term source term reduction design change activities such as cobalt reduction, recognizing this as one of the most effective dose-management techniques.

g. The Committee continually evaluates industry products and services with exposure impacts and shares experiences: e.g., shutdown chemistry, operating chemistry, chemical decontamination, zinc injection, microfiltration, and mechanical decontamination of equipment to effectively manage our individual utility resources.

With the mission and principles identified, the Committee identified its scope as both, the maintenance of exposure ALARA and inherently sound radiation protection. The inclusion of the daily radiation protection aspect was identified to be necessary based on recognition that the most effective ALARA program available is predicated by thorough job planning.

To administer the Committee, an organization composed of chairperson, vice-chairperson and six steering committee members was set in-place. The chairperson would serve a one-year term. The vice-chairperson would serve a one-year term, become the chairperson-elect, and would serve as chairperson of the steering committee. The number of steering committee members were selected to allow a mix of plant and corporate staff to serve in setting direction for the Committee. All members of the administration would be elected by the full membership.

Beyond the direct benefit gained by the interaction from the activities of the Committee meetings, a deliverable product in the form of detailed meeting minutes would be provided. These minutes would be produced by the committee secretary. In order to reduce administrative burden on utility members, the committee secretary has been a staff member from one of the three primary PWR architect/engineering firms of Babcock and Wilcox, Westinghouse, and Combustion Engineering.

In recognition of the fact that plant programs are often influenced by outside agencies, Committee meeting attendance from utility and architect/engineering firm staff is often enhanced by representation from the U.S. Nuclear Regulatory Commission, the Institute of Nuclear Power Operations, and occasionally, the firm of the American Nuclear Insurers.

Committee Meeting Structure

As one of the primary functions of the Committee is the face-to-face sharing of positive and negative experiences, each member utility prepares a written Plant Status Report for each of its operating plants. Information presented on these reports includes:

a. Exposure summaries for power operational periods and outages, along with the respective goals and average daily exposure accrual.

b. Examples of significant contributors to personnel dose (including major outage experiences).

c. Regulatory concerns, including examples of recent violations, findings, open issues, or items identified by utility self-assessment.

d. Recent significant health physics experiences such as; unplanned exposures, near misses, and general lessons learned since the last meeting.

e. ALARA good practices in the areas of; source term reduction, shutdown chemistry practices, and specialized tooling.

As these reports are summarized for entire membership, other members have the opportunity to inquire further on unclear information or simply identify a specific contact for a future time.

Following the presentation of the Plant Status Reports, Committee members will either make formal presentations on defined high interest topics or workshops will be initiated. Specific topics will be covered in presentations by a minimum of two Committee members. This provides for a minimum of redundancy in presentation, yet often provides a totally different approach to solving a like issue. Topics covered by member presentation have included:

a. Source term reduction programs.

b. Exposure reduction programs.

c. Radiation Work Permit program.

d. Expert system technology application.

e. Radiation work practice compliance and enhancement.

f. Steam generator nozzle dam installation robotics.

g. Sub-system chemical decontaminations.

To date, only one workshop forum has been utilized for the Committee meeting. This forum proved to be successful with the general consensus indicating it to be a preferred format. The preference for this type of forum appears to stem from the fact that member participation is increased, and the deliverable product to the utility is more conducive to direct application. The topic for this workshop forum was Cost Effective Radiation Protection/ALARA Programs. Over the years, most utilities have instituted the large payback items such as:

a. Refueling machine overhauls and enhancements,

b. Steam generator inspection robotics and manway door shields,

c. Reactor cavity decontamination systems, and

d. Reactor head shielding.

Having completed these items, dose reductions must now be realized from programs and techniques that are less visible and take longer to provide payback. The intent of this workshop was to facilitate group interaction to highlight smaller scale, cost-effective techniques for dose reduction. From these efforts Committee members identified several categories, under which techniques would be listed and categorized as having high/low payoff, and if the technique was or would be "easy" or "hard" to implement. Tables 1 through 3 provide sample deliverables from this process. After completion of this data gathering session, techniques which were believed to have merit, were listed and attribute plans developed. These plans, samples of which are provided in Figure 1, incorporated the positive and negative experiences of the Committee, as members often times had undertaken similar techniques.

The workshop concluded with a discussion on current topics in radiation protection such as; implementation of the revised 10CFR20, ultra-filtration and hot spot flushing.

Future meetings of the Committee will occur semi-annually. The next meeting is scheduled to occur in conjunction with the BWR Owner Group meeting in Denver, Colorado, July 26-29. The success of this Committee depends on the participation and ownership of its members. Information is the key to success. Members are encouraged to share it and then implement what they have learned.

Author Biography

Daniel Malone is the Radiological Services Superintendent and Acting Radiological Service Department Manager for Consumers Power Company's, Palisades Nuclear Plant. Additionally, he is currently the Chairman of the PWR Radiation Protection/ALARA Commmittee. Prior to these positions, he served as the Palisades plant ALARA Supervisor and Health Physics Instrumentation Supervisor for several years. Beyond his radiation protection activities, he has served as Mechnanical Engineering Section Head, Project Manager for the Safety Related Piping Reverification Program and as the Senior Nuclear Licensing Analyst for the Plant. He has a B.Sc in Environmental Health, with a major in Health Physics from Purdue University.

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Table 1. **JOB PLANNING**

HIGH PAYOFF EASY TO IMPLEMENT	HIGH PAYOFF HARD TO IMPLEMENT
Ensure planning and revisions are done by people doing work (RP ALARA) QA oversight, engineers, contractors, etc. Develop job scope by instilling a questioning attitude, doing cost benefit analysis, considering ALARA goals. Include specific details in job plan (photos, walk-down information, special tools & tooling, probe pusher location, use of mockups, dose reduction evaluation, etc.). Assign responsibility for project work to a designated person to achieve more ownership, accountability. Designate a field coordinator for complex jobs/multi involvement jobs to ensure good handoffs, establishment of time line and working the plan. Minimize impact of radiological conditions and workability on other jobs by proper scheduling, and grouping jobs by location. Strive for off-the-shelf work packages. Ensure planning includes post-job critiques, peer experiences, historical data, past job experiences and assignments. Define the decision tree for contingencies and resolution of emergent issues to achieve consistent assessment and evaluate impact on ALARA goals.	
LOW PAYOFF EASY TO IMPLEMENT	**LOW PAYOFF HARD TO IMPLEMENT**

Table 2. **HIGH TECH APPLICATIONS**

HIGH PAYOFF EASY TO IMPLEMENT	**HIGH PAYOFF HARD TO IMPLEMENT**
Elmer's Glue/NODAC in H_20 for contamination control. ALARA scheduling program. Modular shielding program. H_20_2 shock of tanks for decontamination. Penetration Modifications for services (if spare exists). Electronic teledosimetry. Camera surveillance. Integration camera/teledosimetry/communication. Bar coding/scanning technology.	Chemical decontamination of systems. Broadband cable applications. Electronic dosimetry trending. ALARA electronic dosimetry. Surrogate tours. Digital Imaging. Robotics Seismic analysis. Computerized access/RWPs.
LOW PAYOFF EASY TO IMPLEMENT	**LOW PAYOFF HARD TO IMPLEMENT**
Motion detector. Electronic access to vendor dosimetry data.	Multi-media briefs/training. Transmitting CAMs.

Table 3. **PLANT MODIFICATIONS**

HIGH PAYOFF EASY TO IMPLEMENT	**HIGH PAYOFF HARD TO IMPLEMENT**
Permanent storage of scaffolding and/or shielding. Insulation modification to blanket. Permanent temporary shield supports. Incorporate RP costs in modifications project. Permanent tool room in containment. Fuel transfer system blind flange modification. Quick disconnect fitting in lieu of hard pipe fittings. Change from bolts to studs with SG manways. ALARA design reviews by design engineers.	RTD bypass elimination. Stellite reduction. Piping & test connect modifications in HRA. ie, test connect in lieu of blank flange for ILRT. Modifications for quick install/reserve of NIs access plates. Cavity seal (permanent). Permanent Rx head shield. Permanent platform modifications. Penetration modifications for servers access to containment.
LOW PAYOFF EASY TO IMPLEMENT	**LOW PAYOFF HARD TO IMPLEMENT**
Eliminate filters if possible.	Increase use of "live load" packed valves.

Item 2:

1. **Description of Recommended Practice:**

Enhance management oversight
-Plant walkdowns
-Coach workers

2. **Program Category:**

Do more with less

3. **Intended Goal and Benefit to Plant:**

Goal: Coaching for quality
Improve radiation worker practices

Benefit: Reduce exposure and cost
Improved productivity, ALARA awareness

4. **Recommended do's and don'ts for implementation:**

Do: Have written objectives
Lead by example and coach
Correct on spot and follow-up

Don't: Use as discipline unless repetitive problem

5. **Advice on presenting or selling the practice to management:**

Builds teamwork
INPO/NRC relations
Very low cost

6. **Contact person for information/support (and phone no.):**

Ted Bast	805-545-4588
Dave Ethridge	717-948-8011
Bruce Watson	410-586-2200

Figure 1. Attribute Plan (Cont.)

Item 1:

1. Description of Recommended Practice:

Work Planning and Scheduling

2. Program Category:

Do more with less

3. Intended Goal and Benefit to Plant:

Goal: Reduce rework, optimize work schedule

Benefit: Improved use of resources
Better unit availability
Improve moral of workers

4. Recommended dos and don'ts for implementation:

Do: Accurately schedule time and utilization of manpower
Develop integrated schedule
Work system windows
Open communications between all work groups
Management commitment
Control emergent work and update schedule

5. Advice on presenting or selling the practice to management:

Reduced exposure/costs
Outage criteria path control

6. Contact person for information/support (and phone no.):

Ted Bast	805-545-4588
Dave Ethridge	717-948-8011
Pat Burke	203-447-1291

Figure 1. Attribute Plan

Item 3:

1. **Description of Recommended Practice:**

   Designate a field coordinator for complex jobs/multi involvement jobs to ensure good hand offs, establishment of time lines and working the plan.

2. **Program Category:**

   Job planning.

3. **Intended Goal and Benefit to Plant:**

   Reduce delays
   Increased efficiency
   Reduce outage length
   Resolve conflicts quickly

4. **Recommended do's and don'ts for implementation:**

   Don't assign to inexperienced person
   Do use experienced person
   Do clearly define responsibilities and authority

5. **Advice on presenting or selling the practice to management:**

   Save time, dose, money
   Provides continuity

6. **Contact person for information/support (and phone no.):**

   Chris Hubbard ANO 501-964-5070
   Gary Sturm, Palisades 616-764-8913

Figure 1. Attribute Plan (Cont.)

SESSION 7B

BWR AND GAS-COOLED PRESENTATIONS

Co-chairs:

Harvey J. Cybul
John F. Schmitt

ALARA EFFORTS IN NORDIC BWRs

Tor Ingemansson
Klas Lundgren and Jan Elkert
ABB Atom, S-721 63 Västerås, Sweden

ABSTRACT

Some ALARA-related ABB Atom projects are currently under investigation. One of the projects has been ordered by the Swedish Radiation Protection Institute, and two others by the Nordic BWR utilities. The ultimate objective of the projects is to identify and develop methods to significantly decrease the future exposure levels in the Nordic BWRS.

As 85 to 90% of the gamma radiation field in the Nordic BWRs originates from Co-60, the only way to significantly decrease the radiation doses is to effect Co and Co-60. The strategy to do this is to map the Co sources and estimate the source strength of Co from these sources, and to study the possibility to affect the release of Co-60 from the core surfaces and the uptake on system surfaces. Preliminary results indicate that corrosion/erosion of a relatively small number of Stellite-coated valves and/or dust from grinding of Stellite valves may significantly contribute to the Co input to the reactors. This can be seen from a high measured Co/Ni ratio in the feedwater and in the reactor water. If stainless steel is the only source of Co, the Co/Ni ratio would be less than 0.02 as the Co content in the steel is less than 0.2%. The Co/Ni ratio in the reactor water, however, is higher than 0.1, indicating that the major fraction of the Co originates from Stellite-coated valves.

There are also other possible explanations for an increase of the radiation fields. The Co-60 inventory on the core surfaces increases approximately as the square of the burn-up level. If the burn-up is increased from 35 to 50 MWd/kgU, the Co-60 inventory on the core surfaces will be doubled.

Also the effect on the behavior of Co-60 of different water chemistry and materials conditions is being investigated. Examples of areas studied are Fe and Zn injection, pH-control, and different forms of surface pre-treatments.

INTRODUCTION

The annual collective exposures in the Nordic BWRs have traditionally been low, however, showing an increasing trend during the last five years (Figure 1). Significant increases have especially been experienced during 1992 and 1993. In order to counteract this trend, some ALARA-related projects have been initiated by ABB Atom. The first is called DORIS (Dose Reduction in Swedish BWRs), and is ordered by the Swedish Radiation Protection Institute. The second project is called ALARA 2000, and is ordered by the utilities OKG, Ringhals, Barsebäck, and TVO. A related project, an update of the ABB Atom developed computer code BKM CRUD, is separately ordered by Forsmark NPP. The third project, KEMOX 2000, is ordered by the Swedish utilities and Nuclear Inspectorate, and is jointly run by ABB Atom and Studsvik Material. The main purpose of that project is to develop methods to optimize the performance of oxides in the BWR primary system. One method currently being investigated is pH control.

The purpose of this presentation is to report the current status of the DORIS and ALARA 2000 projects.

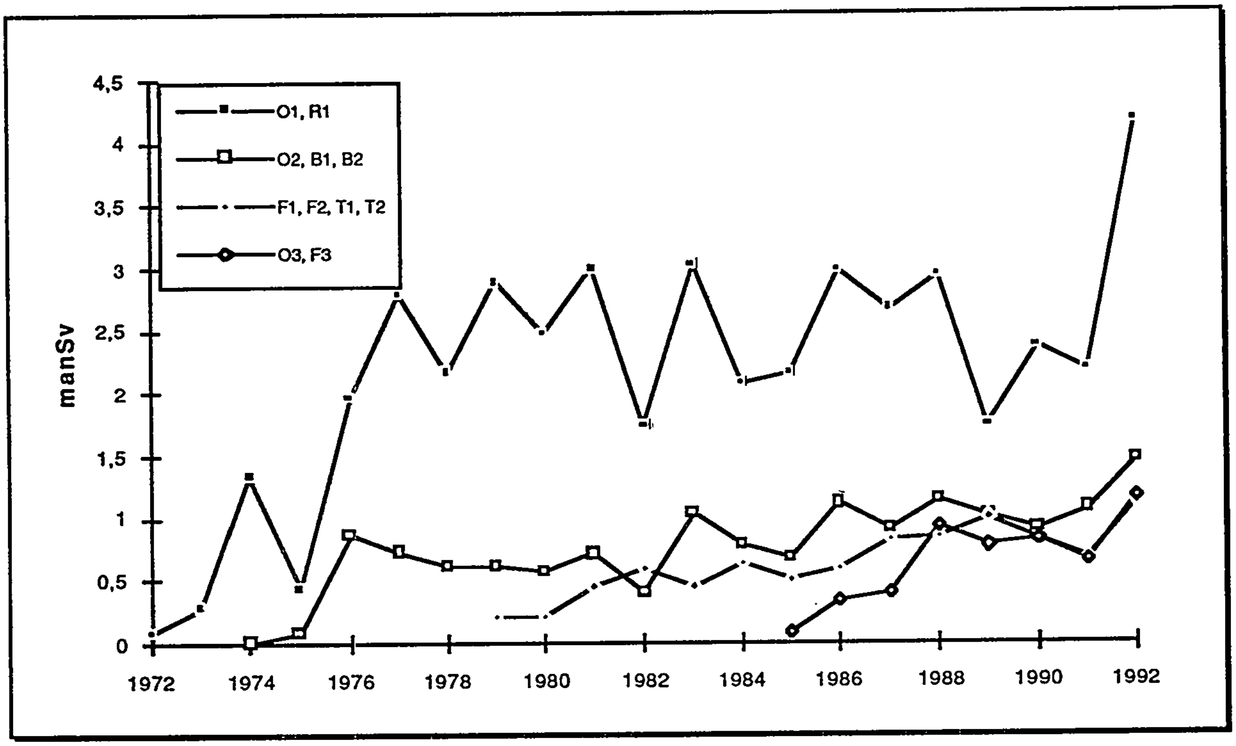

Figure 1. Annual occupational exposure in different ABB Atom reactor generations (manSv per reactor unit)

THE DORIS PROJECT

General Outline

In foreign BWRs, the annual occupational exposures have been gradually decreasing for some years (Figure 2). In Nordic BWRs, the occupational exposures have been low. During the last five years they have increased significantly. This trend is of concern. The DORIS project has been initiated in order to map factors which significantly affects the exposure. Among others the following items will be addressed:

- The effect of increased burn-up of the fuel.
- The effect of HWC-operation.
- Exposure statistics for different types of jobs.
- The possibility to use extra shielding to decrease the exposure. Chemistry control, e. g. ph-control, Fe injection to the feedwater, etc.
- Optimization of inspection programs.
- Optimization of operational procedures, e.g. fuel failure management.

The ABB Atom developed computer code BKM-CRUD2 will be used as a tool to assess the importance of the different items.

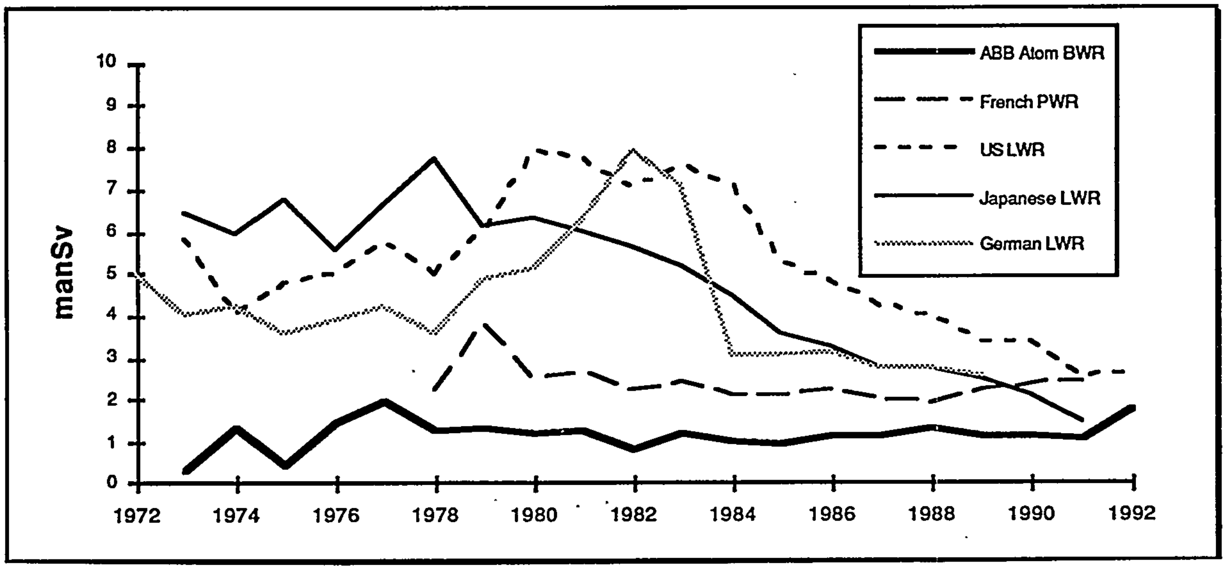

Figure 2. Annual occupational exposure per unit of ABB Atom BWRs compared to international LWR standard

Some Results

Increased burn-up levels of the fuel will increase the fuel crud Co-60 activity. The combined effect of a with time linear build-up of Co and increasing specific activity of Co-60 means, that the Co-60 inventory increases proportional to the square of the burn-up level (Figure 3).

An increase of the burn-up level from 35 to 50 MWd/kgU increases the amount of Co-60 on the fuel by a factor of two. The potential for release of Co-60 from the fuel will increase correspondingly.

Another fuel related problem affecting the radiological conditions is operation with defected and degrading fuel. Then the core will be contaminated by tramp uranium, causing a significant background activity level of fission products. The dose rates in the turbine system will increase because of the presence of Ba- and La-140. Ba-140 (Tl/2 = 12.8 d) is a daughter product of Cs-140 (65.5 s), which is a daughter product of the noble gas isotope Xe-140 (13.6 s). La-140 (40.3 h) is a daughter product of Ba-140. The high gamma energies emitted from Ba- and La-140 significantly increase the dose rates especially in the high pressure part of the turbine. Another important result of contamination of the core with tramp uranium is an increased release rate of fuel crud (e.g. Co-60) because of knock-out reactions.

Models for estimation of the radiological impact because of dispersion of fuel during operation with defected rods will be assessed. Models for estimations of the dissolution rate of U02 and the uptake of uranium on core surfaces have earlier been discussed.[1]

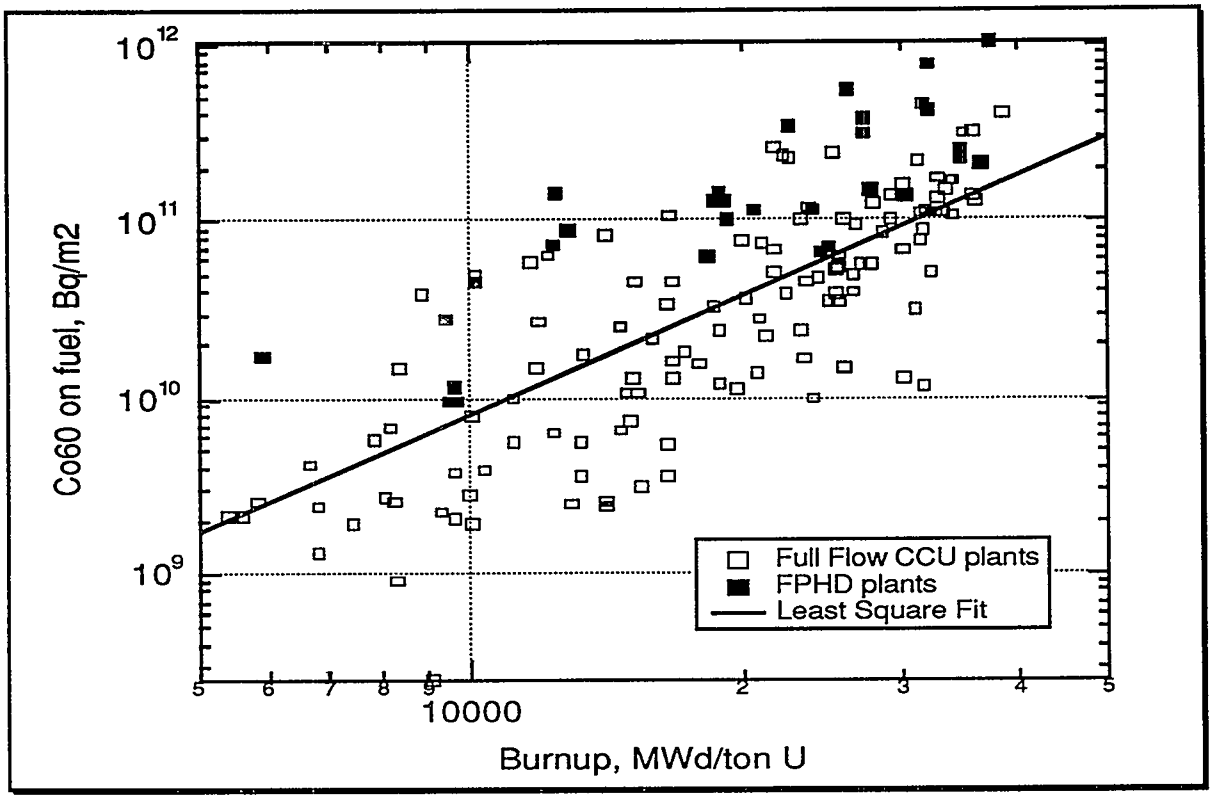

Figure 3. Co-60 in fuel crud as a function of fuel burnup level.

THE ALARA 2000 PROJECT

General Outline

The ALARA 2000 project is divided into three sub projects:

1. Estimation of the future radiological conditions for each of the participating plants by use of the BKM-CRUD code.[2] This estimation shall be carried out with the current measured feedwater concentrations extrapolated to the year of 2010 as input to the code. Three to five parameters dominating the effect on the radiological conditions shall be identified plant specifically.

2. Corrosion product balances. The objective is to assess the mass balances for Fe, Ni, Cr, Co, Zn and Cu for the plants. As discussed above, the mass balance and source strength study for Co is especially important.

3. Within the third ALARA 2000 sub project a correlation study between measured dose rates and gamma scan data and operational data from the plants will be performed. The objective of this task is to identify specific operational practices at the plants that have significantly affected the radiation levels.

Some Results

Figure 4 shows measured shut-down radiation levels on a vertical RWCU pipe in the different ABB Atom BWRS. The data clearly indicate, that significantly increasing levels are still experienced after 15-20 years of operations.

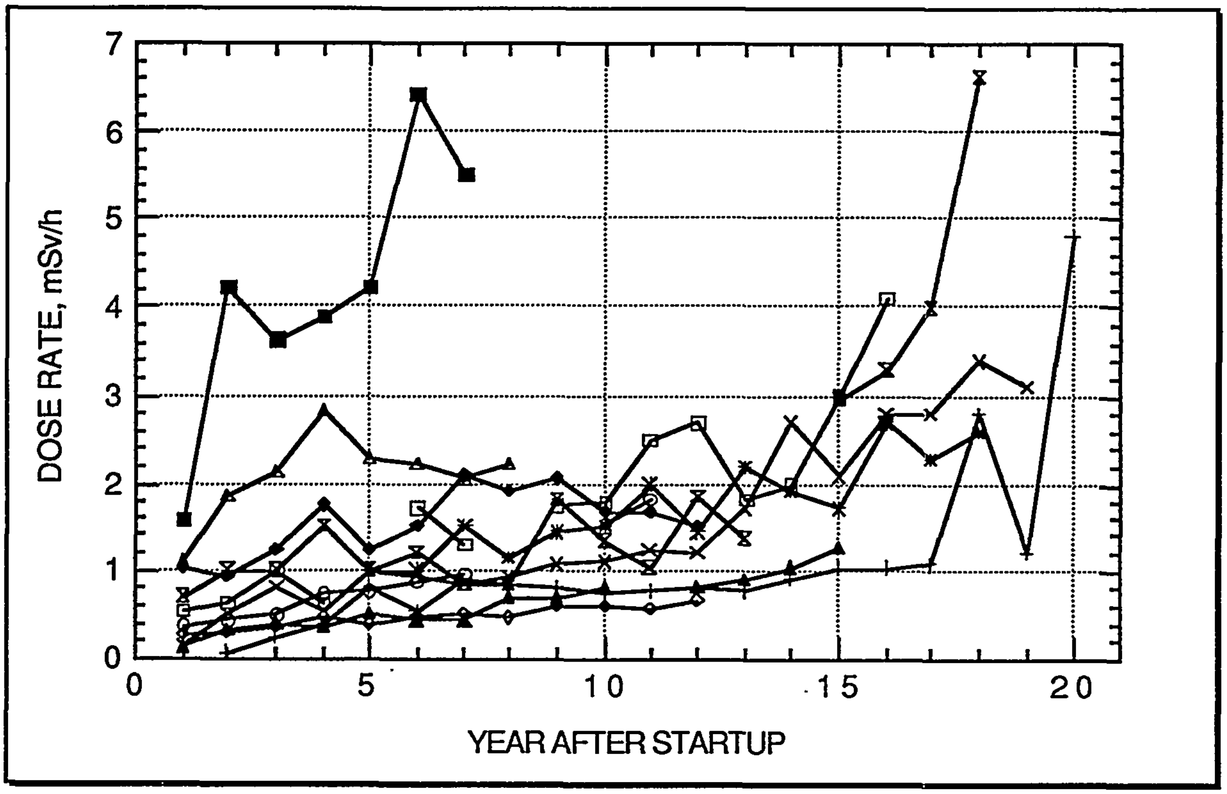

Figure 4. Shut-down radiation levels on a vertical RWCU pipe in different ABB Atom BWRs

Gamma scan data show that Co-60 contributes with about 70-90% of the total dose rate from the pipe.

The currently adopted strategies for the ALARA 2000 project are:

1. To reduce the input of Co to the reactor.
2. To minimize the release of Co-60 from the core surfaces.
3. To minimize the uptake on system surfaces.
4. To shield components which significantly contribute to the gamma radiation fields.

The stainless steel surfaces, approximately 7000-14000 m^2, in the feedwater train contains less than 0.2% Co in the Nordic reactors. This means that the Co/Ni ratio in this steel is less than 0.02. In Figure 5 the measured Co/Ni ratio in the feedwater in one Nordic BWR is presented, and in Figure 6 the corresponding ratio in the reactor water.

The Figures 5 and 6 show that the Co/Ni ratios are significantly higher than 0.03, indicating that there are other Co sources than corrosion of stainless steel.

A preliminary study has indicated that there are two possible sources for this additional Co: corrosion/erosion of specific valves containing Stellite, and Stellite dust produced during valve grinding. Both this sources have earlier been recognized.[3,4,5]

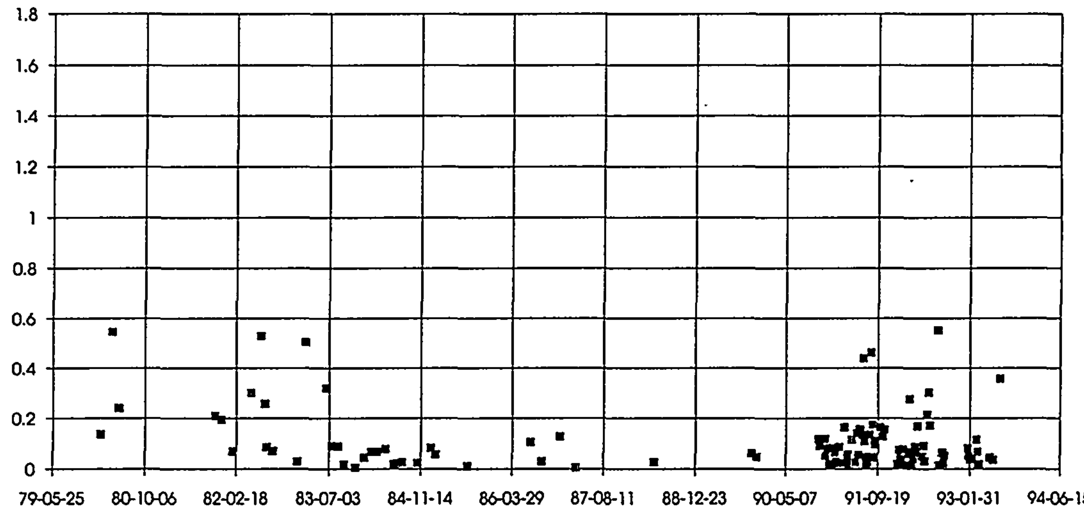

Figure 5. The Co/Ni ratio in the feedwater in one Nordic BWR

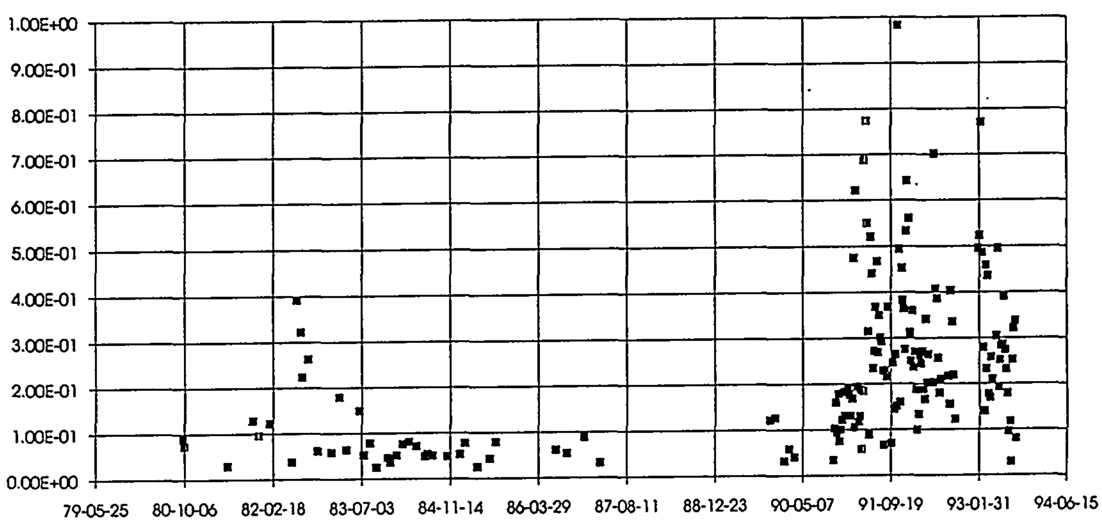

Figure 6. The Co/Ni ratio in the reactor water in one Nordic BWR

For the proceeding discussions, valves containing Stellite will be separated in three categories depending on technical function and radiological impact:

- <u>Red valves.</u> These are characterized by such high corrosion/erosion rates, up to 900 mdm (mg/dm2 and month) has been recorded,[6] that their technical function will be affected. Of course, these valves are significant sources for Co to the reactor. These valves are not acceptable for future use. It is recommended for the outage -94 that as many as possible of these red valves are to be identified and the Stellite exchanged. All red valves should be exchanged at latest during the outages 1995.

- <u>Yellow valves.</u> These are characterized by a high corrosion/erosion rate which significantly contribute to Co inflow to the reactor. However, technically the valves can be accepted. These valves should be identified during the outage -94. Corrosion/erosion rates for individual valves should be estimated. Some of these valves will be exchanged and made Stellite free. A cost-benefit study of the exchange of yellow valves will be carried out plant and valve specifically.

- **Green valves.** These are characterized by low or very low corrosion rates and thus do not have any significant radiological impact. There are no reasons to make any changes of the green valves.

The total input of Co to a 1000 MWel reactor of ABB Atom design can be estimated to 50 to 100 g per 8,000 EFPH (Effective Full Power Hours) and the total number of valves are about 12,000. There are, however, only a very limited amount of valves which causes radiological problems.

Also the chemical behavior of Co and Co-60 will be studied. The release of Co-60 from the fuel surfaces and uptake on system surfaces will be treated statistically to try to find parameters which govern the behavior of Co-60. This will be done by comparisons between fuel scraping data, reactor water concentrations, and system deposits of metals and activated corrosion products. Also sampling and measuring procedures will be examined.

The Nordic BWRs have very low Fe concentrations in the feedwater. The effects on the behavior of Co-60 of Fe injection to the reactor are therefore addressed.

As the DORIS and ALARA 2000 projects are running at the same time, the results produced within one of the projects will be used in the other.

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Tor Ingemansson is an ABB Atom specialist in Radiochemistry. He has worked for 8 years at Forsmark Nuclear Power Plant and for 3.5 years at ABB Atom. He has a Ph. D. in Nuclear Physics from the University of Lund, Sweden. Dr. Ingemansson's specialties are fuel failure management and ALARA-related issues.

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RADIATION FIELD CONTROL AT THE LATEST BWR PLANTS -- DESIGN PRINCIPLE, OPERATIONAL EXPERIENCE AND FUTURE SUBJECTS

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ABSTRACT

Improvements of operational procedures to control water chemistry, e.g.,nickel/iron ratio control, as well as application of hardware improvements for reducing radioactive corrosion products resulted in an extremely low occupational exposure of less than 0.5 man.Sv/yr without any serious impact on the radwaste system, for BWR plants involved in the Japanese Improvement and Standardization Program. Recently, ^{60}Co radioactivity in the reactor water has been increasing due to less crud fixation on the too smooth surfaces of new type high performance fuels and to the pH drop caused by chromium oxide anions released from stainless steel structures and pipings. This increase must be limited by changes in water chemistry, e.g., applications of modified nickel/iron ratio control and weak alkali control. Controlled water chemistry to optimize three points, the plant radiation level and integrities of fuel and structural materials, is the primary future subject for BWR water chemistry.

INTRODUCTION

Occupational exposure at nuclear power plants is determined by the three factors:
1) the radiation level where major inspection and maintenance operations are carried out;
2) the work time for each maintenance operation; and
3) the number of personnel needed for each maintenance operation.

In order to reduce the radiation level, control of radioactive corrosion products is essential and the procedures for controlling corrosion products without any serious impact on the radwaste system have been applied for plant operation as well as design of plant systems and major hardware at BWRs involved in the Japanese Improvement and Standardization Program (JISP)[1]. At the JISP BWRs, systematic applications of the procedures led to extremely low radiation levels which reduced occupational exposure to less than 0.5 man.Sv/yr.

As a result of analyzing current water chemistry data, it is found that ^{60}Co radioactivity in the reactor water at the JISP BWRs is again increasing with the application of high performance fuel and the pH drop caused by Cr anion being released from stainless steel structures and pipings. Changes in fuel and structural materials have some impact on corrosion product behavior, while improvement of the water chemistry for controlling radioactive corrosion products has some impact on behaviors of fuel and structural materials.

HISTORIC ASPECTS OF BWR WATER CHEMISTRY

Since the first Japanese BWR, Tsuruga-1, started commercial operation in 1970, a lot of experiences with water chemistry and radiation control in BWR plants have been accumulated. Some historic aspects of water chemistry and radiation control of BWRs are summarized in Table 1. The experiences can be divided into six periods and topics of greatest interest have changed from simply fuel integrity, structural integrity and occupational exposure to their combination.

Table 1 Historic aspects of water chemistry and radiation control in BWRs

Period	Major events	Water chemistry & radiation control
1 (Before 1975)	observation of fuel defects radioactive effluent increase	fission product removal
2 (1975 - 1980)	IGSCC occurrence occupational exposure increase	impurity control oxygen injection
3 (1980 - 1985)	Japanese Improvement and Standardization Program	dual condensate demineralizers low cobalt containing materials
4 (1985 - 1990)	BWR plants with low shutdown radiation level	Ni/Fe ratio control
5 (1990 - 1995)	^{60}Co increase at every latest operating cycle	advanced Ni/Fe ratio control
6 (After 1995)	to establish a trio of water chemistry requirements	controlled water chemistry (Ni/Fe ratio & pH control, HWC)

During the first period, poor water chemistry often caused defects on fuel claddings. Some impurites, e.g., suspended iron crud, deposit on the fuel surfaces to form thick deposit layers, which not only deppress reactivity of the core but also prevent sufficient heat transfer from the fuel to the coolant which increases temperatures at the fuel surface and subsequently enhances the corrosion rate. In order to improve fuel integrity, suppression of crud concentration in the reactor water was applied along with structural improvements, e.g., moderation of mechanical interactions between cladding and UO_2 pellets, and operational improvements, e.g., pre-conditioning for densification of UO_2 pellets.

The major target in the second period was IGSCC of stainless steel pipings. Some impurities in the reactor water, e.g., chloride ion, enhanced IGSCC of stainless steel pipings. Concentrations of radiolytic species, e.g., H_2O_2, OH, and O_2^-, should be suppressed to moderate corrosive circumstances, while those of metallic and organic impurities, which also enhance corrosion, should be controlled. In order to avoid IGSCC of the piping, suppression of chloride concentration and conductivity in the reactor water were applied along with developments of IGSCC resistant materials, e.g., low carbon containing stainless steel, and residual stress improvement. During this period, inspection and maintenance operation of primary pipings caused high levels of occupational exposure, which hindered effects to win public acceptance of nuclear power plants. Then, lowered exposure become the main target for the third period.

Small amounts of corrosion products released into the cooling water, such as ^{60}Co, are activated in the core, becoming radioactive. Some of these products deposit on the walls of the recirculation pipings and their components, which causes shudown doses around the primary cooling system, and then, radiation exposures of personnel working on inspection and maintenance around the primary system. In the third period, the desire for reduced occupational exposures required much severer criteria for water chemistry, particularly radioactive corrosion product control. The primary procedures for reducing corrosion products, the effects of which were evaluated on the shutdown dose rate, were incorporated into the Japanese Improvement and Standard Program (JISP), the details of which are described in the following section.

In the fourth period, Fukushima Daini-2, which was the first BWR designed and constructed through involvement in the JISP, started its commercial operation. In spite of its application of radiation reduction procedures too low iron crud concentration caused an increase of ^{60}Co radioactivity in the reactor water, which was moderated by assistance of additional water chemistry control. ^{60}Co in the reactor water could be successfully reduced by adding suitable amounts of iron crud in the feed water and enhancing redeposition and fixation of ^{60}Co as cobalt ferrite at fuel surface. In order to suppress insoluble ^{60}Co radioactivity in the reactor water, the added iron crud should be controlled to keep a suitable nickel/iron ratio (Ni/Fe ratio < 0.5) of minimum amounts to form $CoOFe_2O_3$ and $NiOFe_2O_3$ at the fuel surface.

Recently, ^{60}Co radioactivity in the reactor water at the JISP BWRs is increasing due to less crud fixation on the too smooth surfaces of the new type high performance fuels and to the pH drop caused by chromium oxide anions released from stainless steel pipings. Presently, in the fifth period, attempts are being made to decrease ^{60}Co radioactivity by applying weak alkali control and improved Ni/Fe ratio control.

Water chemistry is affected by behaviors of fuel and structural materials, while fuel and structural materials are also affected by water chemistry. In the future's sixth period, establishment of "controlled water chemistry" will be the goal which will satisfy a trio of requirements, shutdown radiation reduction, integrity of fuel cladding and integrity of structural materials.

JAPANESE IMPROVEMENT AND STANDARDIZATION PROGRAM

The basic design concept of the JISP BWRs is characterized by the following features.

a) High reliability: duty factor of 70%.
b) Minimum shutdown period for scheduled refueling and annual maintenance: 85 days.
c) Minimum occupational exposure: 1.3 - 1.5 man.Sv/yr.

Occupational exposure is determined by three factors 1) - 3) listed in the Introduction. The basic concepts and main procedures for reducing occupational exposure through these three factors are shown in Table 2.
In order to reduce the radiation level, control of radioactive corrosion products is essential. Procedures with sufficient effects and high reliabilities have been systematically applied in JISP BWRs, and efforts have also been made to reduce the radwaste sources, which accompany the increase of corrosion product removal. A schematic diagram for radioactive corrosion product behavior is presented in Figure 1. The processess for

radioactive corrosion product behavior are divided into three steps, e.g., generation, activation and deposition, where suitable procedures for controlling corrosion products are applied. The procedures for reducing shutdown dose rate applied in JISP BWRs are shown in Figure 2.

Occupational exposure at the JISP BWRs during the first refueling and annnual inspection periods can be reduced to less than 0.5 man.Sv/yr by applying suitable Ni/Fe ratio control (Figure 3).

Table 2 Basic concepts and main procedures for reducing occupational exposure

Basic concepts	Shutdown radiation reduction	Improvement of personnel mobility	Automation for inspection & operation
Main procedures	(1) Low cpobalt containing materials (2) Oxygen injection into feed water (3) Prefiltration of condensate water (4) Feed water recircu-lation system	(1) Application of improved Mark II type PCV*1 i) Increasing operation space ii) Improvement of carriage path	(1) Automatic ISI*2 machine (2) Automatic CRD*3 exchanger (3) Automatic fuel assembly exchanger (4) Improved LPRM*4 exchanger

*1 pressure containment vessel *2 in-service inspection
*3 control rod drive *4 local power range monitor

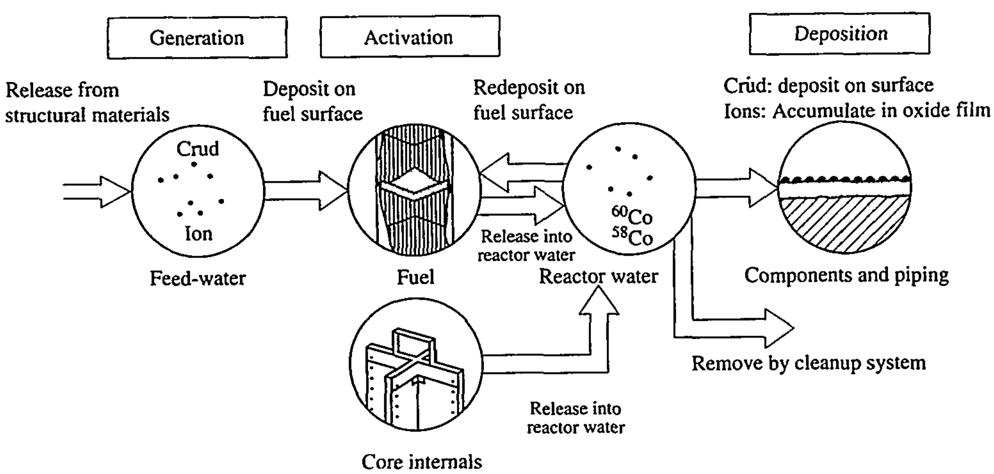

Figure 1 Radioactive corrosion product behavior

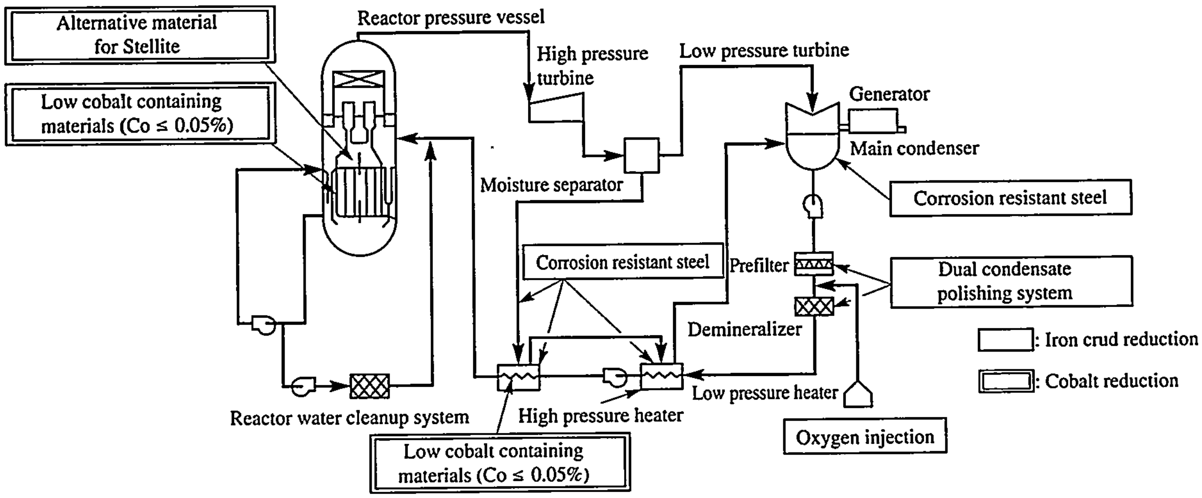

Figure 2 Application of procedures for reducing shutdown dose rate

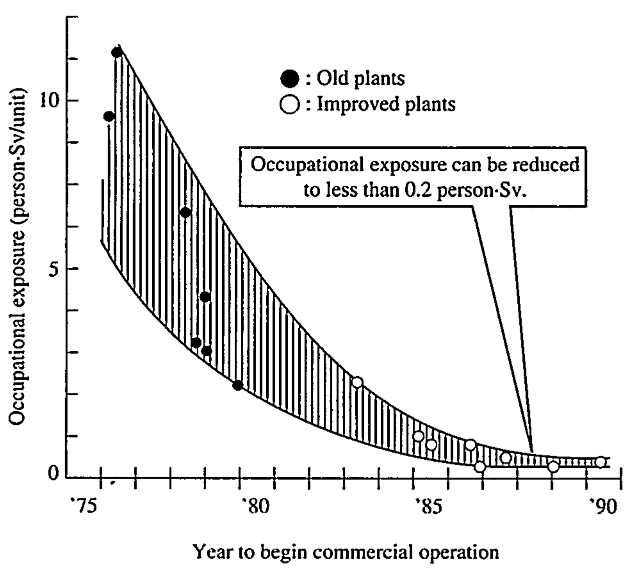

Figure 3 Occupational exposure during first refueling and annual inspection periods

EXPERIENCES WITH WATER CHEMISTRY OF THE LATEST BWRS

The relationship between shutdown dose rate and occupational exposure is shown in Figure 4. With a few notable exceptions, occupational exposure is in proportion to the shutdown dose rate, which is also proportional to ^{60}Co radioactivity in the reactor water. The average ^{60}Co radioactivity in the reactor at each operational cycle of several JISP BWRs is shown in Figure 5. Some values for the latest cycle exceed the target value of 2 Bq/ml necessary to keep a shutdown dose rate of less than 0.5 mSv/h. Increased ^{60}Co radioactivity seems to be caused by changes in water chemistry and fuel.

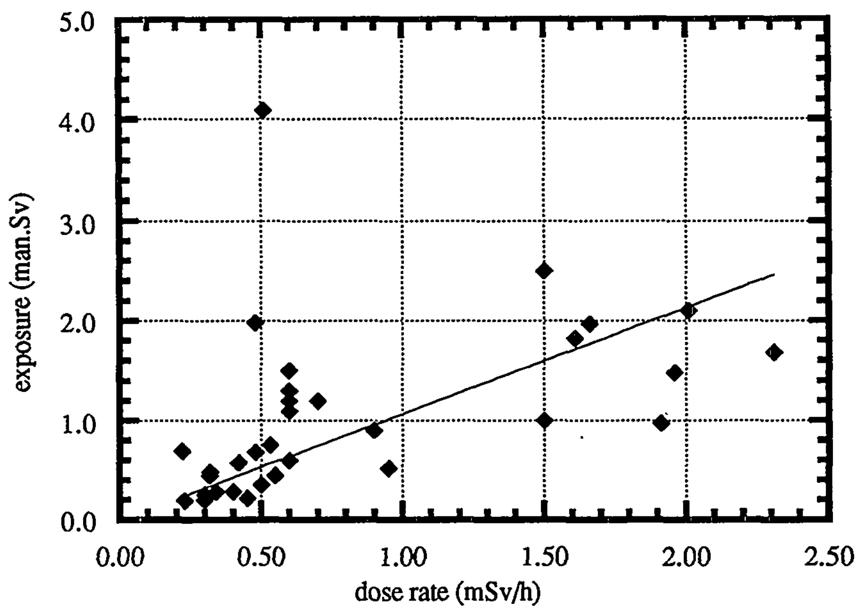

Figure 4 Relationship between shutdown dose rate and occupational exposure

One of the typical changes in water chemistry in the latest BWRs is decreasing pH in the reactor water[2]. Increasing chromium anion concentration decreases pH in the reactor water, making it slightly acidic (6.6 - 6.8). This enhances ^{60}Co release from deposits on the fuel surface which in turn increases ^{60}Co radioactivity in the reactor water. The relationship between ^{60}Co radioactivity and pH in the reactor water is shown in Figure 6. Although ^{60}Co radioactivity is affected not only by pH, but also by concentrations of iron curd and cobalt, Figure 6 shows that rather high pH causes low ^{60}Co radioactivity, while low pH causes rather high radioactivity. As a result of injecting sodium hydroxide in the reactor water for weak alkali control, the contribution of Cr anion to pH is moderated and then pH increases to moderate the ^{60}Co radioactivity buildup.

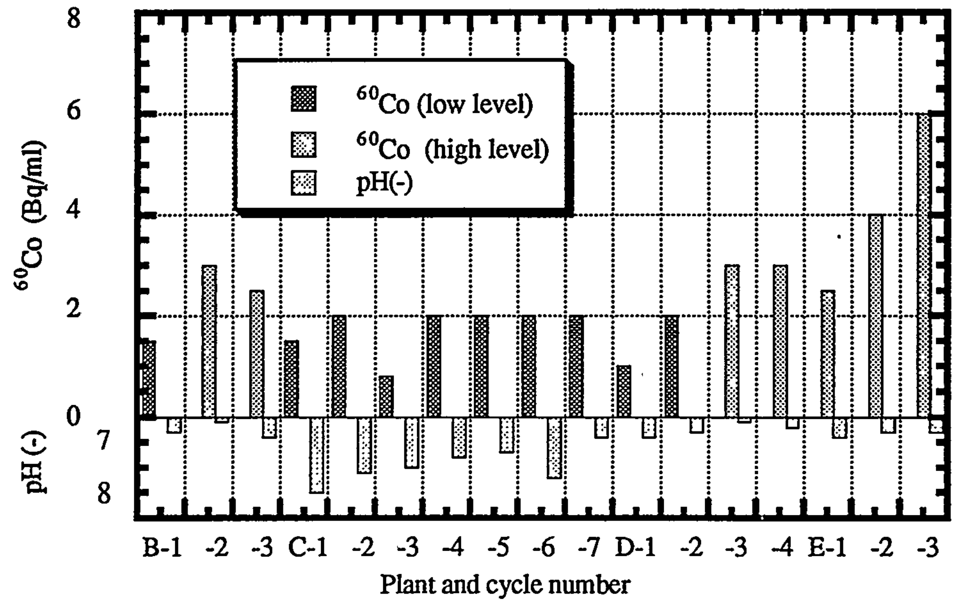

Figure 5 Relationship between ^{60}Co radioactivity and pH in the reactor water at each operational cycle of four BWR plants

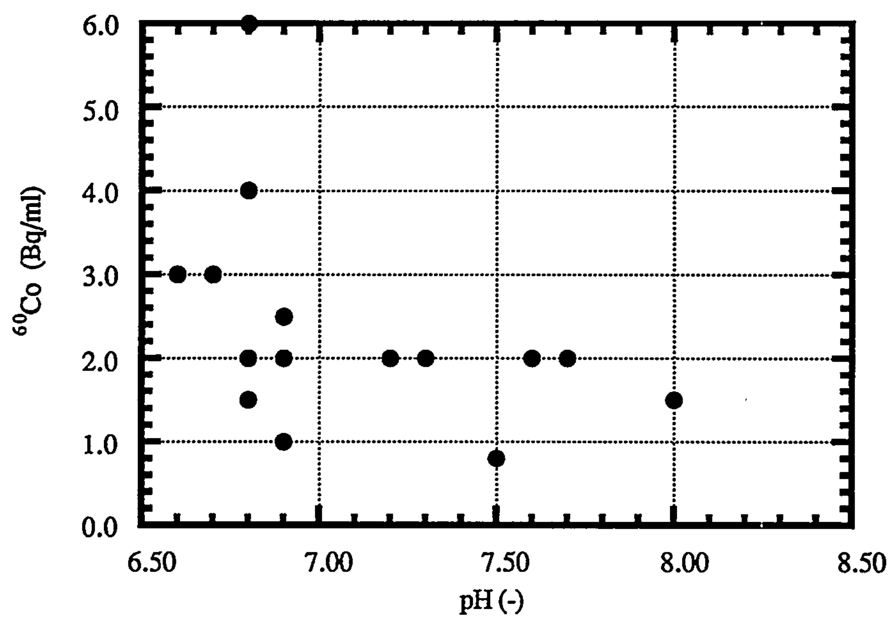

Figure 6 Relationship between ^{60}Co radioactivity and pH in the reactor water

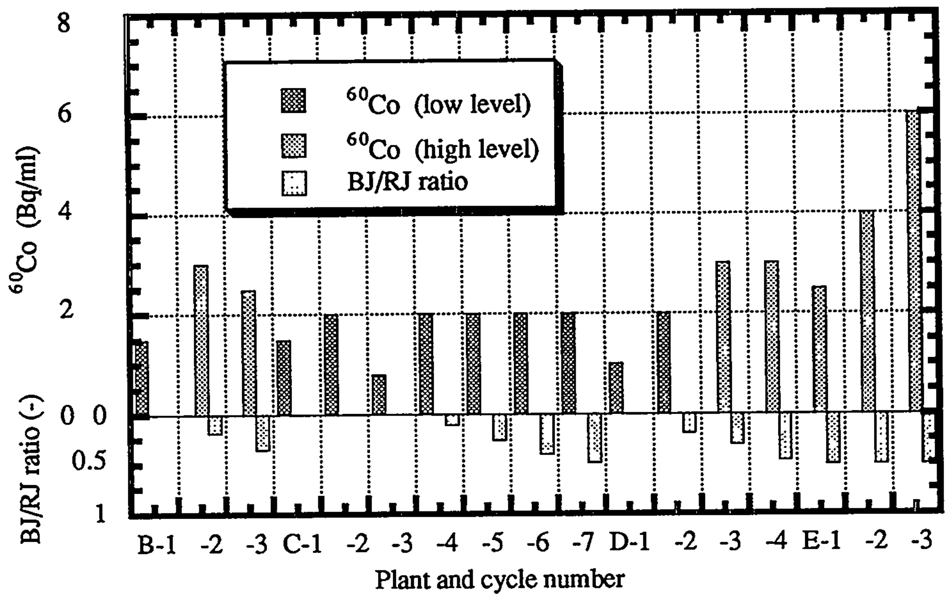

Figure 7 Relationship between ^{60}Co radioactivity and BJ fuel load factor in the core at each operational cycle of four BWR plants

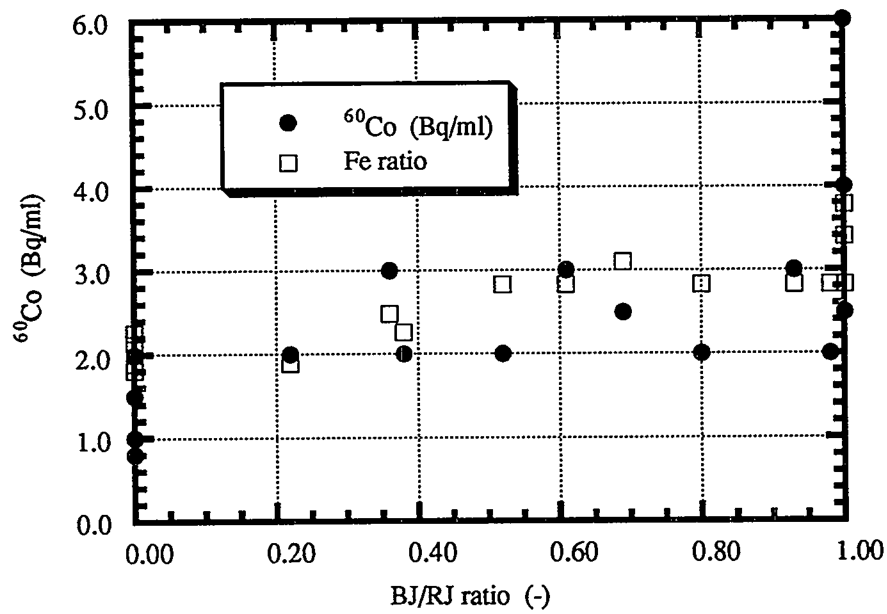

Figure 8 Relationship of ^{60}Co radioactivity and Fe ratio of [real crud] to [ideal crud] to the BJ fuel load factor

Another change in water chemistry is iron crud concentration in the reactor water[2]. As a result of applying new type high performance fuel (designated as BJ fuel) with mechanically polished smooth surfaces, the deposition rate of crud on them is reduced, which causes some delay in fixation of cobalt as cobalt ferrite to increase ^{60}Co radioactivity in the reactor water (Figure 7), even if enough iron crud is supplied to satisfy a suitable Ni/Fe ratio. Iron crud comes mainly from the feed water line, while it is removed mainly at the fuel surface. It is easy to estimate crud concentration in the reactor water by using crud concentration in the feed water and the crud deposition coefficient at the fuel surface. As a result of applying the high performance fuel, the iron crud concentration in the reactor water is higher than the estimated values.

The ratio of measured iron crud concentration (the real concentration) to the calculated ones (the ideal concentration) is proportional to the load factor of BJ fuel in the core as shown in Figure 8. Once deposited on the fuel surface, iron crud is easily released into the reactor water, and then repeats of deposition and release follow to cause a high concentration in the reactor water. Cobalt deposited on the fuel surface missing its partner to form cobalt ferrite, and there is some delay for fixation to occur which increase ^{60}Co radioactivity in the reactor water. The relationships between ^{60}Co radioactivity or the ratio of real crud concentration to the ideal one, to the BJ fuel load factor are shown in Figure 8. The data support the contributions of BJ fuelapplication to increasing ^{60}Co radioactivity.

SHORT TERM COUNTERMEASURES FOR RADIATION REDUCTION

In a few years, water chemistry control will be focused on reduction of radioactive corrosion product accumulation, so as to avoid any effects of changes in fuel cladding and structural materials.

In order to reduce ^{60}Co radioactivity, weak alkali control (pH 7.0 - 8.0) is being applied to moderate ^{60}Co release from fuel surfaces, while improved Ni/Fe ratio control, in which much iron crud is supplied at an early stage to cover the fuel surface completely with crud and thus improve the crud deposition rate, is also being applied to enhance fixation of ^{60}Co on the fuel surface as cobalt ferrite[2].

LONG TERM COUNTERMEASURES FOR CONTROLLED WATER CHEMISTRY

Cooling water, fuel claddings and structural materials should be optimally selected to satisfy a trio of requirements, reduction of shutdown radiation level, integrity of fuel cladding and integrity of structural materials. Water chemistry should be controlled to improve performances of fuel cladding and structural materials, while fuel cladding and structural materials should also be moderated to improve water chemistry. A combination of Ni/Fe ratio control, weak alkali control and fuel cladding modification for radiation reduction has been proposed.

"Controlled water chemistry" seems to be the major route in the future which will allow the three requirements to be satisfied through reliable and easy chemical procedures and with less influences of changes in plant operation procedures and structural materials[3-4]. Severe water chemistry control is necessary, but some of the target values for the different requirements conflict with each other and optimal target values must be determined to balance them. In order to establish "Controlled Water Chemistry," first effects of major water chemistry factors on materials and corrosion product behaviors at elevated temperatures under radiation (and under the reactor water condition) should be quantitatively evaluated, from which the mechanism determining the effects can be derived. It is important to control water chemistry by evaluating the current plant status as well as predicting long term effects on materials.

CONCLUSIONS

Water chemistry control with mono-purpose optimization for reduction of occupational exposure has been established by evaluating actual plant operational data in JISP BWRs with lower shutdown radiation levels. In the future, "water chemistry control" with tri-purpose optimization can be established by confirming fundamental data of water chemistry and materials and then estimating plant future trends.

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Author Biography

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PAPER 7B-2
DISCUSSION

Unknown: I have a collective question. I know that your major enemy is cobalt-60, but I would like to know the different strategies in the different countries dealing with fuel failures in BWRs. Do you perform a major shutdown, or do you wait for a plant outage? If you didn't, have you an idea of the evolution of the dose rates of the alpha contaminations with the activity of neptunium or fission products?

Uchida: Fortunately, for several years we have not had any serious fuel damage. Fuel defects are caused by some impurity of coolant, but we have very few experiences of fuel leakage, so we don't worry about this contamination of neptunium or the fission products. We only think only about cobalt-60 for determining the shut-down dose rate.

ALARA DATABASE VALUE IN FUTURE OUTAGE WORK PLANNING AND DOSE MANAGEMENT

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ABSTRACT

ALARA database encompassing job-specific duration and man-rem plant specific information over three refueling outages represents an invaluable tool for the outage work planner and ALARA engineer. This paper describes dose-management trends emerging based on analysis of three refueling outages at Clinton Power Station. Conclusions reached based on hard data available from a relational database dose-tracking system is a valuable tool for planning of future outage work. The system's ability to identify key problem areas during a refueling outage is improving as more outage comparative data becomes available.

Trends over a three outage period are identified in this paper in the categories of number and type of radiation work permits implemented, duration of jobs, projected vs. actual dose rates in work areas, and accuracy of outage person-rem projection. The value of the database in projecting 1 and 5 year station person-rem estimates is discussed.

INTRODUCTION

Clinton Power Station is a General Electric Boiling Water Reactor, BWR-6, Mark III Containment. Clinton Power Station commenced commercial operation in 1987. The station achieved 320 days of continuous operation in 1992-93. Highlights of the ALARA aspects of the radiation protection program are summarized below:

- Dose to the public is minimized due to an aggressive program to avoid failed fuel and maintain zero liquid discharges to the environment. Clinton Power Station is the only BWR-6 which has not experienced fuel failures. CPS has achieved zero liquid discharge in 1993 and year-to-date in 1994.

- Personnel contaminations have been maintained at a low level from 1988-92. Each year the station experienced approximately 33 skin contaminations and 63 clothing contaminations for the first four years of operation. In 1993, a 30% increase in personnel contaminations was experienced largely due to the implementation of new 1OCFR20 requirements.

- High emphasis has been given to plant and system cleanliness. Refueling contractors are indoctrinated on the importance plant management places on plant cleanliness and foreign material exclusion from the reactor vessel and primary system. This emphasis has resulted in the Mark III containment entries continuing to be a street clothes entry zone during non-outage periods. We understand that other BWR-6s require protective clothing to be worn for containment entries.

- An in-house developed personnel radiation exposure management system (PREM) was accomplished from 1988-90 to achieve real-time dose tracking capability, support outage work planning and improve accuracy of dose estimation for repetitive work activities. This paper provides a comparison of dose estimation and actual results for refueling outages 2, 3, and 4 at CPS using the relational database methodology. Senior management at Clinton Power Station recognized early that the cobalt-60 source term was higher than many BWRs. Hence, strong support was given to the development of a computerized dose-tracking system after the first refueling outage to assist in improvements in work planning and dose management necessary to reduce dose whenever possible.

Five-Year Station Dose Projection

A five-year dose projection was developed by the ALARA staff in 1989 based on industry experience from five sister BWR-6s and estimation of recirculation piping dose build-up. Figure 1 shows the five-year dose projection made for the period 1990-95. Actual man-rem experience tracked very closely to actual annual man-rem for years 1990-93. 1994 annual man-rem was projected to be 120 man-rem in 1989 based on no refueling outages scheduled. Monthly, non-outage man-rem has been below 5 man-rem so far in 1994, hence, this projection appears to be good.

Work Planning and Dose-Management System Description

The in-house developed relational database system has provided automated radiation work permit (RWP) generation, access control for personnel entry to the radiologically controlled area (RCA) and real time tracking of work duration and dose accumulation. The system architecture was specifically designed to capture important radiation protection data in the field to achieve a nearly paperless dose management system. Each permanent radiological control point is provided a PREM terminal for access to RWPs and input of dose and job duration data.

During each outage, over one million separate ingress and egress transactions from the RCA and seven radiological control point zones within the RCA were recorded on the computer. As shown in Figure 2, the system is like a home intercom system. It tracks work duration and dose accumulation a literally 20-30 work locations during the peak of outage maintenance activities. Over 90,000 ALARA and dosimetry tracking reports were executed by the system during each outage. Areas of concentrated work activities could be monitored on a real time basis by radiation protection management. Areas of greatest attention at CPS were the drywell, refueling floor, and auxiliary building.

Outage Dose Trend Analysis

One of important reasons for developing a detailed dose history database on repetitive BWR work activities was for improvement in planning for future refueling outages. Detailed dose and duration data needed to be collected for three to five outages to achieve the types of trends information that could be used for the following reasons:

- outage staffing levels for both maintenance crews and support groups (i.e., radiation protection technicians, etc.)
- planning outage work by zone in the plant and schedule
- selection of outage work scope with accurate projection of impact on outage dose goals

- decision on amount of temporary shielding to be installed in specific plant area where work is concentrated

- daily monitoring of dose accumulation to provide greater management attention to work which is proceeding above projections either in dose or duration

- assurance of a good match between work scheduled and available staffing especially during peak outage work periods

- improvement in accuracy of annual man-rem goals and five-year man-rem projections based on identified operation and maintenance schedule.

This paper represents radiation protection's first attempt to use the relational database from three refueling outages to identify and verify trends in dose accumulation and job duration for improvements in the estimation of dose for work selected or deferred in future refueling outages. For purposes of the presentation at this conference, trends are presented based on their impact on the overall outage dose and outage schedule estimates. Of course, many other associations can be made. But, for brevity, impact on overall outage dose is the primary basis for the comparison.

It should be emphasized that it was important to collect data at the job step level with each maintenance work package covered by the RWP for this analysis to have the greatest accuracy.

Also, note that Refueling Outage-2 and 4 are similar in work scope since in-service inspections (ISI) in the bioshields where performed in even year outages. Hence, Refueling Outage-3 contained less high-dose work scope. PREM was not available for Refueling Outage-1. Finally, the comparison should be viewed has "reading the tea leaves" in terms of validity of trends identified until two more refueling outages are added to the database.

Trend Analysis Results

A total of 40 trend graphs have been developed in the analysis of Refueling Outages 2, 3, and 4. Slightly less than half of these graphs are described in this paper to highlight general conclusions reached from the relational database analysis. Discussion of the selected graphs is provided below:

Figure 3: RWPs - The number of RWPs generated to support refueling work dramatically reduced from 952 in RF-2 to 306 in RF-4. The number of RWPs issued represents the degree of radiological control exercised to manage work in the RCA. As plant personnel and contractors have more experience in successive outages, the number of RWPs issued can be reduced accordingly.

Figure 4: RWPs/Day - Comparison of RWPs issued per day allows for normalization of the variables of interest. The effect of outage length can be properly evaluated (e.g., RF-2 was for I 1 5 days, RF-3 was for 90 days, and RF-4 was for 7 5 days).

Figure 5: RWP/Day Chart - Comparison of RWPs issued per day based on estimated dose categories (less than 1 man-rem, greater than 1 man-rem and blanket RWPs) shows the Radiation Protection Manager that the number of RWPs greater than 1 man-rem remained remarkably the same over the three outage period. Reduction in RWPs issued occurred in the less than I man-rem category as more work was grouped under single RWPS.

Figure 6: Man-Rem - Man-rem charts are often the main comparison made by station management. Clinton Power Station experience shows that ISI work in the bioshield improved between RF-2 and RF-4

based on implementing lessons learned from the first ISI work. RF-3 outage man-rem will not have a meaningful comparison until RF-5 data is available.

Figure 7: Man-Rem % - Man-rem % shows that 75-85% of outage dose was received on RWPs estimated to have more than 1 man-rem for the work planned. This validates the ALARA planning rule of thumb for CPS that the bulk of outage dose is in greater than 1 man-rem RWPs. This focuses radiation protection management attention on reduction opportunities within this category of RWPs.

Figure 8: Man-Rem/RWP - As number of RWPs is reduced, the dose accumulated per RWP increases. This confirms the assumptions made regarding Figure 3.

Figure 9: Man-Rem/Day - This comparison shows that as outage duration is shortened, dose accumulated per day increases. The man-hours in the RCA increase in shorter outage and the figure properly reflects this conclusion.

Figure 10: Mrem/RWP-Hour - Mrem/RWP-Hour increases similar to increase observed in daily dose. More workers are engaged in work activities in the RCA.

Figure 11: Man-Hours - Man-hours for RF-3 and RF-4 are essentially the same.

Figure 12: Man-Hours/Day show that RF-2 and RF-3 are essentially the same. These two figures illustrate the importance of normalizing the data sets before comparisons are made in outages of varying lengths.

Figure 13: Man-Rem Delta - Man-rem delta figure shows the difficulty that can be encountered in under estimation of man-hours for work or dose rate estimates (RF-4).

Figure 14: Example of 1994 cumulative site exposure tracking graph which is distributed to plant personnel to inform them of actual site dose vs. annual goal.

Figure 15: Man-Hours/RWP - Man-hours/RWP illustrations the increase in man hours assigned to each RWP from RF-2 to RF-4.

Figure 16: Man-Rem Estimates - Man-rem estimates are of concern if they are less than actual man-rem experience (non-conservative).

Figure 17: Man-Hour Estimate - Man-hour estimates were found to be highest in RF-2. Man-hour estimates improved in subsequent outages. Demonstrates value of database use in future outages planning.

Figure 18: Dose Rate Estimates - Dose Rate Estimates are of particular concern when they are underestimated which has been a minor problem in all three outages. Estimates have been good for all three outages.

Figure 19. Man-Hours Delta - Man-hours delta was overestimated most in RF-2. However, RF-4 underestimated manf-hours which will need to be avoided in future outages.

Figure 20: Dose Rate Delta - Dose Rate Delta shows similarities between RF-2 and RF-4 where actuals were half of estimates.

CONCLUSION

Relational database comparison of similar work outages can be a valuable ALARA tool for outage planners and ALARA engineers in refining man-rem and duration estimates for future outages. Also, outage workers are provided realistic man-rem and duration goals to achieve based on previous outage crews performance. More importantly, it offers the Radiation Protection Manager key, plant-specific insights into daily dose accumulation management on a real-time basis. Proper management attention can be focused on problem areas to explore opportunities for dose reduction.

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PAPER 7B-3
DISCUSSION

Baum: I may have misunderstood, but I thought you said something increased by 30% due to new part 20. Could you repeat that, and explain how the part 20 change caused that?

Miller: I'd be very happy to. We have put people in "Pillsbury doughman" suits for 20 years in the United States. This significantly degrades their manual dexterity, and their visual acuity is blurred because of the respirators. We also put double plastic gloves on an individual who may only have to put a screw in a junction box in a very slight couple of DAC/hour airborne area but a 200-300 mR area of our drywell. So we have decided, in really understanding ICRP 26, that requires a TEDE ALARA evaluation for all jobs that may have an impact in terms of a respirator causing up to 20% of a slowing up of productivity of the actual work occurred. We take into account the total effective dose and make the decision on whether the individual wears a respirator or not. We had a 2-year effort of going through what we call a "cultural change" in America, because we have been telling everyone that internal dose is bad, you're going to take it home with you, it's not what we want you to do, and to respect all radiation -- even the smallest amount. In this case, we had to educate the process that a small amount of airborne, if it creates a higher total effective dose equivalent, is not in keeping with ICRP 26 recommendations. We actually had the use of respirators drop all across the United States to a factor of about 80-90% reduction after implementation of part 20 in January 1994. We implemented early in 1993, and we actually dropped our respirator use much earlier. We had sold our senior management on this, and when we saw that our doses were going up higher than our goal in our last outage this fall, our fourth refueling outage, they thought it might be just because people were wearing fewer respirators and that they were more comfortable being in low airborne areas and that they were staying in this low-dose, <1 man-rem RWPs. We simply punched up the button and a number came up, and we compared it to the two outages that were under the old part 20, where 6,000-7,000 respirators were worn. We interviewed several individuals, and, to a man, they all said they were still very uncomfortable going into low airborne areas. They would go in there and get out as quickly as they could. We used the intercom system to quickly check that, and sure enough, the same types of entries to airborne areas were much longer duration in the pre-part 20 period than in the post new part 20 area. That's what we were able to quickly prove, and we are going to continue to look at that because it is an important parameter. We do not require people to not wear a respirator. We think there is going to be a 2 to 3 year period, where if they are really uncomfortable and have concerns about airborne, then they will still be able to wear that respirator.

Lazo: The man-hours that you displayed, did you have any problems with unions in terms of collecting that data in terms of its being sensitive? It is really valuable information, but some people have some questions about that.

Miller: This is something we have learned and have been benefitting from, particularly by looking at other plants. We've had this in place for four years, and, in fact, it was just the opposite. When we installed it, we went out and talked to all of our 800 outage workers, and we talked to our own plant people. First of all, they liked it because their name was on the computer when they first came in the front door. It's like having your name on the mailbox. They feel a part of the team. We push team effort, as we all do, and even more so now that the company is organized by teams instead of by supervisor levels. Second, they like not having the clock nuisance business that they used to perform on manual doses, where one had to mark down 1723 I went in, and 1952 I went out, and subtract in one's head. That was now done by computer and they liked that. Third, instead of having

2,000 watches running around this plant, we liked the accuracy of the one computer watch. Fourth, we were better able to plan work, not only by schedule, but also plan your work and your outage, and subsequent third and fourth outages, by zoning the plant, so you don't stack the guy who is going to drop wrenches on top of two or three other work groups in the drywell. We were able to have a drywell coordinator who does nothing more than coordinate or police the activity in the drywell, and we would have all the linking of all the work in that very hot box done 4-5 months before the outage actually occurs because it's easy when 85% of your work is repetitive. We've now done that pump four times, and you know the duration. Finally, we have used the best part of ALARA motivation or incentive -- peer pressure. On going in to that particular crew, we post the 1992 performance duration dose. Here's the 1990s performance, and here's Sweden's performance, and France, and Germany, and Japan. Now guys, welcome. Do your job. And that's how we have been able to motivate in the true concept of optimization. You give your people the confidence and trust that they will perform well, and you provide them with the information. Until recently, it's been difficult to get that kind of dose information, even though 85% of our jobs are repetitive. We don't always do things well at Clinton or at many plants yet, and the biggest problem is that we still have a lot of construction trade people coming in as brand new carpenters off that guy's house right down the road who have never seen a nuclear plant. They are still getting 85% of our dose. We have a lot of effort in the mock-up and the training so there is more briefing of those individuals because your risk is as weak as the least experienced radiation worker in your RCA.

Viktorsson: I appreciated your emphasis on work management actions. But I have two other more specific questions. The first on the fuel failure. You said that you had the senior management commitment to shut down the plant if you exceeded some level of impurities in the reactor water. Could you tell us a little bit more about this -- what the levels are?

Miller: Yes, it was a year ago January that a sister plant in our region experienced very high off-gas levels in failed fuel. We recommended to the senior management to send people up there. They brought a report back and, in a nutshell, we showed that you have four times the problems we are having with cobalt if you start getting into alpha, transuranic monitoring, and the additional cost of decommissioning. So we have a level of 50,000 mCi/cc on the off-gas that we will actually make the decision to shut down within a week or less. We put together a task force who are still working and are monitoring any of our other plants, both in Spain and in Switzerland, and the ones domestically, so that we can grasp any of the lessons learned -- and there are a lot of them in radiation protection -- so that we are the most able to handle that challenge. The management commitment is at the senior vice president level. Unfortunately, my boss just resigned to go up north to handle six other BWRs at another utility, so we will be working on continuing to get the support of the next vice president in some critical areas as far as radiation protection sees it.

Viktorsson: Was this level derived based on sort of a cost-benefit analysis?

Miller: Yes, and that's why I'm bringing in the decommissioning costs. We had not seen before any real analysis of what you are dealing with and happened when this was occurring all the rule-making was going on in terms of decommissioning. So a part of the cost-benefit was not just the normal in additional dose and person-rem that we might see in that year that we have off-gas. Those numbers are readily available because we have three sister plants that have very high increase. We have found that our cobalt, because we have this flaky cobalt, actually goes up 3-5 times because uranium, if it starts circulating and a boiler kicks the cobalt off the fuel. We've seen spikes in some of our sister plants, so yes, we did all that in terms of showing the man-rem and in finally the decommissioning costs in terms of how much most of us have of 100,000,000 to 150,000,000 in our long-term reserve for decommissioning. How much more in terms of cost we might have to put in there if we start seeing transuranics in piping in radwaste system sporation pool, etc.

Viktorsson: The second question related to in-service inspection. Could you tell us the percentage of the total collective dose that relates to annual in-service inspections. I mean, ultrasonics, radiography, and things like that? And do you have any special automatic equipment for this.

Miller: Yes, the in-service inspection dose is normally around 80-85 man-rem in that RF2 and RF4. What we have done, is that the first time entered our bioshield to do that work, we hydrolyzed extensively, and that dropped the nozzles on the N2s and N4s on the boilers down from 30 to 50 R/hr after that second outage down to 300 mR/h as the highest nozzle. So hydrolyzing from the bridge was very effective. Still, we brought in 35 ISI people, but still the work was very high person-rem. Second, we had a lubricant problem which we have now resolved in terms of actually putting the equipment on in that first outage, and third, we had significant problems in building scaffolding and the time that it took. We did not have permanent work platforms. We needed them in that area and we had put that in for our following outage. Those three improvements, along with a whole list that has come out of the team debrief from our last outage, are hoping to reduce that dose and there is also a technical approach that some boilers are using which is induce stress relief, and that can allow you to go up to 10-year cycles, instead of even-year cycles.

Figure 1. Five Year Man Rem Projection

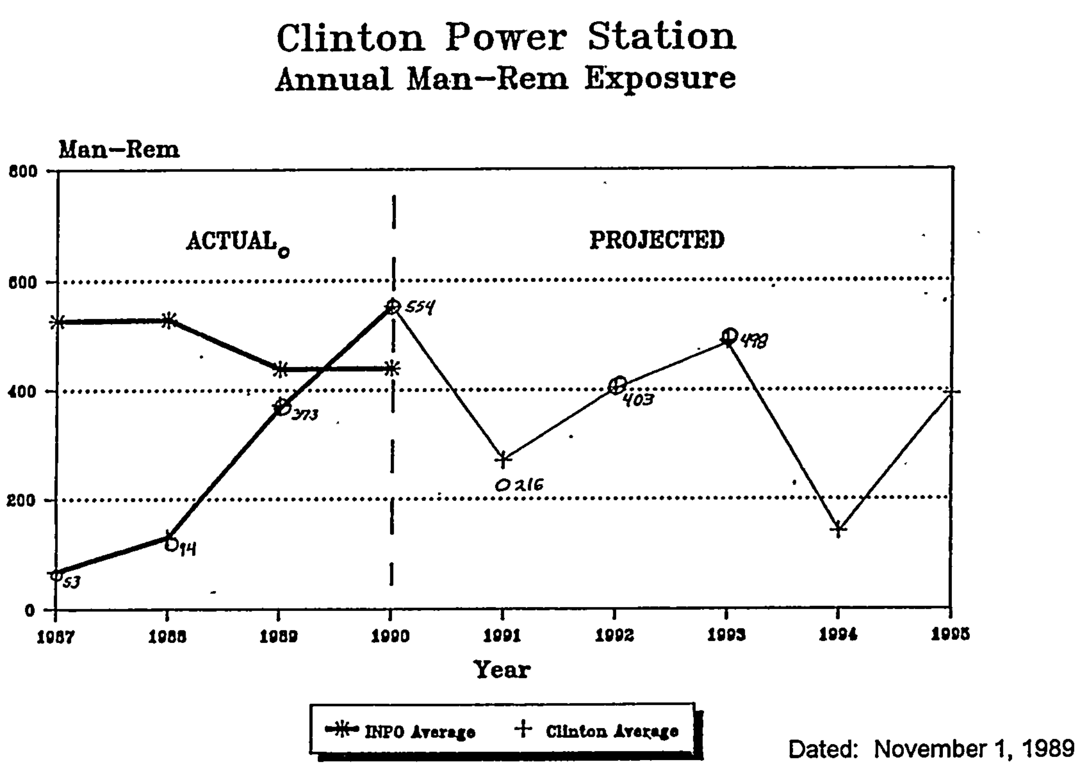


Figure 2. In-Plant Radiation Protection Control Point Computerized Dose Tracking

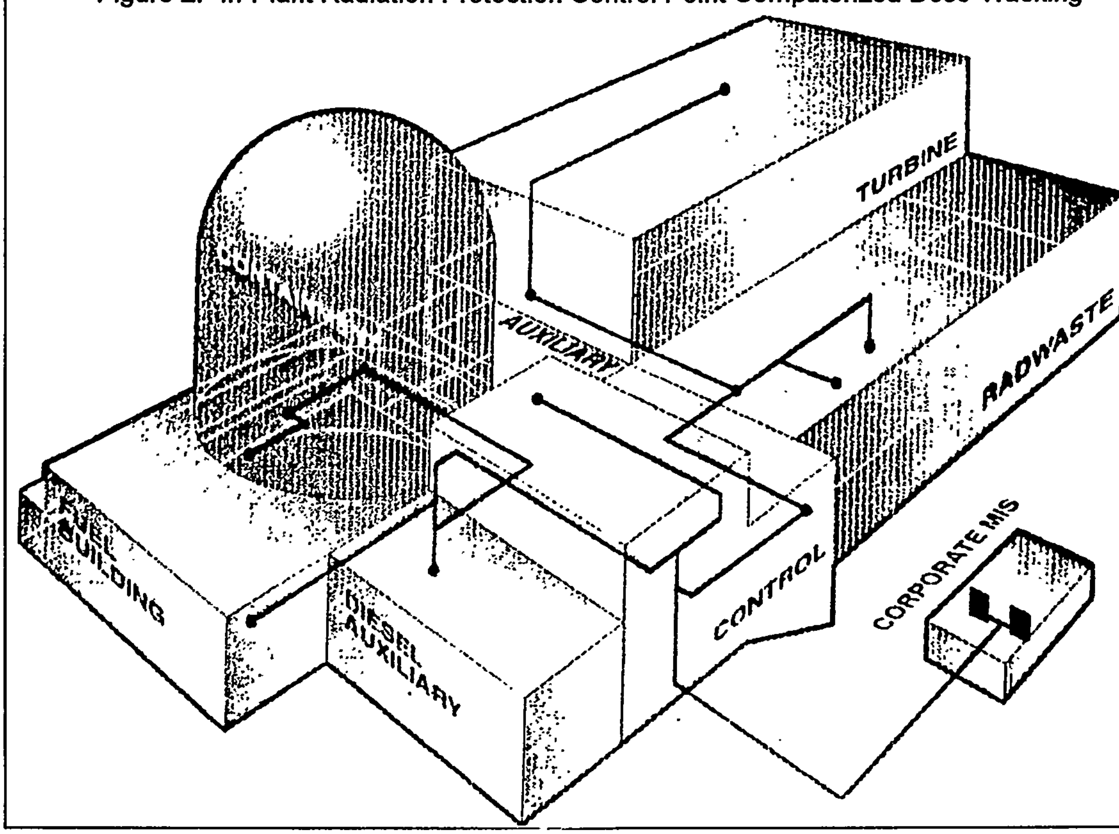


Figure 3.

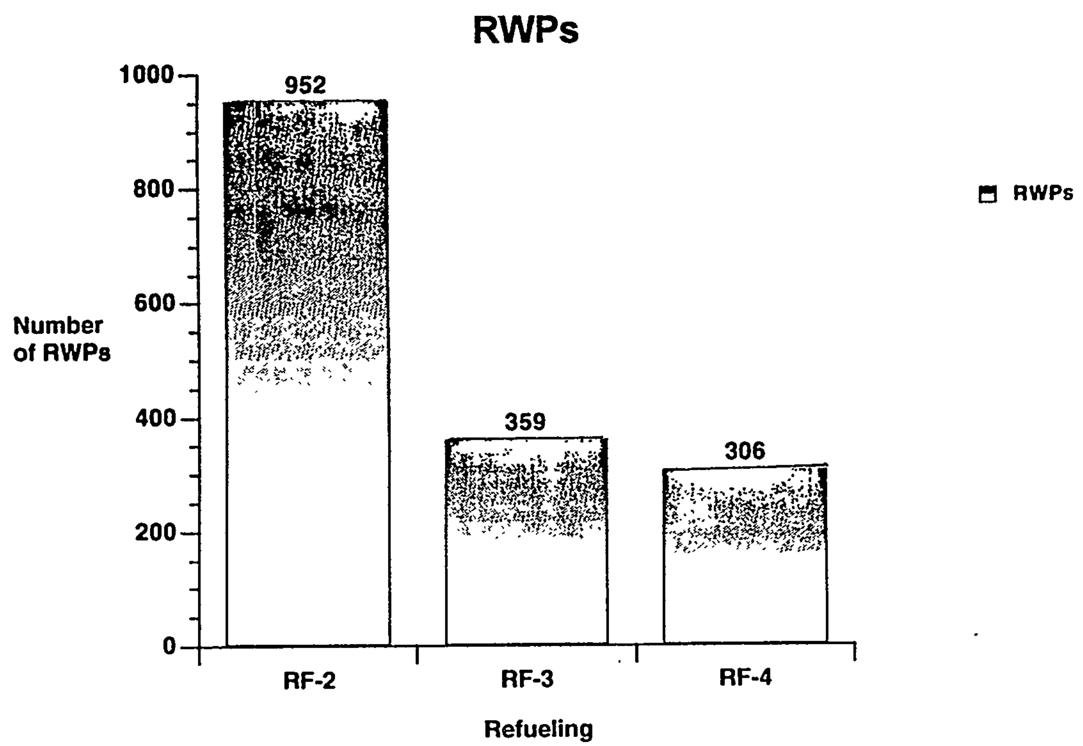


Figure 4

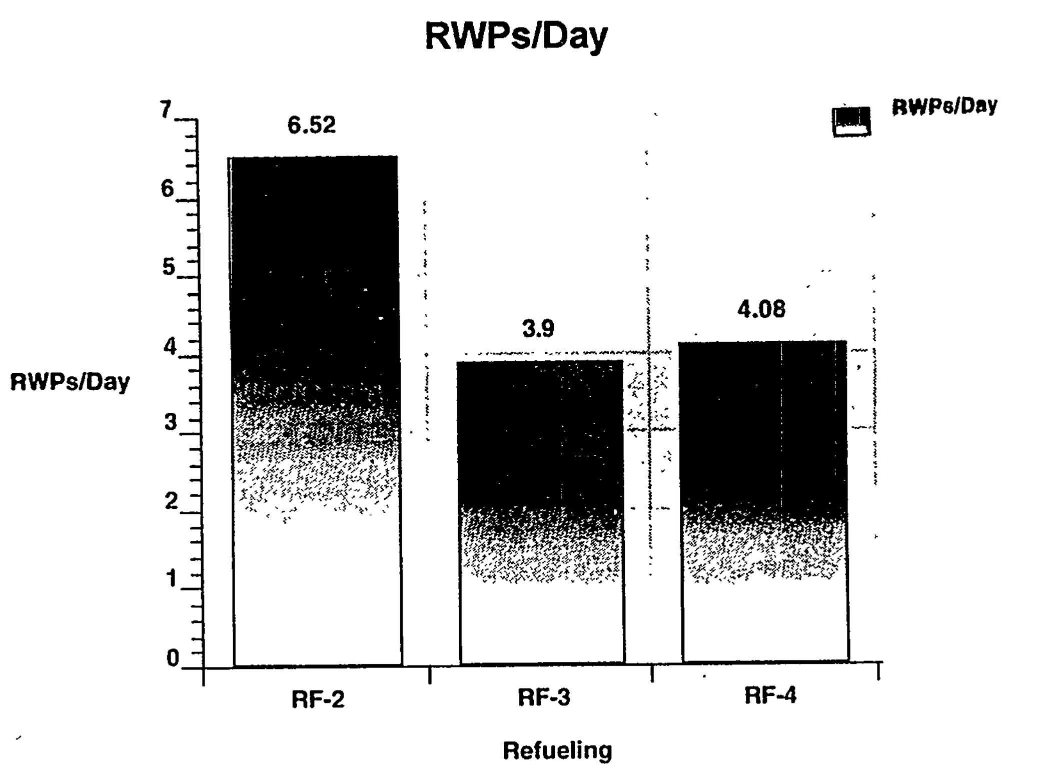


Figure 5

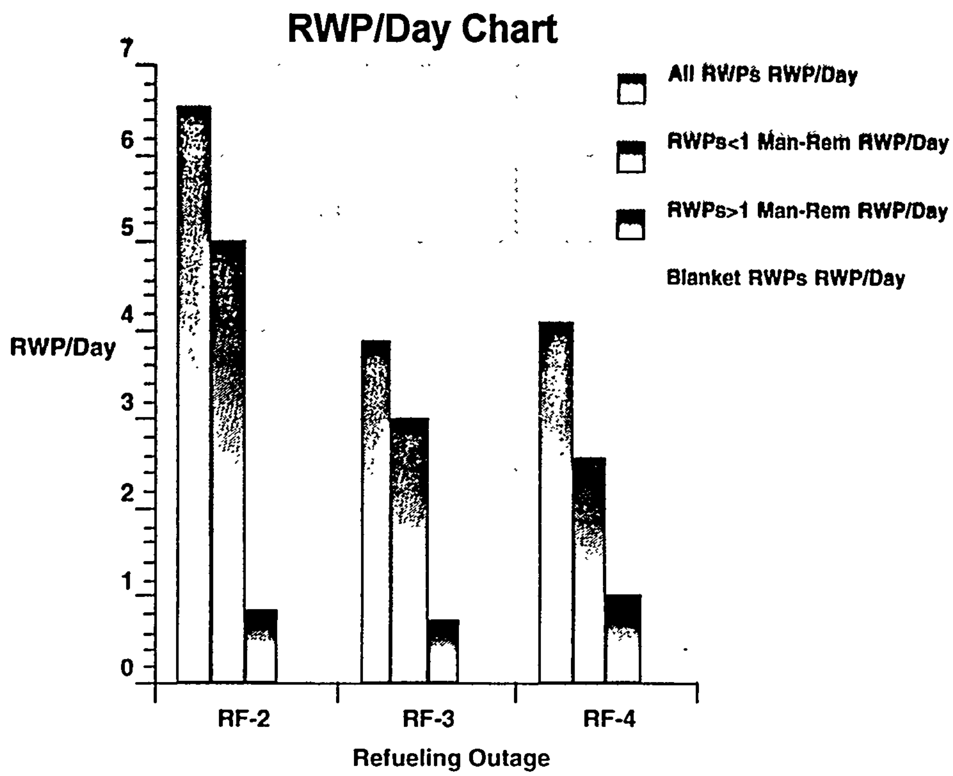


Figure 6

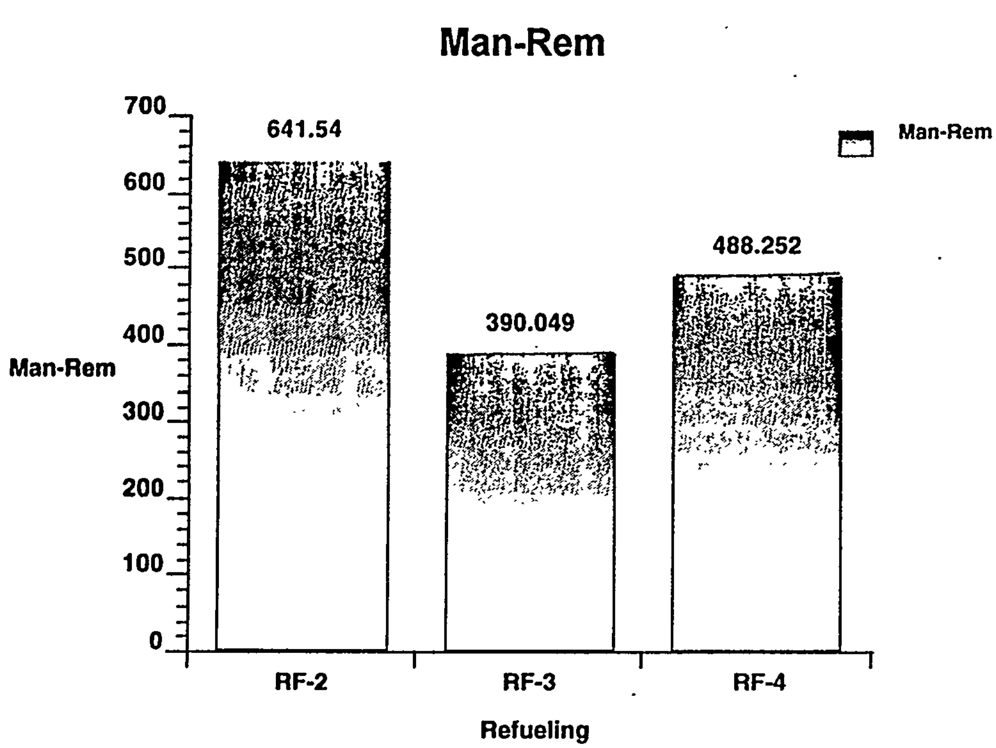


Figure 7

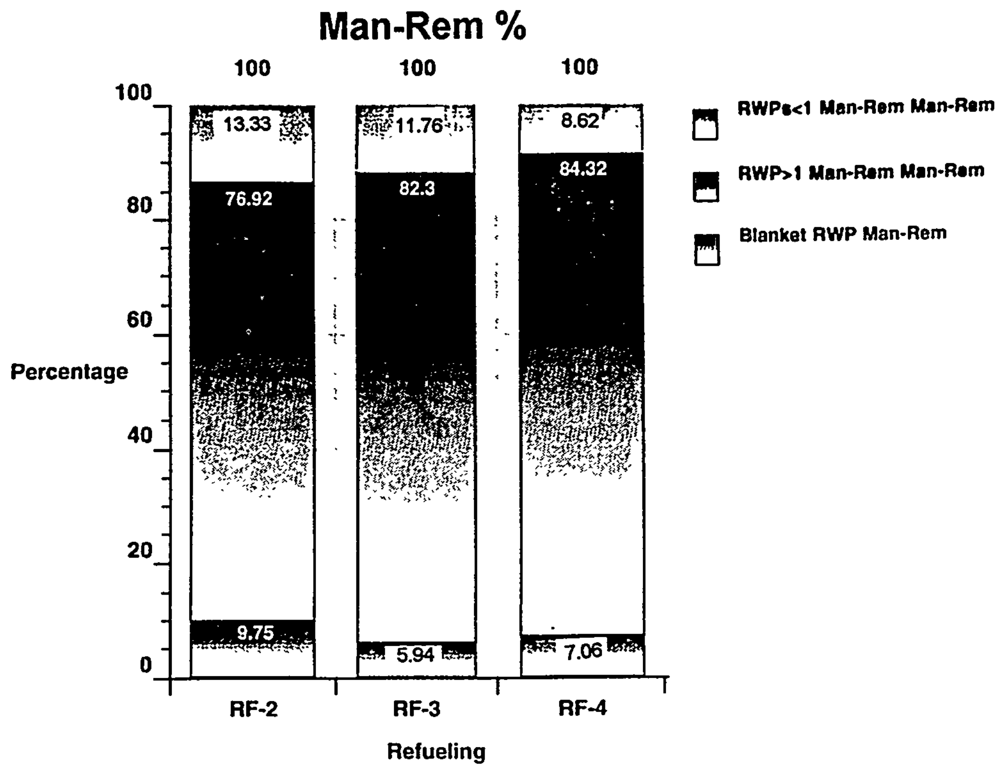


Figure 8

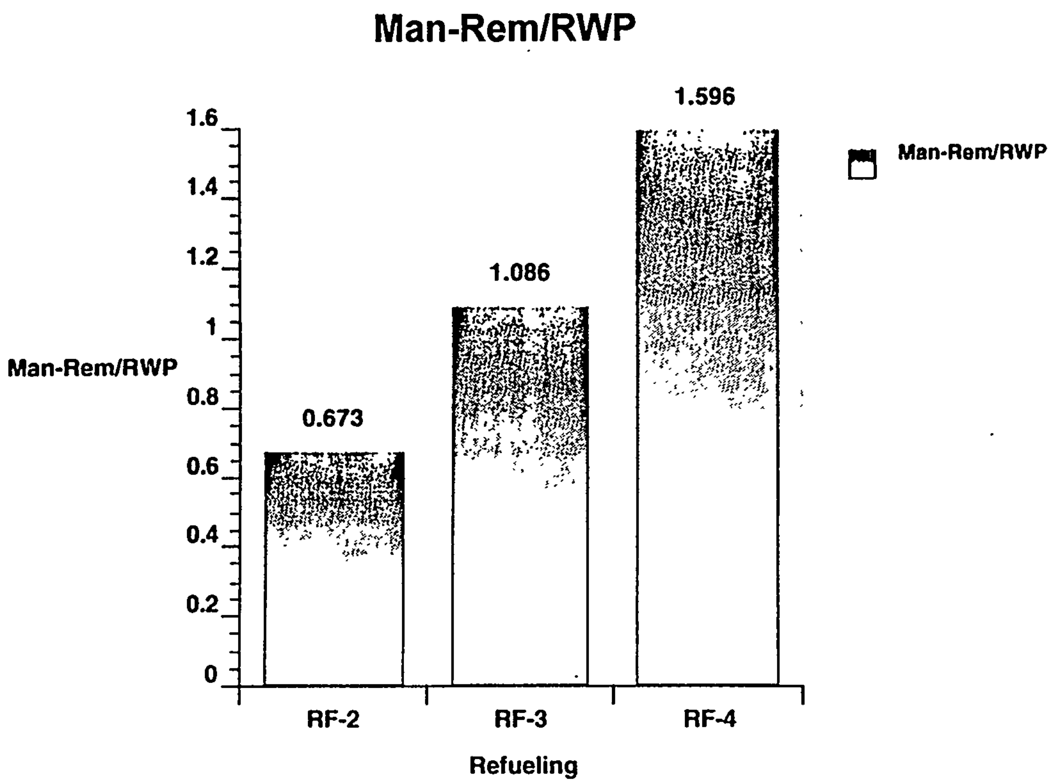


Figure 9

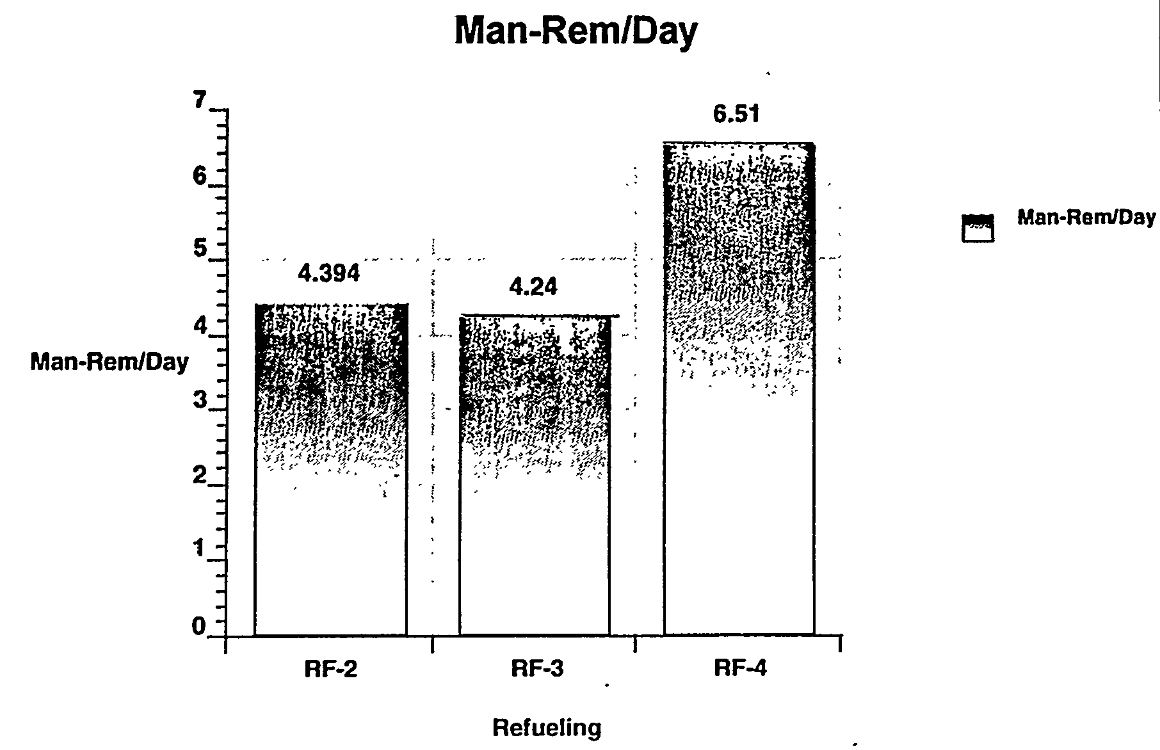


Figure 10

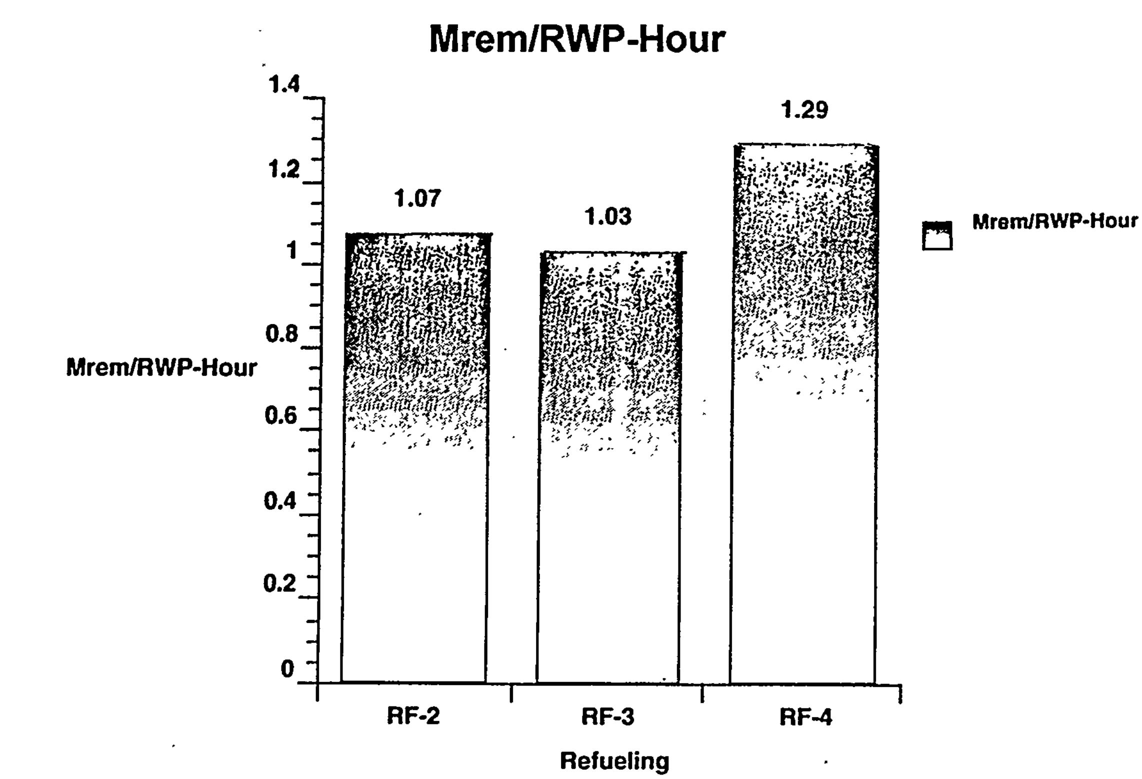
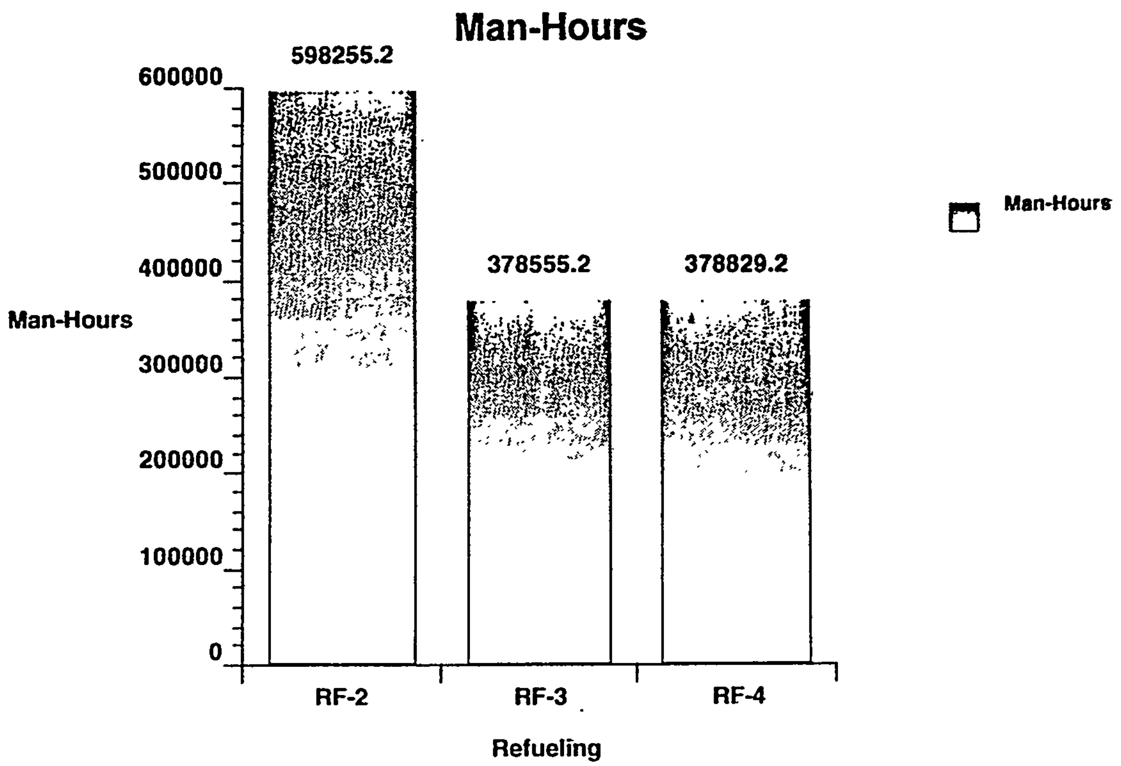


Figure 11



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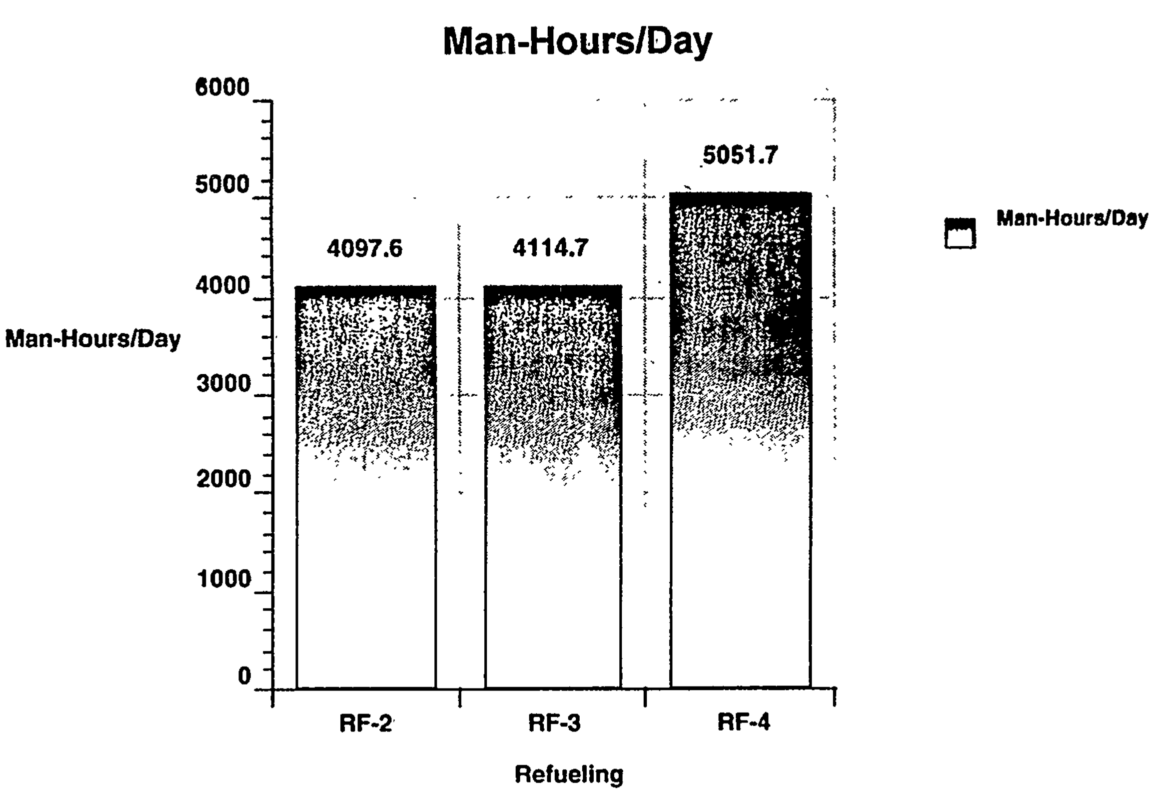

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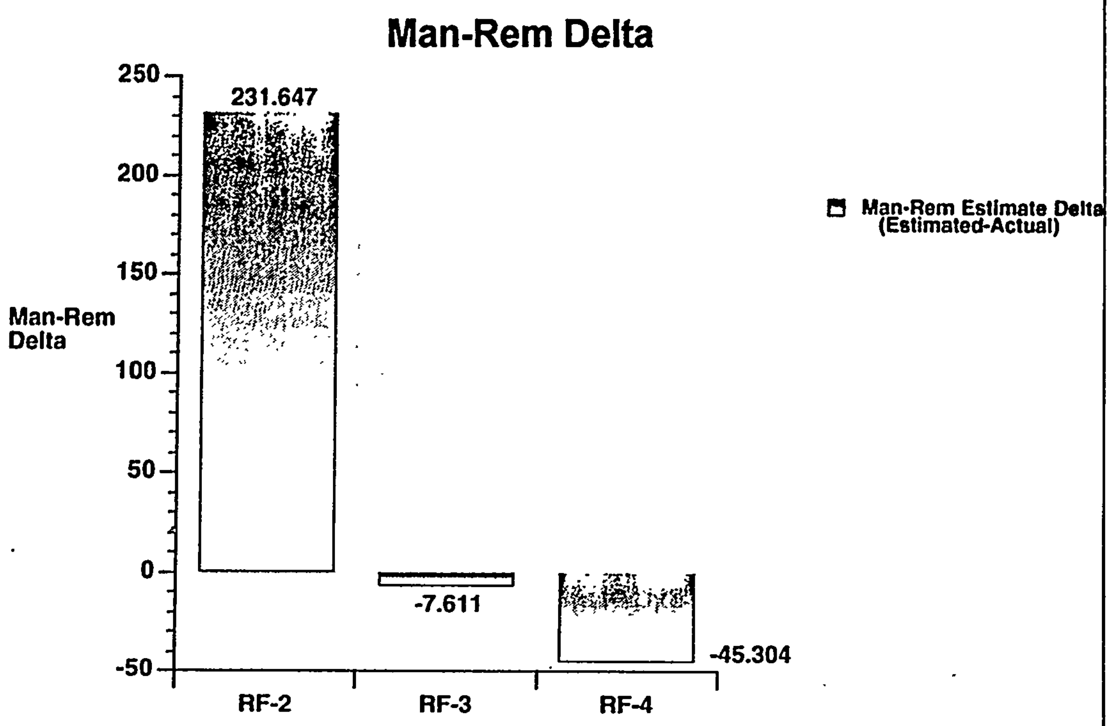
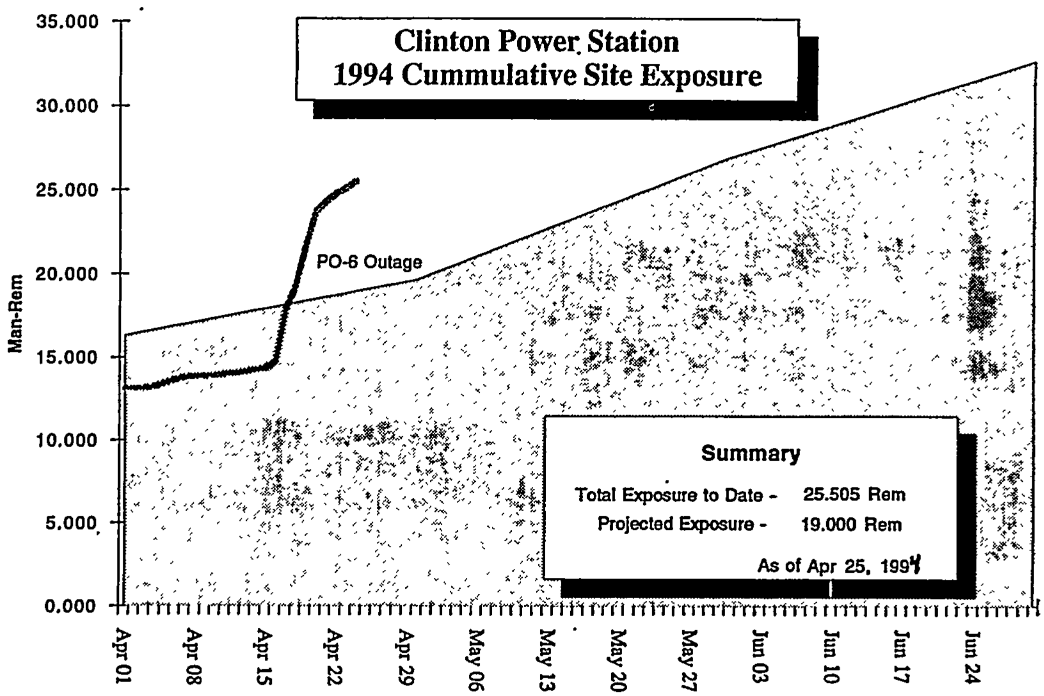


Figure 14



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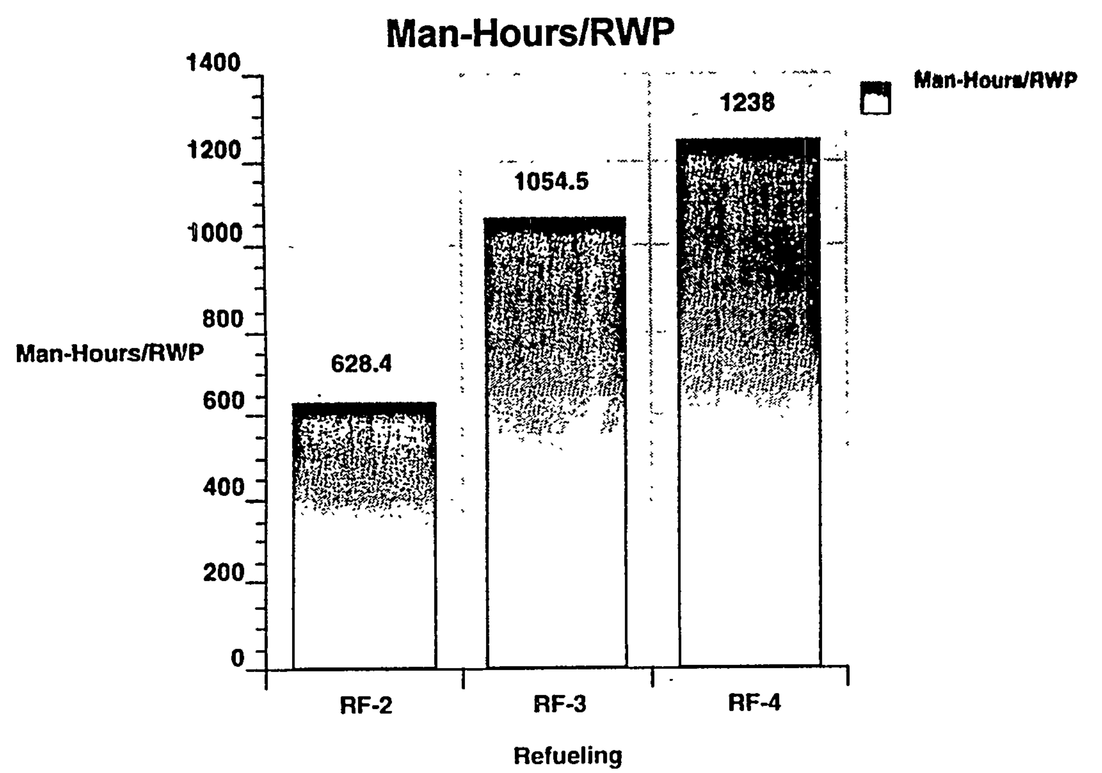


Figure 16

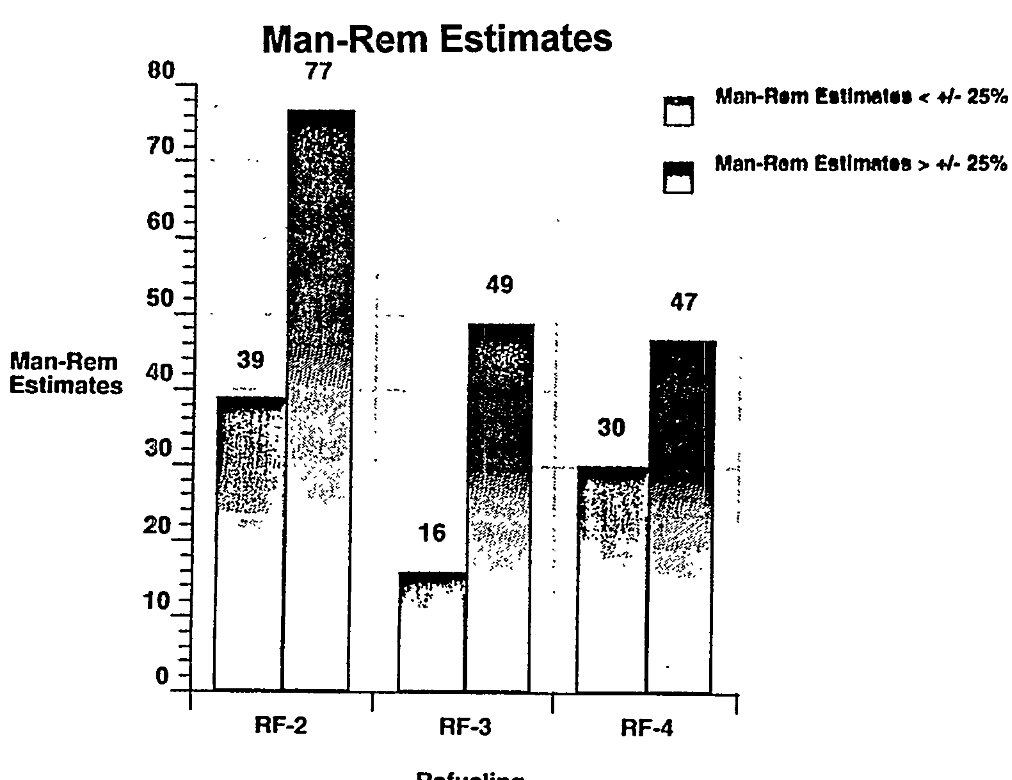


Figure 17

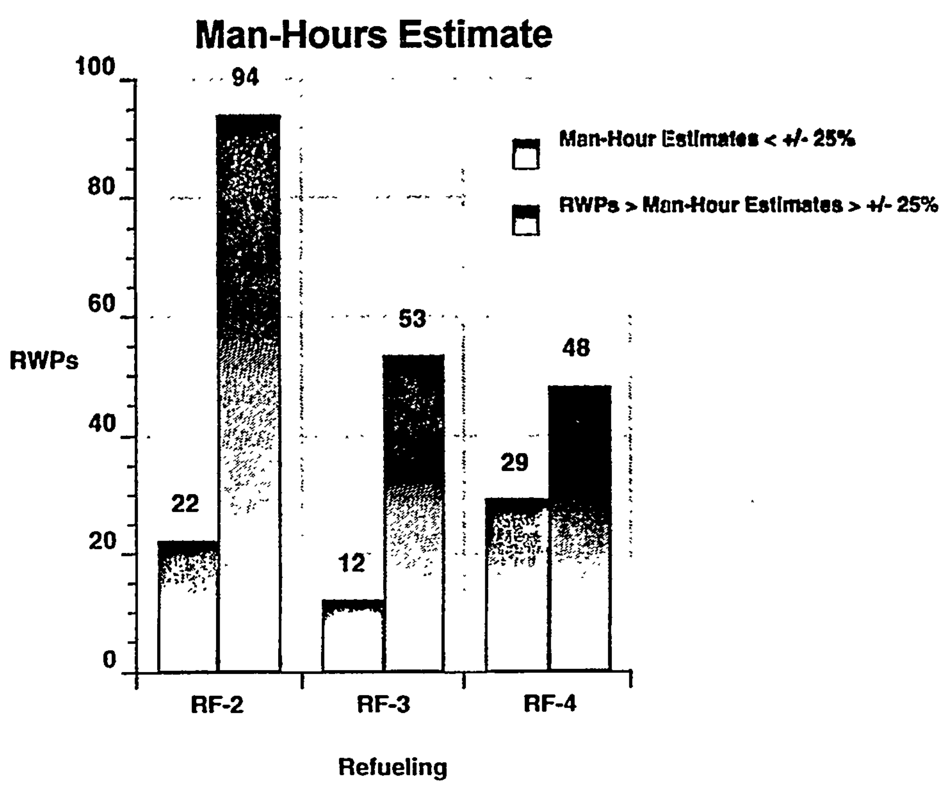


Figure 18

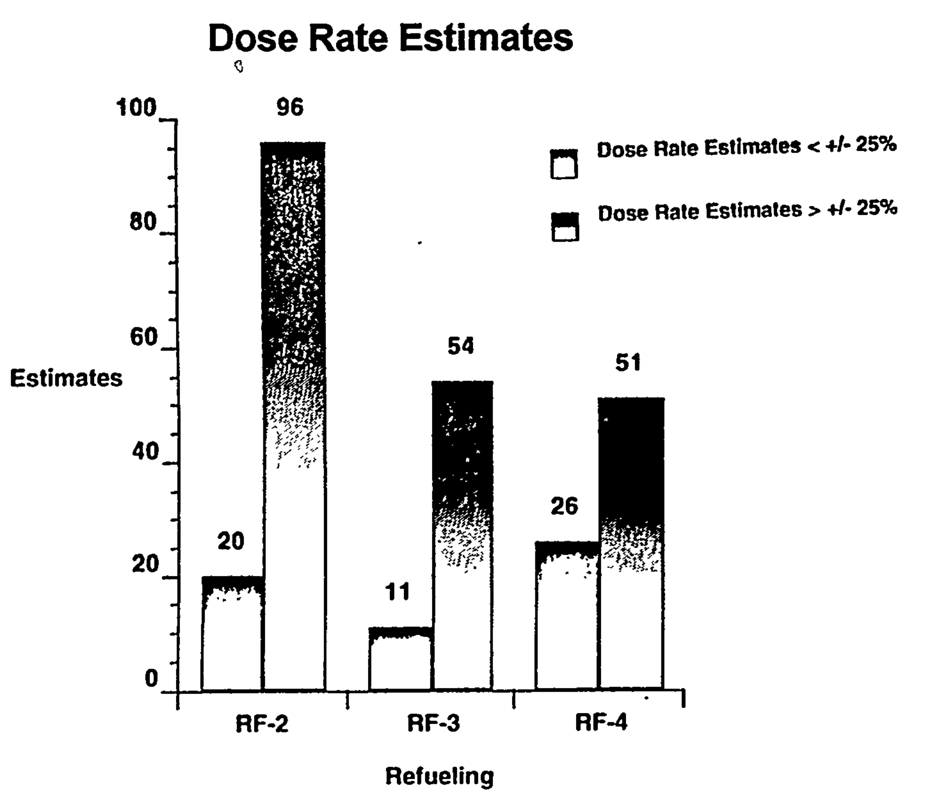


Figure 19

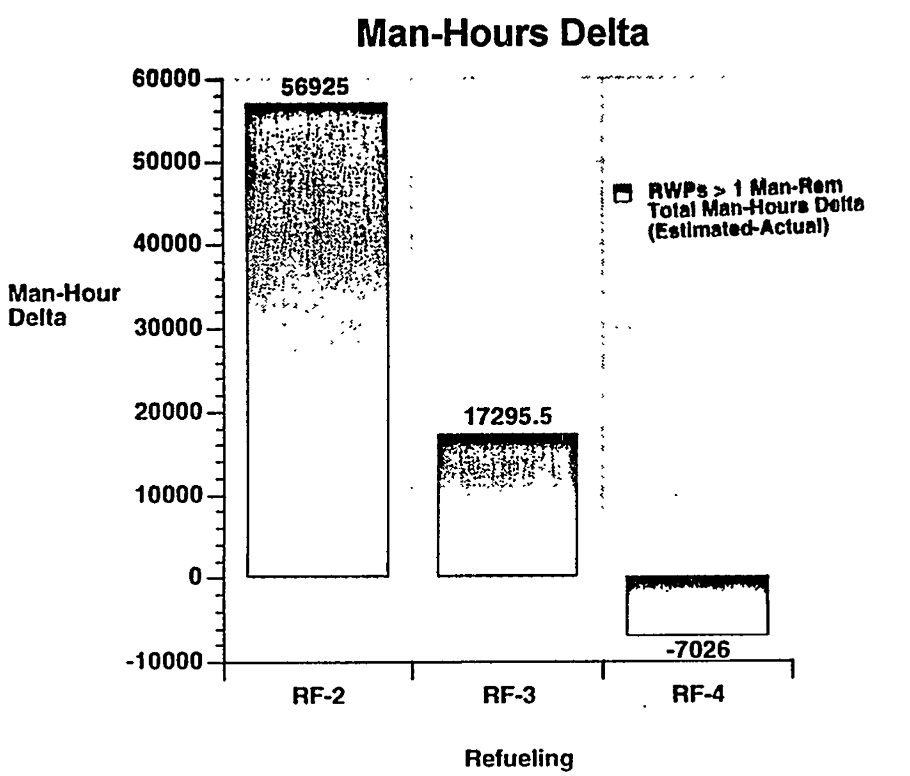
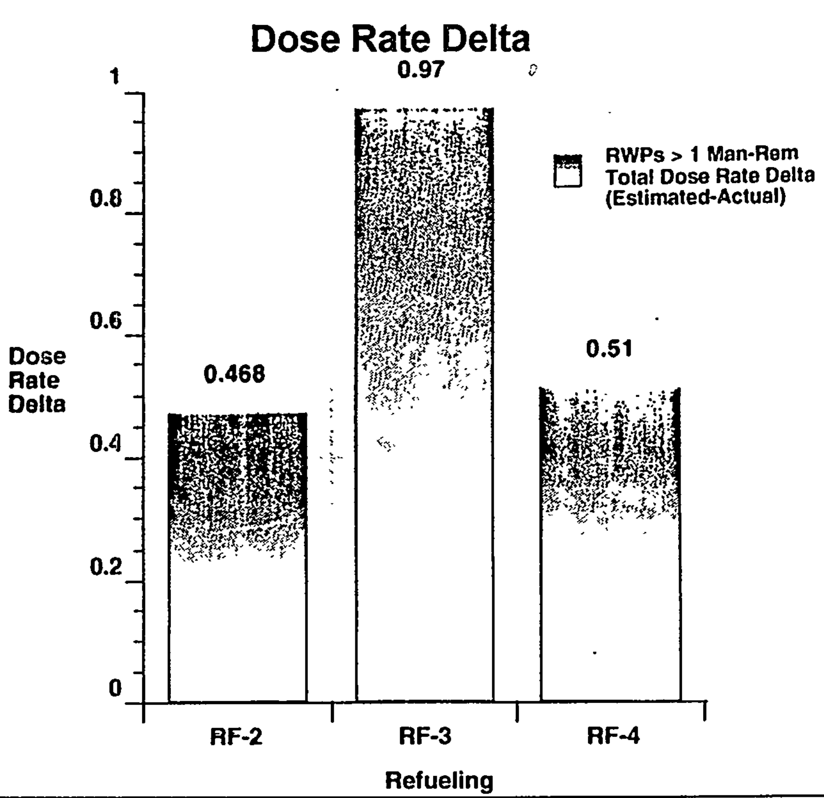


Figure 20



WATER CHEMISTRY CONTROL AND DECONTAMINATION EXPERIENCE WITH TEPCO BWR'S AND THE MEASURES PLANNED FOR THE FUTURE

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The Tokyo Electric Power Company

ABSTRACT

The new TEPCO BWR's are capable of having the occupational radiation exposure controlled successfully at a low level by selecting low cobalt steel, using corrosion-resistant steel, employing dual condensate polishing systems, and controlling Ni/Fe ratio during operation. The occupational radiation exposure of the old BWR's, on the other hand, remains high though reduced substantially through the use of low cobalt replacement steel and the partial addition of a filter in the condensate polishing system. Currently under review is the overall decontamination procedure for the old BWR's to find out the measures needed to reduce the amount of crud that is and has been carried over into the nuclear reactor. The current status of decontamination is reported below.

INTRODUCTION

The new BWR's (Fukushima Daini & Kashiwazaki-Kariwa) of TEPCO have introduced radiation exposure lowering measures in terms of steel and equipment, including the use of low cobalt steel to reduce cobalt radioactivity, selection of corrosion resistant steel, and a dual condensate polishing system to reduce the feedwater crud. Chemical control of feedwater has also been employed to control Ni/Fe ratio. The above control measures, together with automation of the inspection operation, have contributed as a whole to keeping the occupational radiation exposure at a low level since the reactor start-up.

For the old BWR's (Fukushima Daiichi), various control measures have been taken to reduce the occupational radiation exposure using the experience gained from the new BWR's. With respect to steel and equipment, low cobalt steel has been used and a hollow fiber filter (HFF) partially added to the condensate polishing system. Though the occupational radiation exposure could be effectively reduced by these measures, it still remains higher when compared with the new BWR's.

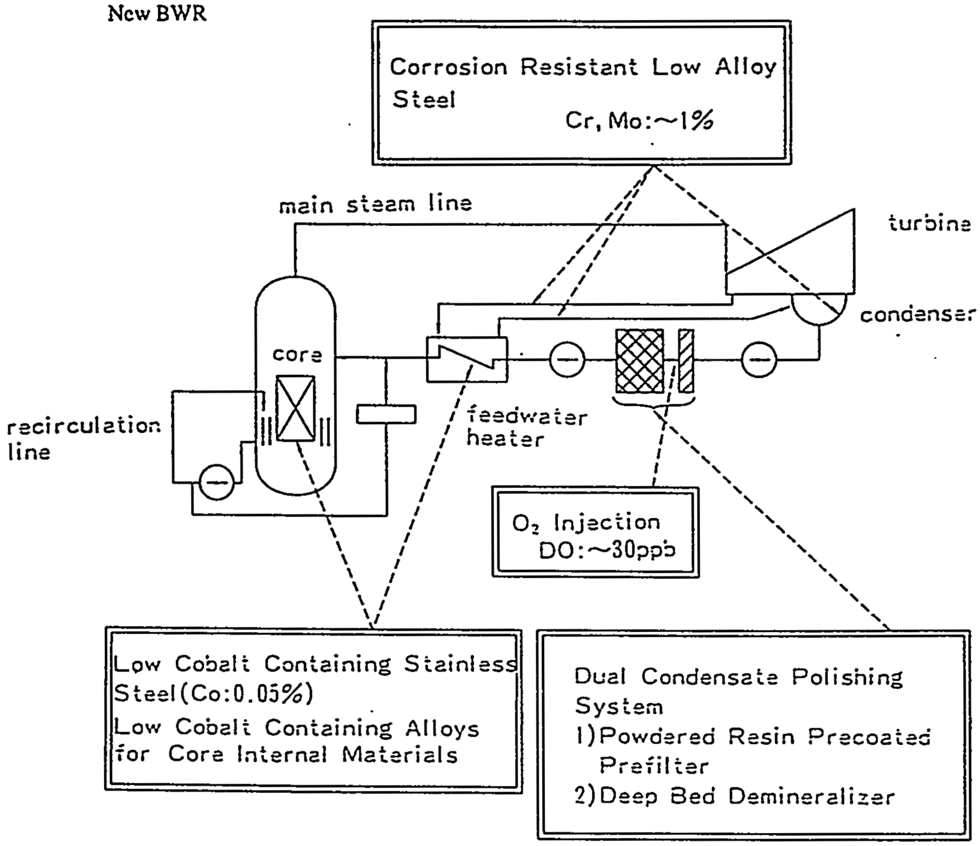
PLANT DESIGN OF TEPCO's BWR's

TEPCO's plant design can be classified into two categories according to the water treatment method of the feedwater and condensate systems.

One is the new BWR's, in which the condensate polishing system is of a dual construction, with a precoated type CF or HFF installed upstream of CD to cut down the crud in the feedwater. Moreover, the new plant employs low cobalt and corrosion resistant steel in the materials and equipment to be used, lowering the Co radioactivity.

The other includes the old plants, which were originally equipped with an independent CD as a condensate polishing system. Certain plants are now either provided with HFF for partial polishing of the condensate or employ the low-Co steel for replacement. (Figure 1)

New BWR



Old BWR

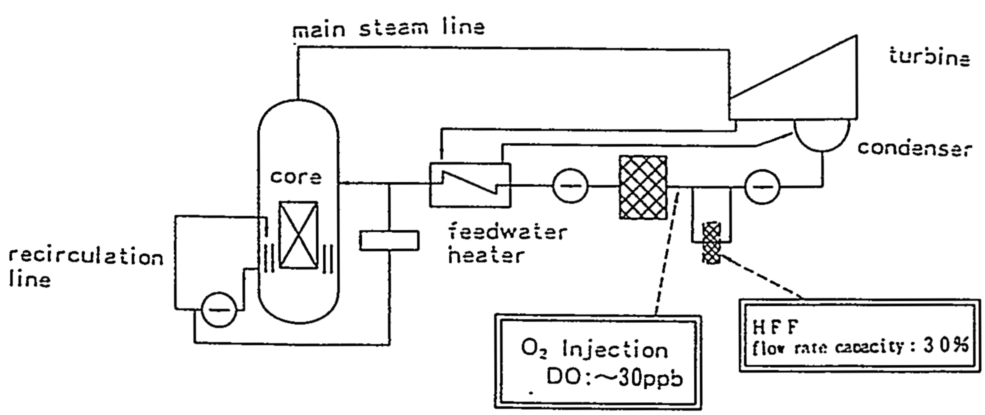


Figure 1. New vs Old BWR's

PRESENT STATUS OF RADIATION EXPOSURE DURING MAINTENANCE OUTAGE

Figure 2 shows the occupational radiation exposure during maintenance outage. The radiation exposure shown here corresponds to that during the standard maintenance works excluding additional works. The exposure level shows an annual downward trend in old plants and has been recently estimated to be around 2 - 4 man.Sv, which still remains higher than the 0.5 - 3 man.Sv for new plants.

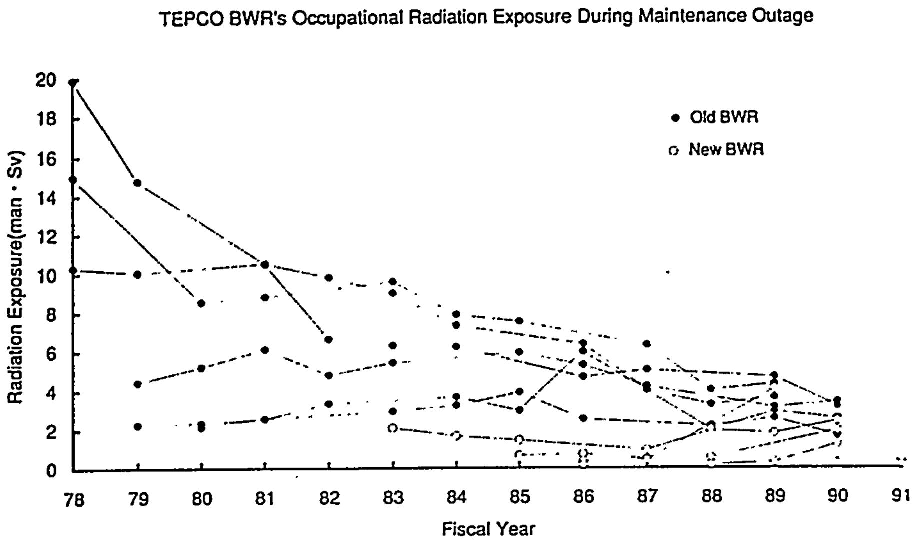
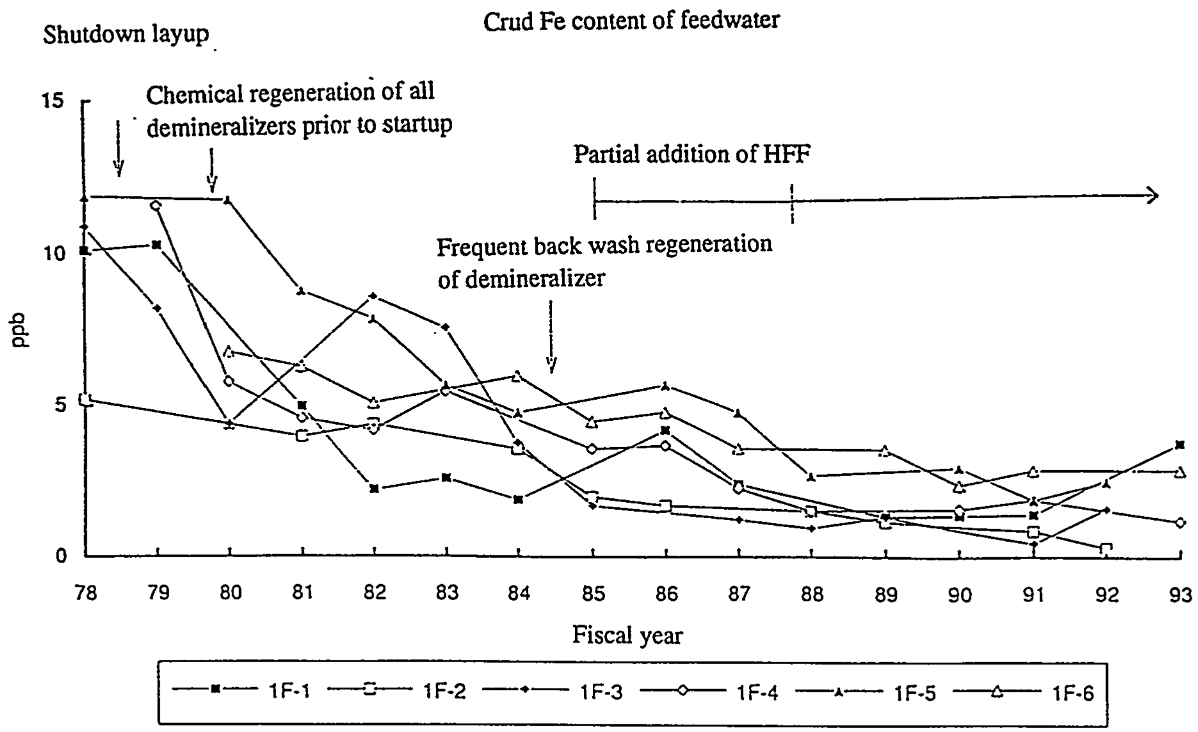


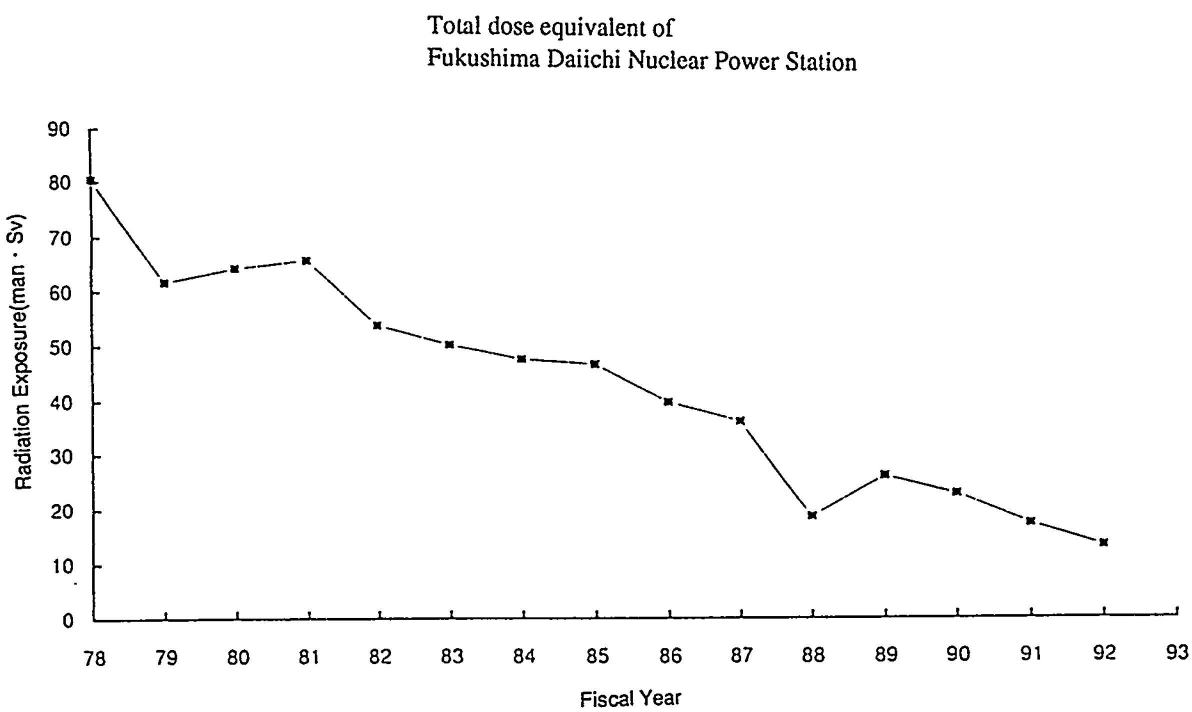
Figure 2

Such an annual decrease in radiation exposure at old plants may be attributed partially to a decrease in the radioactivity caused by crud through reduction of the crud content in feedwater. The crud content was cut down by increasing the back wash frequency of CD to improve its performance in holding down the Fe content or by adding HFF upstream of CD. Decrease in the crud content in feedwater in turn has caused a gradual decrease in the radiation exposure. (Figure 3)

A factor which still keeps the radiation exposure on a higher level in old plants than in new plants, is known to be the higher contribution of crud radioactivity (Figure 4). This has become evident from the comparison between new and old plants in terms of the dose rate of PLR piping by ion and crud.



Shutdown layup
Crud Fe content of feedwater
Chemical regeneration of all demineralizers prior to startup
Partial addition of HFF
Frequent back wash regeneration of demineralizer
15
10
5
0
ppb
78
79
80
81
82
83
84
85
86
87
88
89
90
91
92
93
Fiscal year
1F-1
1F-2
1F-3
1F-4
1F-5
1F-6



Total dose equivalent of
Fukushima Daiichi Nuclear Power Station
90
80
70
60
50
40
30
20
10
0
Radiation Exposure(man · Sv)
78
79
80
81
82
83
84
85
86
87
88
89
90
91
92
93
Fiscal Year

Figure 3

Comparison of dose rate of
PLR piping surface

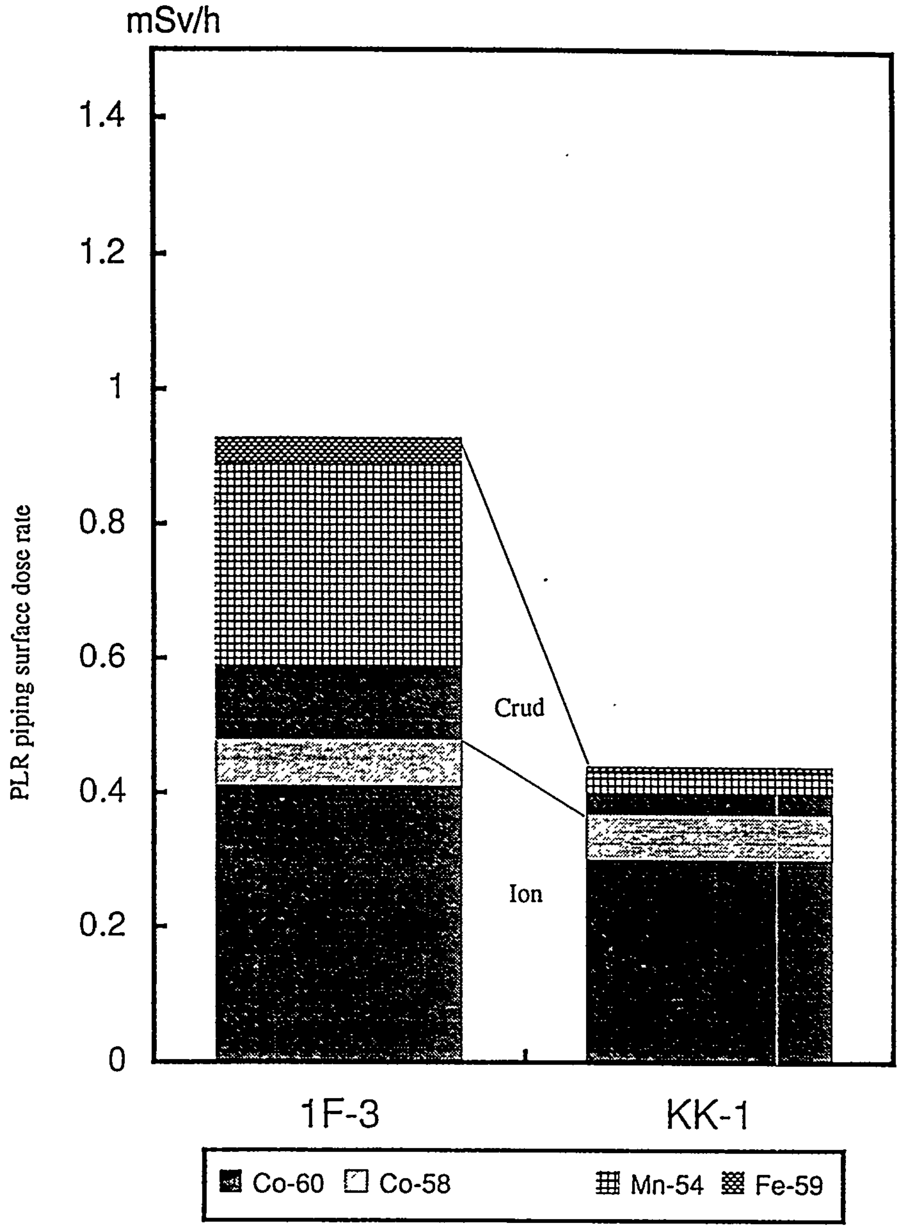


Figure 4

FUTURE RADIATION EXPOSURE CONTROL MEASURES

The radiation exposure control measures for old plants have conventionally included roughly two types. One is to promote a decrease in the crud content of feedwater and the other is to suppress the radioactivity of crud already accumulated in the reactor and systems.

Firstly, reduction of the crud content of feedwater in old plants is discussed. Two methods can be considered. One is to increase the treatment capacity of HFF by undertaking the partial polishing of the condensate while the other is to provide the CD resin with the ability to hold down the Fe content. Introduced below is the CD resin developed recently which has a greater capacity to hold down the Fe content than conventional resins.

Conventionally, in old plants, the crud content of feedwater was controlled by providing the CD resin with the ability to suppress the Fe content through the aging effect. This aging effect is caused when the CD resin is subjected to chemical regeneration for use over an extended period of time while maintaining the deionizing ability. Namely, the ability of the CD resin to suppress the Fe content has been maintained on the basis of the expected enhancement of the ability diffused as "Matrix-diffused crud", which in turn was caused by an increase of specific surface area and water retention capacity during use for a long period (Figure 5). Some resins have been used for as long as 15 years.

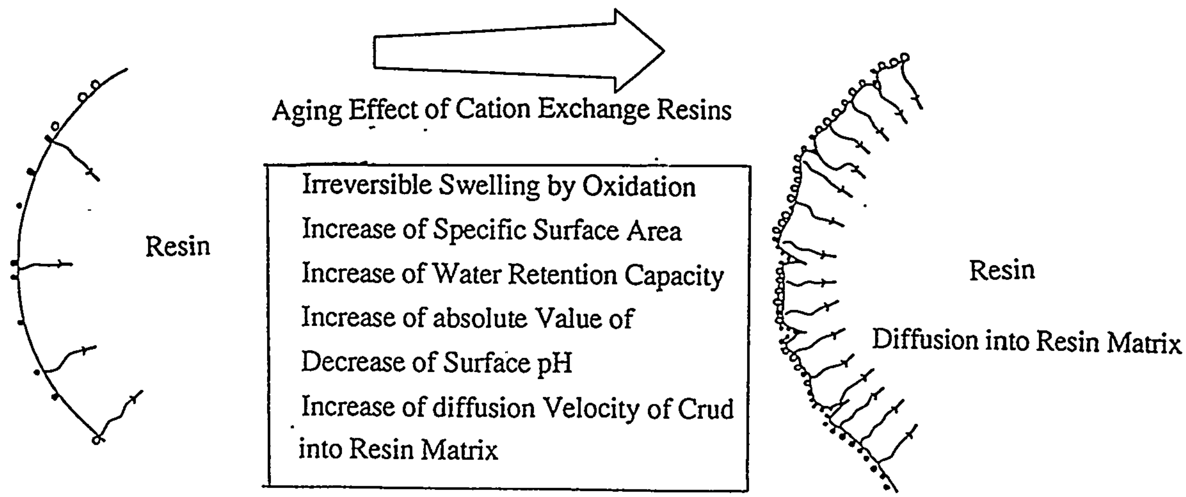
On the other hand, the resin has a general tendency to suffer a larger amount of TOC elution with increasing water retention capacity. In old plants, therefore, the phenomenon of rising reactor water conductivity tends to be observed at the start of reactor operation, which is attributable to elution of TOC from the CD resin. (The reactor water conductivity is normally around 0.3 - 0.4 μS/cm maximum, but amounts to around 0.8 μS/cm in certain old plants.) When the CD resin is replaced by the new one to suppress an increase in the reactor water conductivity at startup, the resin loses its aging effect, resulting in an increase in the crud content of feedwater and finally causing an increase in the crud radioactivity.

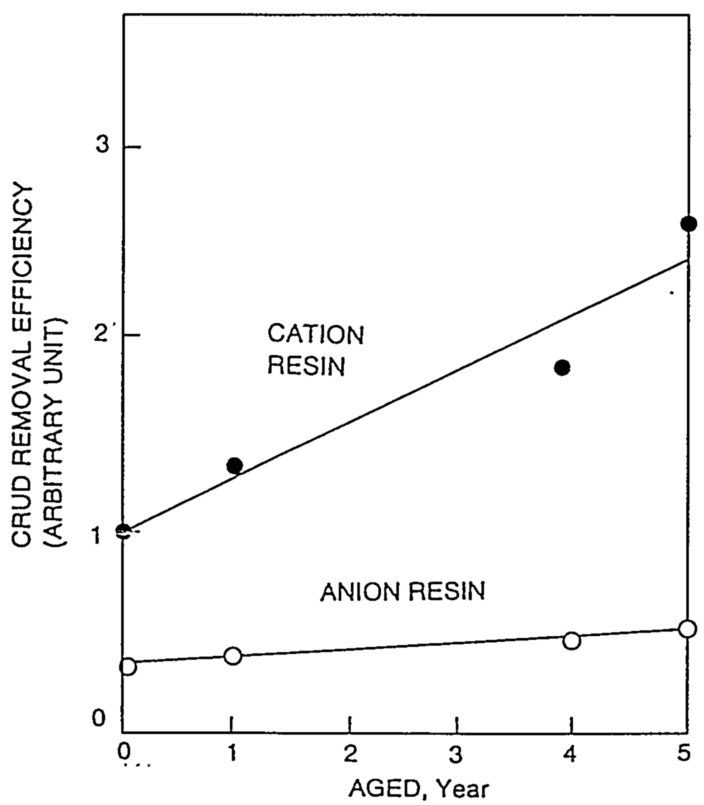
As a countermeasure, we have developed a resin, which shows a sharp water retention capacity distribution curve though the average water retention capacity is similar to that of the conventional resins. We have also conducted a water flow test by incorporating the newly developed resin in one (CD1) of eight actual CD's. The result shows that the newly developed resin demonstrates the ability to suppress the Fe content, approximately equivalent to that of the resin provided with the aging effect gained through 12 years of use. The result also proves the acceptability of the new resin in that it is free from any remarkable increase in the reactor water conductivity even during in-service use of the test demineralizer. (Figure 7)

We plan to introduce the above newly developed resin as a means to control the crud content of feedwater in the future.

As regards lowering of the radioactivity of the crud accumulated already in the reactor and systems, the mechanical or chemical decontamination shown in Figure 8 may be considered. These decontamination measures will be put into practice after an in-depth study on the effect of suppressing radiation exposure and the countermeasures appropriate to preventing re-contamination.

Ability absorbed as "Surfase-Adsorbed Crud"
Ability Diffused as "Mtrix-Diffused Crud"





Improvement of crud removal efficiency
by aging of condensate demineralizer resins

Figure 5

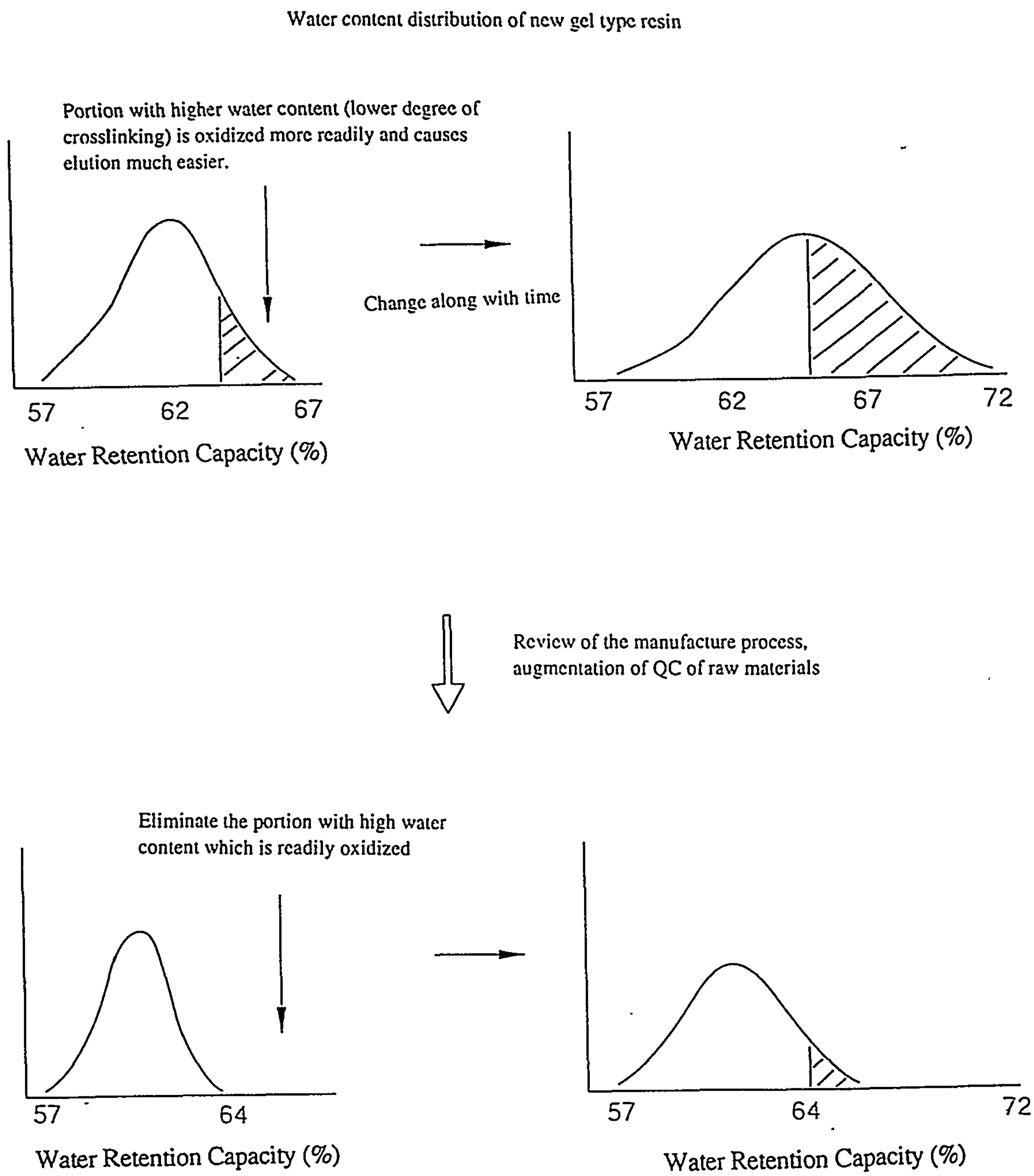
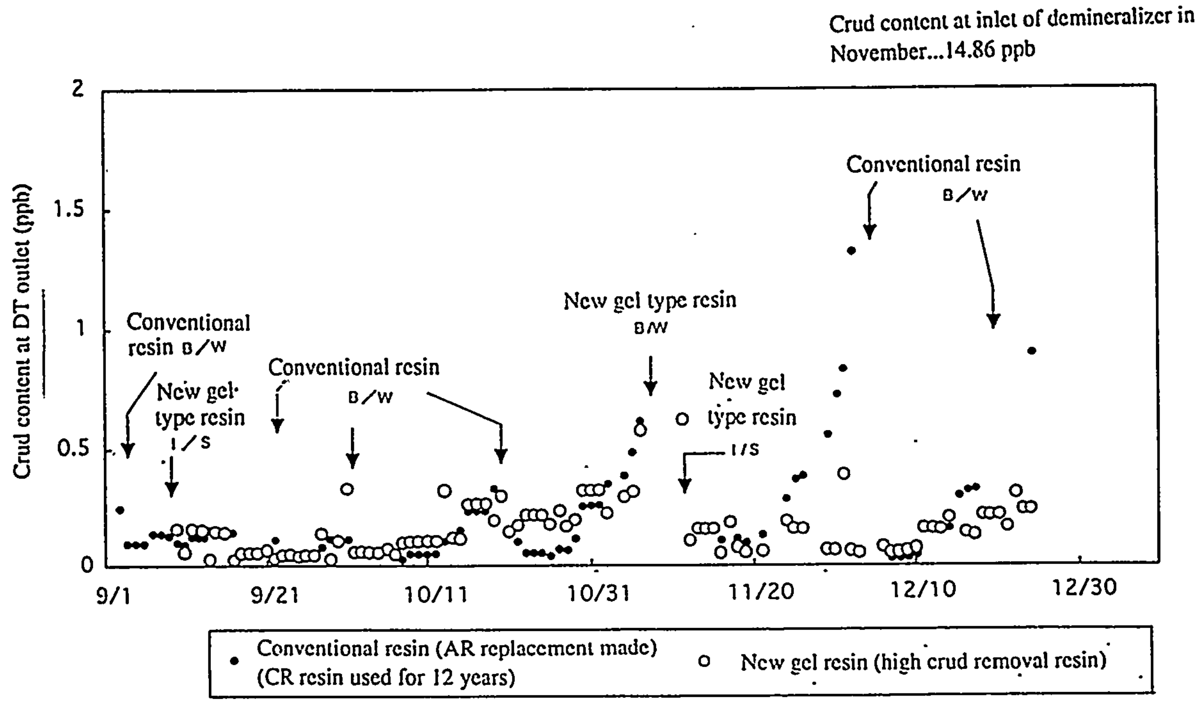
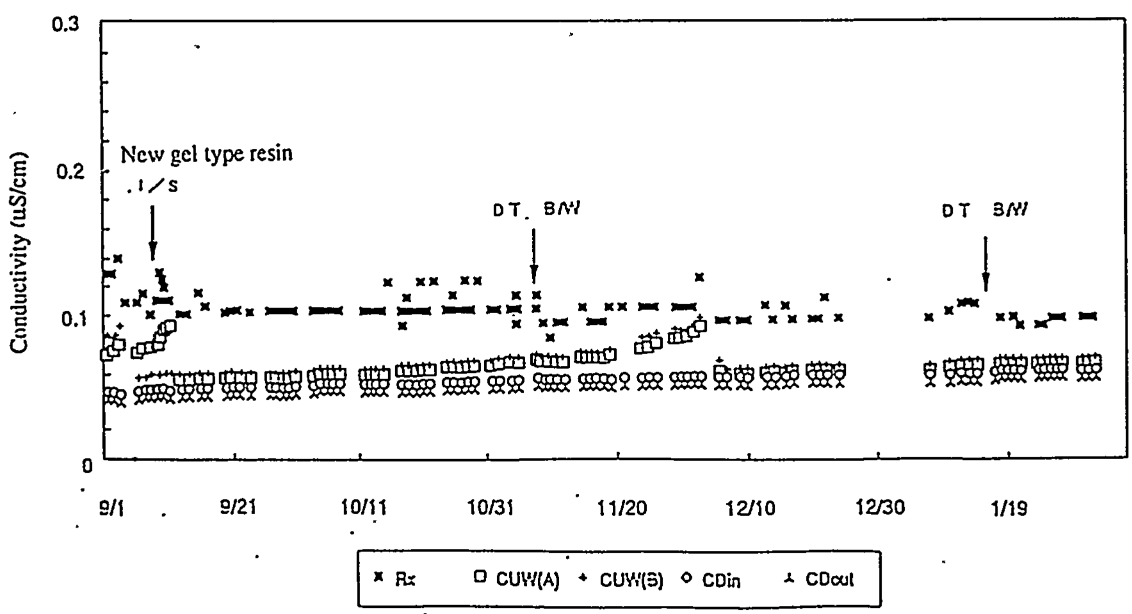


Figure 6 Suppression of elution of new gel type resin
(Improved new gel type resin)



1F-2 Crud Fe content (DT outlet)



Water quality

Figure 7

Plant decontamination plan

Decontamination sequence

Water quality improvement measures

→ Elimination of radiation source

- Reactor (water spray cleaning, cleaning with brush)
- Fuel

→ Decontamination of the portions with high dose rate

- System pipings [Water jet cleaning, chemical decontamination]
- Equipment [CO_2 blast, chemical decontamination]

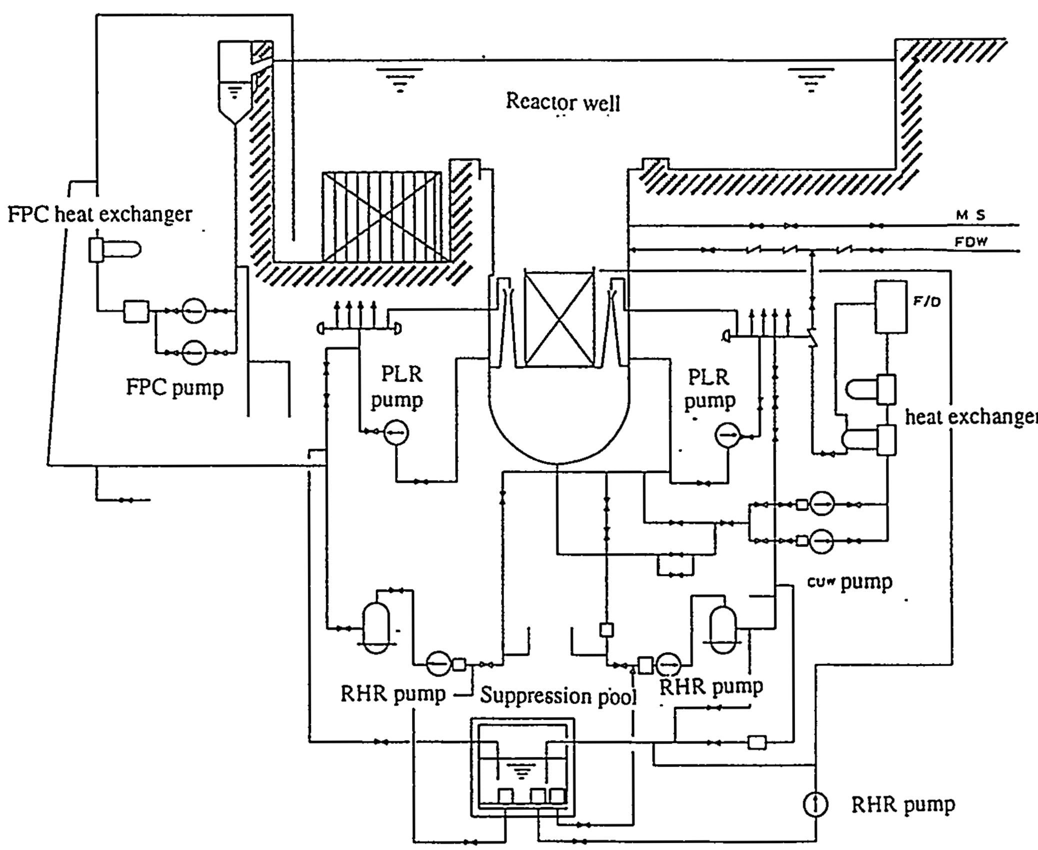


Figure 8

Summary

The old BWRs of TEPCO continue to create radiation exposure higher than the new BWR's, and the lowering of the crud content of feedwater and decontamination of the radioactivity accumulated in the reactor and systems are considered vital. For control of the crud content of feedwater, a CD resin with an improved ability to suppress the Fe content has been developed and will be applied to an actual system. On the other hand, for successful decontamination, it is essential to conduct proactive studies into the effects of suppressing radiation exposure and measures to prevent recontamination. We will proceed with these measures while checking effective decontamination methods and locations while considering crud content suppression measures to prevent recontamination.

Author Biography

Nagao Suzuki is a Deputy Manager, Head Office at the Radiation Safety Control Division of Tokyo Electric Power Company. He is in charge of monitoring of water chemistry at Fukushima Daiichi, Fukushima Daini, and Kashiwazaki Kariwa nuclear power stations. He also undertakes the planning and execution of water chemistry and equipment related measures, as required, by ALARA. In his former position, he was in charge of water chemistry, radiation, and release of radioactive waste controls at the Fukushima Daini nuclear power station. He holds a Bachelor Science in Industrial Chemistry (Engineering Department) from Meiji University.

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REPORT ON THE BWR OWNERS' GROUP RADIATION PROTECTION/ALARA COMMITTEE

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ABSTRACT

Radiation protection programs at U.S. boiling water reactor (BWR) stations have evolved during the 1980s and early 1990s from a regulatory adherence-based endeavor to a proactive, risk-based radiation protection and prevention mission. The objectives are no longer to merely monitor and document exposure to radiation and radioactive materials. The focus of the current programs is the optimization of radiation protection of occupational workers consistent with the purpose of producing cost-effective electrical power. The newly revised 10 CFR 20 defines the term ALARA (as low as reasonably achievable) to take into account the state of technology, the economics of improvements in relation to the state of the technology, and the benefits to the public health and safety. The radiation protection manager now must ensure that the program optimizes the protection of occupational workers and ensures the health and safety of the public while maintaining a cost-effective energy product.

The BWR Owners' Group (BWROG) initially formed the Radiation Protection/ALARA Committee in January 1990 to evaluate methods of reducing occupational radiation exposure during refueling outages. Currently, twenty U.S. BWR owner/operators (representing 36 of the operational 37 domestic BWR units), as well as three foreign BWR operators (associate members), have broadened the scope to promote information exchange between BWR radiation protection professionals and develop good practices which will affect optimization of their radiation protection programs.

In search of excellence and the challenge of becoming "World Class" performers in radiation protection, the BWROG Radiation Protection/ALARA Committee has recently accepted a role in assisting the member utilities in improving radiation protection performance in a cost-effective manner. This paper will summarize the recent activities of this Committee undertaken to execute their role of exchanging information in pursuit of optimizing the improvement of their collective radiation protection performance.

BACKGROUND

The Radiation Protection/ALARA Committee was formed in January 1990 to assist the BWROG Outage Management Committee evaluate methods of reducing occupational radiation exposure during refuel outages. BWRs have typically accumulated significantly more (180-190 Person-Rem/annum) exposure than the domestic Pressurized Water Reactor (PWR) Units (comparing median performance values per unit). Collective radiation exposure has been recognized as a valuable performance indicator of outage success and overall operating performance. Initial topics of discussion involved activities that directly affected outage performance (e.g., sub-system chemical decontamination, in-service inspection exposure management, refueling operations exposure reduction, and work in upper levels of the containment during fuel moves). The committee quickly recognized the success of this forum for information exchange and ventured out into addressing current critical issues facing the industry in all aspects of radiation protection.

MISSION

The mission of the BWROG Radiation Protection/ALARA Committee is to promote information exchange between BWR radiation protection professionals at site and corporate level positions. This information exchange is expected to allow the BWR operators to establish synergy and communicate the lessons learned and good practices utilized to optimize the mitigation of the effects of radiation on the nuclear power industry.

ACCOMPLISHMENTS

Three two-day meetings are held each year to provide the opportunity for each utility to attend at least two (assuming that each member may miss a meeting due to a refueling/maintenance outage). Committee attendance is not mandatory, however, 80-90 % of the member utilities are typically represented at each meeting. This high level of participation results in excellent and timely information exchange, discussions on critical issues facing the industry and improvement initiatives and strategies to address these issues. The committee dedicates approximately 50% of the meeting time to information exchange through the use of Plant Status Reports and the remainder of the time to high interest topics that are selected by a Steering Committee. Recent major meeting topics have included the following areas of interest:

- Long-Term Exposure Reduction
- Source Term Reduction
- Cobalt Reduction
- Chemical Decontamination
- ALARA Planning and Management
- Exposure and ALARA Initiatives for Repetitive Tasks
- Soft Shutdown
- Radiation Work Permit Process
- Exposure Reduction Incentives
- High-Radiation Area Control
- In-vessel Maintenance
- Health Physics Job Planning
- Implementation of the Revised 10 CFR 20 Rule
- Electronic Dosimetry and Access Control Programs
- Temporary Shielding

Presentations, panel discussions, and break-out sessions are typically led by member utility representatives. Institute of Nuclear Power Operators (INPO), American Nuclear Insurers (ANI), Nuclear Regulatory Commission (NRC), Electric Power Research Institute (EPRI), and various contractor and vendor representatives have also contributed significantly to the meetings allowing the communication and clarification of perceptions of industry performance and improvement efforts.

Recently two "WORKOUT" type sessions were held in which the cost effectiveness of radiation protection programs and the Utility/INPO interface were discussed. The cost effectiveness session resulted in 124 ideas for improvement being identified and several items identified for committee action and follow-up. Individual utility representatives were encouraged to further refine the cost effectiveness actions for potential short-term implementation at their sites. The Radiation Protection/ALARA Committee has established sub-committee working groups to further develop selected initiatives for utility wide endorsement. The recent Utility/INPO interface session resulted in a clarification of perceptions of the role that INPO has traditionally played in performance monitoring and assessment. Several improvement strategies were identified that would assist INPO and the industry in developing a synergistic role of INPO/Utility partnerships for assistance and improvement.

BENEFITS ACKNOWLEDGED

Participating utilities have found the information exchange and personal contacts to be invaluable problem-solving aids. Timely issues of high industry interest are discussed during the plant status reports and major meeting topics. In addition, information exchange in between meetings is performed through member-to-member discussions and committee sponsored surveys and questionnaires. The data exchanged during the plant status reports assists the member utilities to perform industry comparisons of their performance and provides for timely benchmarking of critical issues affecting their sites. A sub-committee has recently developed a process to begin routine (annual) collection of repetitive task exposure data for comparison and benchmarking.

Good practices are freely distributed to assist the industry. Temporary shielding program enhancements have assisted members save significant cost due to efficiency improvements addressed in the committee. Committee endorsement of the General Electric (GE) Service Information Letter (SIL) 541 regarding the implementation of "Soft Shutdowns" has greatly aided member utilities to support implementation at their sites. A peer assessment was organized through contacts made at committee meetings. Utilities have initiated sharing of equipment developed for specific high exposure tasks or for trial bases. Exposure and cost-saving ideas have been implemented throughout the membership. One member utility determined, using information obtained at a committee meeting, a way to save $ 13,000,000 in exposure savings through the use of $ 1,000,000 of permanent shielding.

Other qualitative benefits have been realized by member utilities. Participants improve their leadership and interpersonal management skills during this peer interaction. Personnel development is extremely important and value added by this participation.

COSTS

The budgeted funding for the BWROG Radiation Protection/ALARA Committee is developed each year by the Steering Committee and approved at a General Meeting of the BWROG Primary Representatives. The expenditures to hold three two-day meetings and perform the necessary project management functions have averaged approximately $80,000 per year. With 20 member utilities and four associate member utilities sharing the costs, each utility is assessed less than $4,000 per year to maintain participation. Additionally, each meeting attendee incurs travel and living expenses of typically $1,000 - $1,500 per person.

FUTURE ACTIVITIES

The next meeting scheduled for July 27-29, 1994 in Denver, Colorado, U.S.A., will be a joint meeting between the BWROG Radiation Protection/ALARA Committee and the PWR Radiation Protection/ALARA Committee. This meeting will discuss high interest industry critical issues that are common to Light Water Reactors (LWRs) (e.g. radiation protection impacts of zinc addition, radiation protection management of In-Service Inspection programs, spent fuel dry storage issues and litigation mitigation and defense). This meeting is expected to bring approximately 100 radiation protection professionals from 20 BWR utilities and 22-24 PWR utilities together to discuss initiatives to improve our industry performance. In addition, there is a third BWROG Radiation Protection/ALARA Committee meeting scheduled for December 1-2, 1994, in San Antonio, Texas, U.S.A. This meeting will focus on communication techniques for internal risk and Total Effective Dose Equivalent (TEDE) ALARA evaluations, permanent shielding applications and radiological concerns and management of failed fuel operations.

SUMMARY

The BWROG Radiation Protection/ALARA Committee provides a value added service to the member utilities to exchange information to assist them in their pursuit of optimizing their radiation protection programs. The benefits received by each member utility are significant and more than justify the costs associated with participation. The good mix of site and corporate personnel who participate provides for a broad base of expertise and understanding of all aspects of the issues discussed. A structured committee is essential to success of the identified mission. A program manager and steering committee provide the required long range planning, committee focus and continuity necessary to ensure effective and efficient meetings that meet the expectations of the membership.

Author Biography

Lary Aldrich is a Staff Health Physicist in the Health Physics Support Department of the Commonwealth Edison Company. Mr. Aldrich has over 15 years experience in Nuclear Power Health Physics. Mr. Aldrich's primary responsibility is the functional management of the radiation protection improvement initiatives for the company's six Boiling Water Reactor (BWR) units (Dresden 2 & 3, LaSalle County 1 & 2, and Quad Cities 1 & 2). Previously he has been responsible for the long-term exposure-reduction planning efforts as well as the research, development and application of advanced technologies used to reduce occupational radiation exposure for all twelve of the company's nuclear units. Prior to joining the corporate office staff, he worked as Radiation Protection Manager of the two-unit LaSalle County Station. He has a B.S. degree in Environmental Health/Health Physics from Purdue University and is a member of the Health Physics Society. In addition, he is a past Chairperson for the BWR Owners' Group Radiation Protection/ALARA Committee and was the founder and first Chairperson for the PWR Radiation Protection/ALARA Committee.

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REPORT ON THE BWR OWNERS' GROUP RADIATION PROTECTION/ALARA COMMITTEE

BACKGROUND

- FORMED IN JANUARY 1990 TO ASSIST OUTAGE MANAGEMENT COMMITTEE
- INITIAL TOPICS INVOLVED OUTAGE PERFORMANCE
  - Sub-system Chemical Decontamination
  - In-service Inspection Exposure Management
  - Refueling Operations
  - Work in Upper Containment Regions during Fuel Moves
- INFORMATION EXCHANGE

MISSION

- PROMOTE INFORMATION EXCHANGE
- ESTABLISH SYNERGY
- COMMUNICATE LESSONS LEARNED AND GOOD PRACTICES

MEETING FORMAT

- THREE TWO-DAY MEETINGS HELD PER YEAR
- *80 - 90 %* MEMBER UTILITY PARTICIPATION TYPICAL
- 1st DAY:
  - Plant Status Reports - *Timely Information Exchange*
  - Committee Administrative Issues
- 2nd DAY:
  - High Interest Topics on Committee Identified Critical Issues
  - INPO, ANI, NRC, EPRI, Contractor & Vendor Contributions

ACCOMPLISHMENTS

- CRITICAL ISSUES & STRATEGIES TO ADDRESS DISCUSSED
- PERCEPTIONS OF INDUSTRY PERFORMANCE CLARIFIED
- COST EFFECTIVENESS SESSION
  - 124 Ideas for Improvement Identified
  - Sub-Committees Established to Develop Selected Initiatives for Utility Wide Endorsement
- UTILITY/INPO INTERFACE SESSION
  - Perceptions Clarified
  - Improvement Strategies Identified to Foster Partnership

BENEFITS

- DURING MEETING:
  - Information Exchange - *Timely Benchmarking*
  - Personal Contacts Developed
- BETWEEN MEETING:
  - Information Exchange
  - Committee Sponsored Surveys & Questionnaires
- GOOD PRACTICES FREELY DISTRIBUTED:
  - Temporary Shielding Program Enhancements
  - "Soft Shutdown" Implementation
  - Equipment Sharing
  - Exposure & Cost Saving Ideas
- LEADERSHIP & INTERPERSONAL MANAGEMENT SKILLS DEVELOPMENT

COSTS

- BUDGETED FUNDING APPROVED BY BWROG PRIMARY REPRESENTATIVES AT GENERAL MEETING
- ANNUAL EXPENSES AVERAGING $ 80,000 FOR ENTIRE COMMITTEE
- MEMBERSHIP CURRENTLY:
  - 20 Member Utilities (*Domestic*)
  - 4 Associate Members (*Includes 3 Foreign Utilities*)
  - Each Utility Assessed < $ 4,000 per Year
- ATTENDEES INCUR TRAVEL & LIVING EXPENSES

FUTURE ACTIVITIES

- JULY 27-29, 1994 MEETING IN DENVER, COLORADO, U.S.A.
  - Joint Meeting Between the BWR & PWR RP/ALARA Committees
  - Discuss Critical Issues Common to Light Water Reactors
- DECEMBER 1-2, 1994 MEETING IN SAN ANTONIO, TEXAS, U.S.A.
  - Communication Techniques for Internal Risk & Total Effective Dose Equivalent (TEDE) ALARA Evaluations
  - Permanent Shielding Applications
  - Radiological Concerns & Management of Failed Fuel Operations

SUMMARY

- VALUE ADDED SERVICE TO MEMBER UTILITIES
- BENEFITS ARE SIGNIFICANT & JUSTIFY COSTS ASSOCIATED WITH PARTICIPATION
- MIX OF SITE & CORPORATE PROFESSIONALS PROVIDES BROAD BASE OF EXPERTISE
- STRUCTURED COMMITTEE IS ESSENTIAL TO SUCCESS - *Focus & Continuity*
  - Program Manager
  - Steering Committee
  - Membership Involvement & Direction

SESSION 8A

PWR AND CANDU PRESENTATIONS

Co-chairs:

Rolf Riess
Frank Rescek

USE OF MOCK-UP TRAINING TO REDUCE PERSONNEL EXPOSURE AT THE NORTH ANNA UNIT 1 STEAM GENERATOR REPLACEMENT PROJECT

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PROJECT BACKGROUND AND OVERVIEW

The North Anna Power Station is located on the southern shore of Lake Anna in Louisa County, approximately forty miles northwest of Richmond, Virginia. The two 910 Mw nuclear units located on this site are owned by Virginia Electric and Power Company (Virginia Power) and Old Dominion Electric Cooperative and operated by Virginia Power. Fuel was loaded into Unit 1 in December 1977, and it began commercial operation in June 1978. Fuel was loaded into Unit 2 in April 1980 and began commercial operation in December 1980.

Each nuclear unit includes a three-coolant-loop pressurized light water reactor nuclear steam supply system that was furnished by Westinghouse Electric Corporation. Included within each system were three Westinghouse Model 51 steam generators with alloy 600, mill-annealed tubing material. Over the years of operation of Unit 1, various corrosion-related phenomena had occurred that affected the steam generators' tubing and degraded their ability to fulfill their heat transfer function. Advanced inspection and repair techniques helped extend the useful life of the steam generators, but projections based on the results of the inspections indicated that the existing steam generators would not last their design life and must be repaired. To this end Virginia Power determined that a steam generator replacement (SGR) program was necessary to remove the old steam generator tube bundles and lower shell sections, including the channel heads (collectively called the lower assemblies), and replace them with new lower assemblies incorporating design features that will prevent the degradation problems that the old steam generators had experienced. Virginia Power contracted with Bechtel Power Corporation to perform the design engineering and construction for the SGR project.

Replacement Methodology

The procedure for the replacement of the three steam generators at North Anna Unit 1 was to sever the generators from the primary and secondary piping at the nozzle to piping welds and to separate the steam dome from the lower section of each steam generator by cutting in the transition cone area. This is commonly referred to as the "two-piece method" of SGR. This method was necessary because the diameter of the equipment hatch would not allow the passage of an entire steam generator. The steam domes were stored in containment for reuse with the new lower assemblies. Each old lower assembly was removed and replaced with a new lower assembly.

While the lower assemblies were being changed out, the existing steam dome transition sections were prepared to mate with the new lower assemblies. The steam domes were then fit to the new lower assemblies and the transition cone welds were performed. The welds were made to reconnect the vessels with the primary and secondary systems, and start-up testing was performed.

The project replaced each generator without having to replace any RCS piping components. This method is termed the "Two-Cut Method." For the two-cut method, each old lower assembly was severed from the existing RCS elbows at or near the existing attachment weld centerlines. The existing RCS elbows were decontaminated in place and the pipe ends were machined to correspond to the respective new lower assembly nozzles. When fit up, the new RCS nozzles matched the existing RCS elbows within permitted tolerances. Therefore, this method required only two RCS closure welds per steam generator.

Facilities

In order to support implementation of the SGR project, several temporary systems were installed in the containment building. These included a power distribution system that was fed from one of the reactor coolant pump motor feeds; an HVAC system that provided air conditioning for the work inside the steam generators and removed the smoke generated by the welding processes; a temporary service air system; a tower-mounted hydraulic crane to assist in the movement of light loads in the containment building; and an argon supply system to support the RCS welding activities.

To address the movement of the steam generator components within the containment building, a number of temporary and permanent modifications were made. These included the cutting and removal of sections of the biological shield walls at each steam generator, a section of the operating floor slab in front of the equipment hatch, and a section of the polar crane wall; installation of a deck system over the reactor cavity; installation of a runway and carriage system for movement of the lower assemblies; and the reroute or removal/reinstallation of electrical and piping commodities.

MOCK-UP TRAINING AND STEAM GENERATOR REPLACEMENT

An SGR project is one of the most complex, exposure-intensive projects that a nuclear power station will ever undertake. Keeping exposure As Low As Reasonable Achievable (ALARA) while achieving first time quality can only be accomplished by individuals that have experience in their particular skill, know the specific task which they must perform, and have been properly trained in the ALARA concept. In addition, innovative ideas and processes must be integrated into the work tasks that will enhance them and make the project ALARA.

The mock-up training program plays a vital role in the success of these projects through two primary functions. It allows processes to be performed, modified and enhanced to achieve the most efficient method of performance. It also allows individuals to become proficient in performing their task while contending with the associated radiological requirements and environmental aspects of the task. With the implementation of the new 10 CFR 20 requirements on January 1, 1994, an ALARA program is now a requirement all nuclear power stations must develop and follow to ensure personnel exposure is kept ALARA. A comprehensive mock-up program directed towards the ALARA program will help ensure personnel exposure during an SGR will be kept to a minimum.

Minimizing personnel exposure during an SGR project depends on the three basic ALARA principles - minimize time, maximize distance and effectively employ shielding. A comprehensive and properly implemented mock-up training program will address all three principles and develop methods to make the associated work task ALARA. How a mock-up training program can be implemented to support these three principles is described in the following paragraphs.

Shielding: Numerous source terms in the vicinity of the work area can directly attribute to the SGR project total personnel exposure. The primary method to reduce these source terms is by shielding. The mock-up training program should address both the evaluation of the proposed shielding packages to determine if they are the best packages for the intended purpose, and the qualification of the individuals that will be installing and removing the shielding packages.

Distance: Maximizing the distance from source terms from which an individual can perform a work task will significantly reduce the exposure received. The mock-up program should evaluate methods and processes such as remote welding and cutting operations that would permit as much of the work to be performed or monitored away from the source term in a low dose area. Work evolutions that require an individual to work in close proximity to a source term should be evaluated to determine if the process used to perform the work should be redesigned to position the individual in a low dose area.

Time: Reducing the time to perform a work evaluation is perhaps the most important method in reducing the overall exposure associated with an SGR project. The mock-up training program should include sufficient time to allow individuals to become proficient in their individual work task under the anticipated production work conditions. This will not only reduce the time required to perform the task, but promote first time quality and reduce the potential for rework.

SELECTION OF MOCK-UP ACTIVITIES

The selection of which activities would be subjected to mock-up training was based on both ALARA and Quality considerations. As previously mentioned, poor quality workmanship directly increases the total exposure for the project through the rework of activities and lengthening of the schedule. All scheduled activities were evaluated based on the following criteria:

- Time required to perform the task
- Physical location of the task
- Complexity of the method used to perform the task
- Contact and general area dose rates
- Experience with the technology used to perform the task

Based on the evaluation, the work activities that could significantly increase the total exposure of the SGR project were scheduled for mock-up training. The work tasks that required mock-up training are as follows:

- Installation of temporary reactor coolant piping supports
- Mechanical cutting of the reactor coolant piping
- Removal of the old steam generator support blocks
- Dry blast decontamination of the reactor coolant pipe ends
- Installation and removal of shielding

- Installation and removal of debris dams in the reactor coolant piping
- Machining of the reactor coolant piping
- Rigging of reactor coolant elbows (contingency measure)
- Weld build-up of the reactor coolant piping
- Setting and alignment of the new steam generator on the lower support structure
- Welding of the reactor coolant piping
- Welding of the steam dome to the lower assembly
- Reactor coolant pipe and steam dome internal radiography setup
- Primary system foreign object search and retrieval
- Operation of lower assembly transport carriage
- Tube bundle/annulus protection removal
- Optical templating of the steam generator channel head and reactor coolant piping

HARDWARE CONFIGURATION

The proper execution of a mock-up plan requires that the work be replicated in an environment that matches that of the production work area. In order to provide the proper amount of space for this type of arrangement, a pre-engineered metal building was erected adjacent to an existing fabrication shop. This building, which measured 40' x 40' x 30' high, was used to house the RCS activity mock-up structure. A 25' x 25' section of the fabrication shop was used to house the transition cone activity mock-up.

The full scale RCS activity mock-up of the steam generator channel head and lower support structure was constructed on the slab for the pre-engineered building, and the building was then erected around this structure. The channel head portion consisted of an actual Westinghouse Model 44 channel head, modified to simulate the Model 51 channel heads at North Anna. Heavy gauge sheet metal was added to the outside of the Model 44's shell to increase its diameter, new support feet were welded into place and machined, and stainless steel pipe extensions were added to each of the channel head's RCS nozzles to conform to the physical dimensions of the Model 51. A large steel support tower was fabricated to match the existing towers in containment. When the mockup was completed, dimensional checks verified that it was dimensionally identical to the system in containment.

The full size mockup of the transition cone was manufactured and installed in the fabrication shop. This mockup consisted of tube bundle protection, annulus protection, wrapper plate, and inside shell dimensions similar to those found on the existing North Anna steam generators.

Equally important as the replication of the structures is the simulation of the operational and environmental constraints that would be encountered during the production work. Temporary commodities such as scaffolding and lead were installed as they would be in the containment. It should be noted that the installation of several of these interfering commodities was in itself a mock-up activity. During the personnel qualification portion of the training, personnel wore the appropriate protective clothing (including respiratory protection, when required) and operated under the anticipated environmental conditions, such as elevated temperature and confined space entry, that would be encountered in containment.

PROCESS

In order to effectively implement the mock-up training, a Mock-Up Coordinator was established as a single point of contact for all mock-up activities. The Coordinator was responsible for ensuring that the development of all software and the resolution of all issues related to the training were completed by the responsible individuals in a manner that supported the training schedule.

Detailed plans were developed for both the content and sequence of the training. A list of specific objectives was identified for each of the activities, and acceptance requirements for each objective were established. The plan identified the requirements for implementation and inspection personnel, equipment, tools, material, and physical and environmental constraints. The sequence of steps that would be performed was based on the actual procedure that would be utilized in the field.

A detailed mock-up training schedule was developed that integrated the performance of the mock-up activities with the other work that was being accomplished at the site. This allowed the personnel to be trained with minimal impact on other preparatory activities. The schedule reflected the actual sequence of the work activities as much as possible.

Each activity that was selected for implementation in the mock-up was actually performed twice. The first performance was the qualification of the process for implementation under North Anna Unit 1 conditions. This was important even for processes that had been utilized on previous SGR projects. A good example of the plant-specific nature of these activities is the welding of the RCS piping. Equipment that had been successfully used on previous SGR projects to perform this function would not work at North Anna due to the unique configuration of the steam generator support structure. This was identified during the preliminary steps of the process qualification, and the equipment was modified to accommodate the physical interferences. Had the interferences been discovered during the implementation of the activity during the outage, a significant schedule impact would have occurred.

After the completion of the process qualification, there was a significant amount of proficiency training that was performed as part of the mock-up process. The amount of proficiency training that was required varied with the complexity of the task and the experience of the individual. Satellite training stations were installed in the mock-up building in order to perform this practice. The satellite stations included set-ups for both the hot and cold leg RCS elbows on which the many RCS activities could be practiced and refined prior to the qualification testing.

Once the process had been proven and accepted for use and the personnel had received the required amount of proficiency training, the personnel qualification phase of the mock-up was implemented. All personnel that would be performing the activity in containment were required to demonstrate their technical proficiency. This proficiency was evaluated by both the technical supervisor of the activity and the responsible Radiation Protection supervisor. This approach ensured that the individual was not only capable of performing his or her task, but could do so in the most ALARA-effective manner possible.

Effect on Radiation Protection Procedures

The simulation of the actual work conditions, minus the source terms, that workers would encounter while performing their work in containment allowed Radiation Protection management the opportunity to evaluate their processes and personnel. This evaluation helped to ensure that the radiological controls placed in the work procedures and the Radiation Work Permits were appropriate. It also gave the Health Physics technicians the opportunity to witness the processes that would be used to perform the work task and plan their strategy for providing the required coverage accordingly. An example of this is the opportunity that the Health Physics technicians had to gain experience with the camera system and remote dosimetry that would be used to provide coverage of work tasks inside the RCS Loop Rooms.

One major part of this process was to determine if engineering controls, such as ventilation, could be used to reduce the use of respirators. Virginia Power had instituted a respirator reduction program prior to the scheduled SGR, therefore it was very important to perform the SGR using the same logic. Although many of the procedures initially required the use of respiratory protection, work performed in the mock-up allowed engineering controls to be developed and used in lieu of a respirator, thus reducing the time and dose required to perform the task.

It should be noted that the Health Physics technicians that were to cover the work in containment were brought on site considerably earlier than they would for a typical refueling outage in order to go through the mock-up process. This investment in their training proved invaluable in eliminating the unnecessary stoppage of critical activities due to a lack of understanding as to the process being performed. In fact, because of their background covering these types of work operations on previous projects, many of the Health Physics personnel were able to offer valuable suggestions on how to improve the work processes.

MOCK-UP TRAINING RESULTS

Mock-up training had a big impact on the results for the North Anna Unit 1 SGR Project. The effective training that the workers received was a direct contributor to the excellent safety record that the project achieved, a 0.0 OSHA Incidence Rate.

The effect on the personnel exposure for the project was equally significant. The final estimate for all SGR activities was 480.7 Person-Rem, which included exposure resulting from SGR preparatory activities performed in the outage prior to the actual SGR outage. Based on the higher source terms at North Anna versus SGR projects performed previously at other stations in the United States, this estimate was believed to be aggressive but achievable. The actual exposure for the SGR activities was 239.9 Person-Rem, which was less than half the exposure of any other SGR performed in the United States. Accomplishing this can be attributed in large part to the mock-up training program. Table 1 shows the estimated exposure versus the actual exposure for all the activities that were part of the mock-up training.

LESSONS LEARNED

Many lessons were learned through the implementation of the mock-up training program at North Anna, but none is more significant than the value of the program itself. The results achieved in the areas of safety, quality, schedule minimization, and reduction of personnel exposure all point to the need for a comprehensive mock-up training program.

Other lessons learned from the mock-up program include:

- The program should be jointly developed and implemented, i.e., both the utility and the contractor should have input to the plans and procedures. This ensures that all affected organizations will be part of the process.

- The personnel performing the work tasks in the mock-up should receive inspection and radiation protection coverage from the same individuals that will be performing these functions in the containment. This provides continuity of the working relationships developed during mock-up training and prevents stoppage of the work in containment due to a lack of understanding of the methodologies or objectives of the process.

- The sequencing of the mock-up training should allow the personnel training and qualifications to take place as close to the production work as practical. This will eliminate the need to train additional personnel to compensate for attrition of the labor force.
- Skilled labor should be utilized for activities that require more than brute strength. The increased ability to identify ways to execute work in a more efficient manner more than offsets the additional wage requirements for these personnel.

- The more the mock-up structures and environments are identical to the containment conditions, the better prepared the workers will be to deal with the complexities of their tasks.

- As can be seen in Table 1, there were several activities that exceeded their estimated personnel exposure. These activities have been evaluated as to why the budgets were exceeded and corrective actions, including the expansion of the training program in several areas, have been put in place for North Anna Unit 2 SGR project.

Table 1. Activities that Were Part of Mock-up Training
Estimated vs. Actual Dose

MOCK-UP ACTIVITY	ESTIMATED DOSE	ACTUAL DOSE	PERCENT
Installation of temporary loop piping supports and mechanical cutting of the RCS piping	43.8	8.1	18.6
Dry blast decon of the RCS pipe ends	19.7	13.7	69.4
Installation & removal of general area shielding	19.9	8.4	42.4
Installation & removal of debris dams and internal shielding in the RCS piping	4.1	2.1	51.1
Manual decontamination methods (general area)	14.4	7.3	50.8
Machining of the RCS piping, welding of the RCS piping and weld build-up of the RCS piping	75.1	32.2	42.8
Rig in and out cold leg elbow	N/A	N/A	N/A
Removal of the old and setting and alignment of the new S/G on the lower support structure	17.8	18.5	103.5
Welding of the steam dome to the lower assembly	3.4	3.0	90.4
RCS pipe & steam dome internal radiography setup	5.2	2.4	46.6
Primary pipe foreign object search and retrieval	2.1	5.0	240.2
Operation of lower assembly transport cart and installation of the impact ring	1.1	1.4	125.5
Tube bundle/annulus protection	N/A	N/A	N/A
Optical templating of S/G channel head & RCS piping	4.8	3.6	73.8

Authors' Biographies

Gene Henry is a Senior Staff Engineer at the North Anna Power Station and currently working as a Radiological Engineer in the Radiation Protection Department. Currently, he is responsible for preplanning for the Unit 2 Steam Generator Replacement Project and working with System Engineering to upgrade and balance the station's ventilation systems. He was the Supervisor of the Radiation Protection group responsible for all planning and implementing of radiological controls for the North Anna Unit 2 SGRP. Before being assigned to the SGRP, he upgraded the ventilation program at North Anna, including designing new ventilation systems, developing a ventilation course, developing a DOP test program and working with training to provide the training to implement the program. Prior to joining Virginia Power, he worked for the Norfolk Naval Shipyard as a Nuclear Engineer and was responsible for all nuclear ventilation, both portable and permanent, radiological controls for performing the work task, and Supervisor in charge of the refueling activities.

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PAPER 8A-1
DISCUSSION

Na: What is the benefit of cutting the steam generator instead of complete replacement?

Reilly: There were a lot of considerations in deciding whether to do it one piece or two pieces. One of the considerations was how to get a one-piece generator to the site. Originally, back in the early 1970s when the plant was constructed, they came in one piece. They traveled over about 40 miles of road. The utility didn't feel that the political process that would have to be pursued was a good avenue to take, so we studied the two-piece option and found that it worked. There were some minor trade-offs.

Bush: You had mentioned a monetary reward in your presentation. Was that for the workers? If so, how exactly did that work?

Reilly: We set up an incentive program that was based on safety goal, man-rem goal, personnel contamination event goal, and a schedule goal. We put out a chunk of money and said, "For each one of these goals, this is how much it's worth to you as an individual." It's based on the number of hours the person worked on the project over the total hours work by all the craft and the subs.

ELECTRIC POWER RESEARCH INSTITUTE

EFFECTS OF RESPIRATOR USE ON WORKER PERFORMANCE

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ABSTRACT

In 1993, EPRI funded Yankee Atomic Electric Company to examine the effects of respirator use on worker efficiency. Phase I of Yankee's effort was to develop a study design to determine respirator effects. Given success in Phase I, a larger population will be tested to determine if a statistically significant respirator effect on performance can be measured. This paper summarizes the 1993 EPRI/Yankee Respirator Effects Pilot Study, and describes the study design for the 1994 EPRI/Yankee Respirator Study to be conducted at the Oyster Creek Nuclear Power Plant. Also described is a summary of respirator effect studies that have been conducted during the last ten (10) years.

INTRODUCTION

As of January 1, 1994, the NRC, through implementation of the new 10 CFR Part 20, is requiring licensees to keep internal plus external doses ALARA. With implementation of this regulation, the NRC is requiring licensees to make decisions based on respirator effects on worker efficiency.

As an example, if a worker is in a radiation area where he will accrue an effective dose equivalent of 100 mrem in an hour from inhalation of airborne activity, and accrue an effective dose equivalent of 200 mrem in an hour from an external radiation source, and respirator use reduces worker efficiency by 50%, then the one hour job shall be performed without a respirator to comply with good ALARA principles. The total effective dose equivalent (TEDE) differences for performance of this one hour job, with and without the respirator, are as follows. With a respirator the 1 hour job will take 2 hours to complete because of the 50% reduction in worker efficiency. The TEDE for this job will be 400 mrem (0 internal + 400 external). Without a respirator the job will take 1 hour, and the TEDE is 300 mrem (100 internal + 200 external).

In summary, Subpart H of the new Part 20 requires licensees to decide when to use respirators to achieve the new Part 20 goal of keeping the TEDE ALARA.

OBJECTIVE

The objectives of the current Yankee/EPRI study are to determine if respirator use affects worker performance during a standard nuclear power plant task, and if possible, to quantify this effect. This study has been subdivided into two phases.

Phase I consisted of developing a study design to measure the respirator effect, and then determining if this study design was capable of quantifying the respirator effect by testing a population of 6 workers. This report presents the results of Phase I.

Based on the results of Phase I, Phase II will consist of testing a population of about 50 workers, to quantify the respirator effect with statistical significance.

LITERATURE SEARCH

Three independent literature searches on the topic "Respirator Effects on Worker Efficiency" have been performed. The first search was conducted in 1985, by the Georgetown University Library staff. This literature search was part of a respirator efficiency study conducted at TMI[1]. The second literature search was conducted by Encore Technical Resources, Inc. in 1992, and a third search was conducted at Virginia Polytechnic Institute in April of 1993. Both of these literature searches were conducted as part of the Yankee/EPRI study. The results of these searches identified only one applicable study that quantified respirator effects on worker efficiency.

Most of the studies measured the body's response (i.e., heart rate, temp., etc.) to different work levels while wearing respiratory protection, or stay time limits to avoid heat stress related health effects while using respiratory protection.

In May of 1982, the U.S. ARMY conducted studies which involved the discharge of firearms while wearing gas masks, but these results are difficult to apply to nuclear power plant worker efficiency. A study conducted in 1984, at Ontario Hydro[2], consisted of quantifying worker efficiencies while wearing protective apparel; however, the respiratory protection equipment used in this study (Air-Supplied Suit) is significantly different than the respiratory protection equipment commonly used in the U.S. (Face Mask). Also, a 1992 University of Maryland/Army study predicted worker efficiencies while wearing protective apparatus, but most of this data is based on models rather than actual human experiments.

In 1985, a study was conducted at TMI, in which 48 male nuclear power plant workers performed a 20 minute task with and without a respirator[1]. The objective of this study was to demonstrate the respirator effect on worker efficiency. Each worker performed the test twice. Half the workers wore the respirator for the first trial, and the other half wore the respirator for the second trial. Statistical analysis of the test data showed that all workers performed the task faster the second time, regardless of whether they were wearing a respirator. The workers had not performed the task before the actual testing, and did it faster the 2nd time as a result of "learning" the task. However, because the order of performing the first test with a respirator was staggered, worker efficiency analysis was examined independent of the learning effect. This analysis determined that there was no respirator effect for this 20-minute task.

In 1990, a respirator efficiency study was conducted at Commonwealth Edison by a graduate student at Northwestern University[3]. This study consisted of performing four (4) different tasks with and without a respirator. The tasks were valve repacking, insulation replacement, pipe replacement, and gate valve inspection. The tests were conducted at room temperature, and varied in length from 20 to 47 minutes. Eighteen maintenance workers participated. The valve repacking task was completed, on the average, 29% more efficiently in a respirator. The overall efficiency of the remaining three tasks was 18% more efficient while working without a respirator.

STUDY DESIGN - PHASE I

The objective of Phase I, as stated above, was to develop a study design that measures the respirator effect on worker efficiency, and then test the study design to determine if it is capable of measuring this effect. Six workers were to be used to test the study design, and if possible, quantify the respirator effect. It was during this phase that the study design was to be adjusted, as the preliminary testing was being conducted, in order to meet the quantification objective during Phase II.

A number of meetings were held to develop each aspect of this study design. After careful review of a number of different plant maintenance procedures, the swing check valve inspection was chosen because we anticipated that it would take about an hour to complete, and that it would be a very strenuous task. However, parts of the procedure, such as the inspection steps, were removed because they were qualitative in nature instead of quantitative. Including the inspection steps would have made it possible for the workers to rush if they were uncomfortable, and wanted to complete the task sooner to remove the respirator. (As a result of removing these steps, the average task completion time for the six workers was about 35 minutes.)

The amount and type of training to be given to the workers before they actually performed the test, were also given careful consideration. This was done to avoid testing while the workers were still on the steep part of the learning curve. This learning factor in the TMI study was larger than the respirator factor, and as a result, any small variance brought about by the respirator may have been masked. It was decided that the workers would be trained until their task completion times, without the respirator, no longer showed large differences from one trial to the next.

Consideration was also given to training the workers on a mock-up valve as opposed to the actual in-plant valve. The options were as follows: (1) train the workers on the mock-up and then have them perform the test on the in-plant valve, to be representative of how this task might be performed in the nuclear industry, or (2) train and test the workers on the actual plant valve to obtain a more accurate measurement of just the respirator effect on worker efficiency (rather than introduce more variables by having the surroundings of the valve change between training and testing). When it was observed in training that the first worker performed the task at the mock-up very differently than he did at the in-plant valve, due to surrounding space limitations, it was decided that all workers would be trained and tested at the in-plant valve. The reason for this was to ensure that the respirator effect would not be hidden by the new learning factor associated with the adjustment of performing the task at the in-plant valve.

Humidity and heat directly affect worker efficiency through heat stress limitations. The literature[4] demonstrates that heat stress limits are more restrictive when workers are wearing respirators. Temperature could be controlled with a six foot space heater, and both temperature and humidity were continually recorded.

The type of respirator to be used in the study was the full face negative pressure (FFNP) respirator. Although many different types of respirators are used in the U.S. nuclear industry, the FFNP type was chosen because it is believed to be the most frequently used respirator in the industry. Nose cups were used in the respirators of two of the workers to see if mask fogging was reduced.

Phase I was conducted with 6 experienced nuclear power plant workers who had already been medically approved for respirator use. Three of these workers were from Yankee's maintenance staff, and the other three were Yankee staff electricians. All workers participating in the study, filled out a questionnaire regarding their age, height, weight, work experience, smoking status, eyesight, exercise, and medical status. This questionnaire also asked the worker for their views on internal v.s. external dose, and job performance with a respirator. The results of this survey are given in the Lessons Learned section of this report.

Two workers were to report to the plant for two consecutive days. During the morning of the 1st day the workers attended a classroom training session, where they were trained on each step of the task that was to be performed. After the formal classroom training was concluded, each worker performed the task alone. While worker # 1 was performing the task, worker # 2 was resting. Workers 1 through 4 practiced the task twice before being tested. Two of these 4 workers had differences between their 1st and 2nd training completion times of greater than 10%. To further reduce the learning factor, workers 5 and 6 practiced the task three times before being tested. For workers 5 and 6, completion times for practice performances 2 and 3 differed by less than 6.5%.

On the second day, worker #1 would dress up in full PCs, and perform the task once in the morning with a respirator. Worker #1 would then perform the task a second time after a 1.5 hour rest, with full PCs and without a respirator. Worker #2 would perform the task in the morning, while worker #1 was resting. Worker #2 wore full PCs and no respirator for his first performance, and his second performance would be in full PCs and a respirator.

All test performances were videotaped.

A six foot space heater was used during testing, to bring the temperature up to about 78°F. For testing of workers 3, 4, 5, and 6, a temperature of 78°F was reached in the testing area without the use of the space heater, due to high outdoor temperatures. Humidity could not be controlled. The exact temperature and humidity for each worker's test performance was recorded. The Environmental Conditions section below discusses temperature and humidity in more detail.

TASK DESCRIPTION

A standard plant procedure "Inspection Procedure For 10 inch Swing Check Valve" was used for Phase I. As discussed in Section IV, this task was selected due to the physical activity involved and the estimated length of time to complete. The task consisted of the following nine subtasks. (This check valve is illustrated in Diagram 1.)

1. Loosen the twenty 1 13/16" nuts with combination wrench and breaker bar.

2. Remove nuts and bolts by hand and place them in holding bucket.

3. Place two eyebolts in the bonnet cover, and remove bonnet with chain fall.

4. Perform blue test inspection by placing blue ink around the perimeter of the valve disc, and seeing if a blue ring can be seen continuously where the swinging valve disc meets the pipe. Once complete, ensure that the ink is wiped from the valve so that this same test can be conducted by the next worker.

5. Replace bonnet cover, with chain fall, by lining up the bolt holes as the bonnet cover is lowered onto the valve.

6. Install bolts and tighten nuts by hand.

7. Using a torque wrench with a light that indicates when the pre-set ft-lb pressure is reached, tighten the bolts in the sequence illustrated in Diagram 2, to a pressure of 160 ft-lbs.

8. Tighten the bolts in the sequenced referenced above, to a pressure of 330 ft-lbs.

9. Tighten the bolts in the sequenced referenced above, to a pressure of 500 ft-lbs.

ENVIRONMENTAL CONDITIONS

Training and testing were conducted on the ground elevation of the Yankee Nuclear Power Station Turbine Building. The temperature during testing of workers 1 and 2 was 78°F, as maintained with a six foot space heater. The average relative humidity was 63%. For workers 3 through 6, the average temperature was 78°F, and the average relative humidity was 77%, due entirely to external environmental conditions.

A Heat Index, was calculated from a plant heat stress procedure[4], for all testing. The Heat Index for all testing was between 80 and 85. This Index is a measure of the physiological heat stress imposed on the human body, from the additive effect of both temperature and humidity. NIOSH recommended stay times for a Heat Index between 80 and 85, for heavy work (heavy lifting, pushing, or pulling especially when using a negative pressure respirator) are 90 minutes for single PCs, and 25 minutes for Plastic PCs. Workers that were tested in this study, used paper and cotton PCs, which corresponded to NIOSH recommended stay times between 25 and 90 minutes. Therefore, with an average testing performance time of about 35 minutes, there was no immediate concern with heat stress.

LESSONS LEARNED

Mock-Up Facility

A mock-up of the check valve was used to train workers 1 and 2. It was our intention that following mock-up training, these workers would perform the actual test performance in full PCs, with and without a respirator, on the actual in-plant check valve. This mock-up was set up in a section of the Turbine building where there were no obstructions that could interfere with performing the task. However, by mistake, worker 1 conducted his last training performance on the actual in-plant check valve, and his completion time was noted to increase dramatically. This performance time increase was caused by various pipes and other obstacles surrounding the in-plant valve, that forced the worker to perform the task at angles that were different from his task performance at the mock-up. Noting this difference between the mock-up and the in-plant check valve performance, all subsequent training was conducted at the in-plant check valve. Our primary objective in this study was to quantify the effect of the respirator, only, on worker efficiency. Therefore, to minimize the learning effect that would be experienced during the worker's 1st and 2nd performance of the actual test on the in-plant valve, the mock-up was excluded from the study.

Fatigue Factor

Workers 1 and 2 practiced the task once in the morning of the day they were to be tested. After this practice performance, they completed the task two more times that same day for testing. It was observed that these workers were very tired during their third task performance of the day, regardless of whether they were wearing a respirator. To keep this fatigue factor from influencing the test results, workers 3, 4, 5, and 6, did not practice the task on the morning of test day, and therefore, only performed the task two times on the day they were being tested.

Competition Between Workers

Study participants appeared to perform their test trials more diligently, when they realized they were being timed and videotaped. Four of the six participants inquired about their completion time after each test performance. It appeared that the workers were trying to improve their completion times, and compete against their co-workers. This was especially true if co-workers were allowed to watch the test trials.

To minimize this competition, later workers were: 1) separated such that they could not observe their co-workers performance, 2) told that although we were videotaping their performance, they should work quickly without rushing, and 3) told they would be informed of their completion times and study objective when the testing was complete.

Rushing With Respirator

It appeared that those workers that could rush through the task while wearing a respirator, would. Their goal appeared to be to remove the mask as soon as possible. Only one out of the six workers interviewed after testing, stated that he really didn't mind wearing the respirator. Two of the remaining five workers had body weights of about 160 lbs. These workers were unable to rush through the task while wearing the respirator, because the task was too strenuous. Both of these workers had to take significant rest periods during their performance with a respirator. The remaining three workers were over 200 lbs. in body weight, and were physically capable of rushing through the task. These three workers did not show significant increases in their completion times when performing the task with a respirator; however, upon completion of the task with a respirator, they were significantly more fatigued. One of these workers stated that he couldn't have kept up the pace with the respirator, if the task had been any longer.

Further Reduction of Learning Factor

Workers 1 through 4 practiced the task two times before testing to reduce the learning effect. The greatest difference between the 1st and 2nd training completion times for this group was 25%. In order to further reduce the learning effect, workers 5 and 6 practiced the task three times before testing, and the greatest difference between the 2nd and 3rd training was 6.5%.

Questionnaires

The questionnaires used in this study contained questions regarding the workers' views on respirator use, internal versus external dose, and specific questions regarding work experience and health. The questions concerning respirator use and internal versus external dose could have informed the workers of the study objective. If the worker is aware of the study objective his performance could be biased. In an attempt to reduce any bias, the last two workers were given the questionnaire after they completed the testing.

Phase II will have workers complete a questionnaire before testing. The questionnaire will only contain questions regarding work experience and medical status for population grouping within the study. After the workers have completed their testing, a second questionnaire (survey) can be distributed to the workers asking for their views on working with respirators, and external versus internal dose.

Mask Fogging

Mask fogging in all six workers was only observed during exhalation; however, this fogging effect immediately disappeared upon inhalation. Nose cups were used in the respirators of two of the six workers to see if mask fogging could be reduced. There were no observed differences of this fogging effect between those respirators with nose cups, and those without nose cups.

PHASE I DATA ANALYSIS

Overall, the results showed subjects performed the task 10%[1] slower when using the respirator (Figure 1). This result is not statistically significant, due to the small population of six workers.

Subtask times were obtained from reviewing the videotape of each worker's performance. The average subtask times with and without a respirator are shown in Figure 2. These subtasks were analyzed individually for the respirator effect with a Repeated Measures Analysis of Variance test. This statistical test is similar to a Paired T-Test, in which each individual competes against himself. Of the nine subtasks, only subtask 5, Replacing the Bonnet Cover, was statistically significant at the 95% confidence level ($p<0.05$). The magnitude of this difference was a 20% slower performance time with the use of a respirator. This subtask required clear vision to allow the worker to line up the 20 bolt holes as the Bonnet Cover was lowered onto the valve. This high visual component of subtask 5 seems likely to have contributed to this result.

The reason most of the subtask analyses were not statistically significant is because of the small sample size, and that there was considerable variability between subjects. In particular, the lighter subjects (subjects 2 & 4) showed a much larger increase in task completion time (19%) when using the respirators as compared to the heavier subjects (subjects 1,3,5, & 6) who showed a 3% decrease in completion time. This can be seen in Figures 3 & 4, where the effect of the respirator on each task is shown separately for the light and heavy subjects. Note that the effect of the respirator is much larger for the lighter subjects in the later strenuous subtasks.

This greater effect of the respirator on light weight subjects, especially for later tasks, indicates worker fatigue was probably playing an important part in the results. The lighter workers were also older than the heavier workers in this study. The average age of the lighter workers was 50, and that of the heavier workers was 37.

A statistical power analysis of a respirator effect on the overall task, based on the Phase I performance times of the two lighter workers, shows that in order to have a greater than 90% chance of detecting a significant difference in performance time due to wearing a respirator, at the 0.05 significance level, 20 workers would be needed.

CONCLUSIONS

The conclusions of the Phase I test program are:

1. Overall, the average completion times for the six workers showed a 10% increase when respirators were worn. This increase was not statistically significant due to the limited number of workers tested.

[1]This is preliminary data from the Phase I Pilot Study. The Phase II EPRI Final Report will be available in 1995. For further information, contact Ron Cardarelli (617) 662-3932.

2. When lighter workers were analyzed separately, they showed a 19% increase in completion times. The increase in performance time for these lighter workers was apparently due to the physical demands of the task relative to the strength of the worker.

3. The heavier workers showed essentially no difference in their completion times.

4. The lighter workers coincidentally were older (average age 50) than the heavier group (average age 37), and this may have contributed to the increased respirator effect for the lighter workers.

5. Figure 3 shows that the increase in performance time for the lighter workers when wearing a respirator is greater for the latter subtasks. Fatigue seems to play a very important role in these findings. Understanding this relationship is important for evaluating jobs with different completion times and/or jobs requiring various degrees of physical strength.

RECOMMENDATIONS FOR PHASE II

Based on the results from Phase I, Phase II will consist of two groups of 25 workers each, where the confounding variables such as body size and age are balanced between the two groups. After four training performances, Group 1 will perform the test first without a respirator, and then a second time with a respirator. Group 2 will perform the test first with a respirator, and then the second time without a respirator. This will allow comparison of Group 1 performances without the respirator to Group 2 performances with the respirator. Because the two groups will be balanced, this analysis will show the effect of the respirator on the general nuclear plant work force.

Also, a Paired T Test analysis can be performed, where each worker's performance, with and without a respirator, is analyzed. In this analysis, each worker is competing against him/herself such that the respirator effect can be evaluated independent of the great variances between human subjects. This will also allow for the confounding variables, such as age and body size, to be evaluated separately.

Figure 3 shows that for the lighter workers, the differences in task performance with and without a respirator were greater in the later subtasks. This strongly suggests that fatigue plays a very important role in understanding the effect of respirator use on worker efficiency. To measure the effects of fatigue, and to examine other work tasks in addition to physically strenuous ones, the work task in Phase I should be extended to include a specific dexterity test. This will allow the results of Phase II to be applied to nuclear power plant work tasks that are not strenuous, as well as to those that are. The extended work task will be designed such that a heat stress problem is not created.

The lighter workers in the Phase I study were older (average age 50), than those in the heavy body weight group (average age 37). This may have played a role in the different performances between these groups. Age, as well as physical condition and size (strength), will be examined in Phase II to see if they contribute to the respirator effect.

All testing will be conducted in an environmental chamber to simulate actual work environments and to keep the testing area heat and humidity constant for all tests. Also, the Powered Air Purified Respirator, as well as the Full Face Negative Pressure Respirator, will be evaluated.

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Author Biography

Ron Cardarelli has worked for Yankee Atomic Electric Company (YAEC) as a Health Physicist since April 1986. In addition to providing health physics support for a number of nuclear power utilities, he is the Principal Investigator of the EPRI-sponsored Respiratory Effects Study. Before joining YAEC, Mr. Cardarelli was a Radiation Specialist for the U.S. NRC at the Office of NMSS. While working for the NRC, Mr. Cardarelli attended the Radiation Science Graduate Program at Georgetown University. As part of his thesis, he conducted a study at TMI involving 48 plant workers in which he examined the effects of respirator use on worker efficiency. Mr. Cardarelli has a M.S. in Radiation Science from Georgetown University and a B.S. from U-Mass, Amherst, in Environmental Health.

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Diagram 1. Inspection Procedure for 10" Swing Check Valve

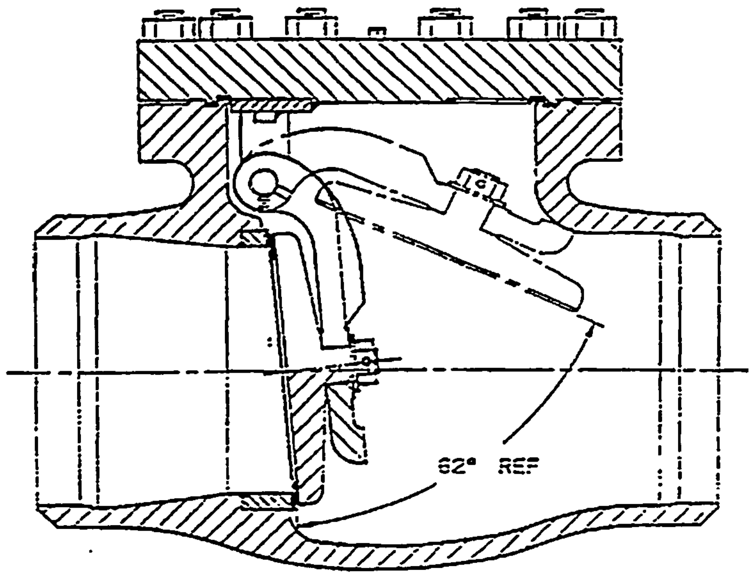

Diagram 2. 20-Hole Bolt Pattern

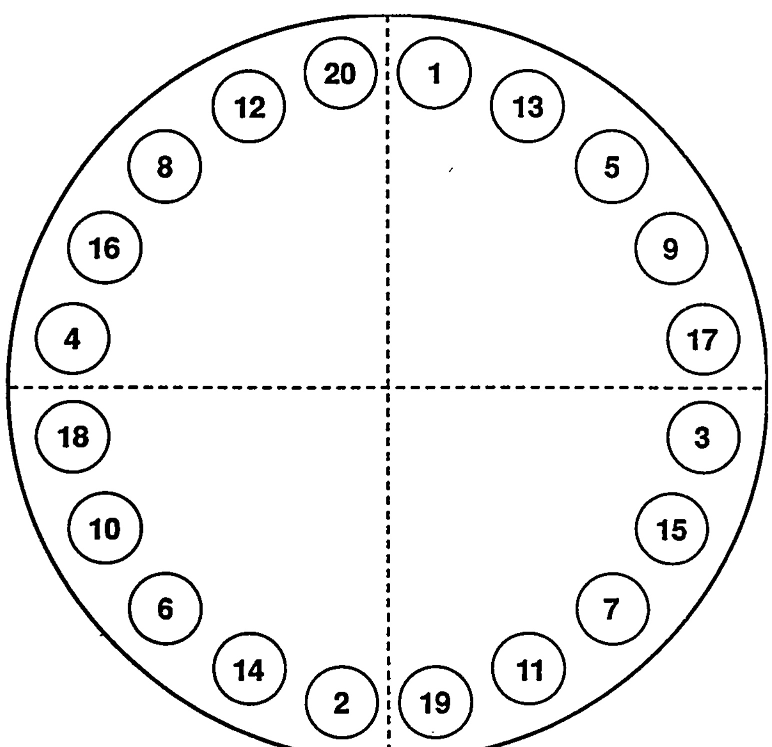

Figure 1. Total of all Subtask Averages

w/Out Resp.	w/Resp.
1463	1624

(Seconds)

10% Slower w/Resp.

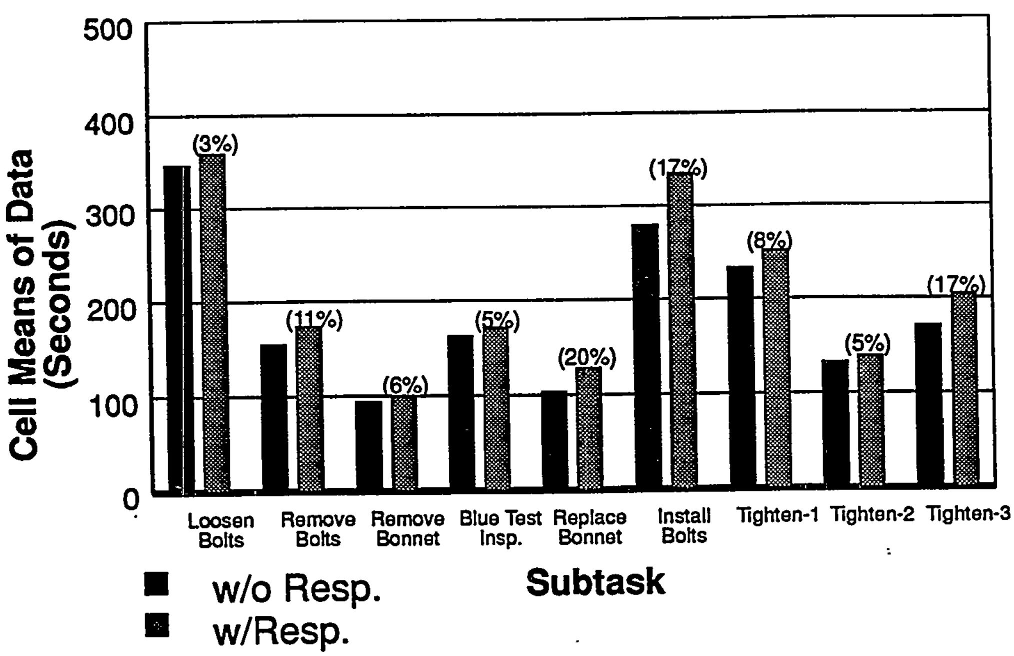

Figure 2. Subtask Averages

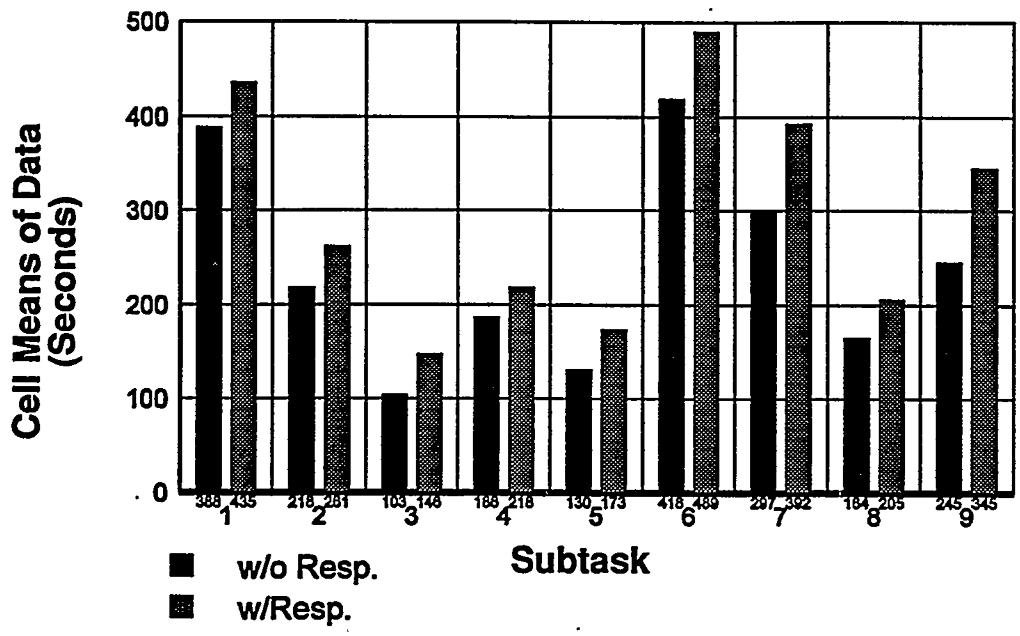

Figure 3. Subtask Averages for Small Subjects: Average Age 50; Average Height 66.5 Inches; Average Weight 157.5 Pounds

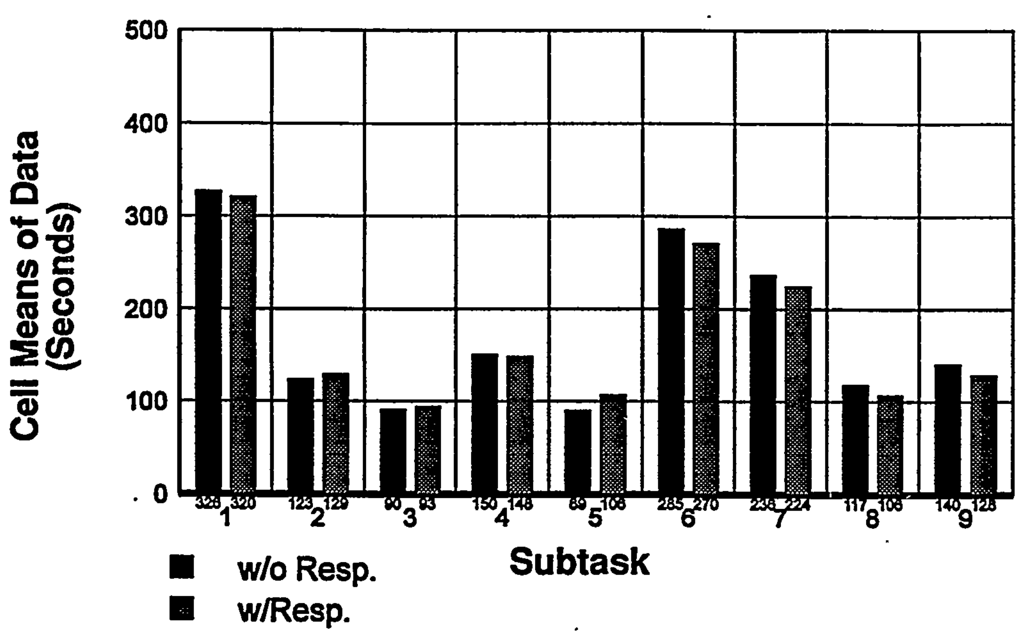

Figure 4. Subtask Averages for Large Subjects: Average Age 37.75; Average Height 72.5 Inches; Average Weight 223.25 Pounds

PAPER 8A-2
DISCUSSION

Tucker: You mentioned a dexterity test and you talked about the fatigue. I wonder, what is your feeling about how this affects concentration and a person's ability to actually complete the task correctly?

Cardarelli: We are hoping that this dexterity test will actually measure concentration. This is the reason for using a dexterity test. We saw fatigue at the end of the 1993 vigorous task and we want to try to measure it.

Westbrook: I noticed your people, with and without respirators, were wearing short-sleeved outfits. How representative is that of the real situation. If we were to put someone in a respirator, he would be wearing at least a single suit of anti-Cs and possibly a double suit, and if it were wet work, he'd be wearing a waterproof suit. In the results, even though you find an advantage without the respirator, the difference doesn't seem to be very strong. You understand, of course, when you try to make this study scientific, you are going against a huge amount of anecdotal evidence.

Cardarelli: I have to stop you for a second. What you saw was just a mock-up. It was the training of the subjects. That photo you are referring to was taken while the subjects were being trained to reduce the learning effect. It is after this training that the subjects are tested. During testing they wear full PCs. the subjects are tested in PCs twice, once with a respirator and once without a respirator.

Westbrook: My second point is that I assume your results are for people wearing full PCs, but one hears from every site, and practically every nuclear facility and for every nuclear activity that there is, that they do notice a significant difference when people wear respirators. If you are finding skinny results like 15-20% difference, that would seem to run counter to this huge bulk of anecdotal experience over time and numbers of people.

Cardarelli: I have looked at tons of those studies, and with the exception of a Canadian study, I have not found one study with human testing that significantly quantified worker efficiency. I was leery to even show the 10% at this time because this is only based on six people, and starting in the next couple of weeks we will be doing it with 50 people. But in 1985, when I did the study at TMI, there were 48 workers, and there was no significant respiratory effect for the 25-minute task used in that study. You watch them do the two tasks and you say, "I know they will do it slower with a respirator," but the video tape and the clock really didn't show that. In a lot of these informal studies conditions with and without the respirator are not the same. For example, last year they had heavy PCs and respirators on. This year they didn't have a respirator, but they also didn't have all those PCs on because they engineered and sandblasted. It is very tough to use those studies, and that is why we are trying to do a detailed study and really just focus on the respirator effect.

Khan: We did similar work, as you know, in Canada, using Canadian attire, and we found results that tend to agree with yours. The respirator had a small effect. But the largest effect was the type of gloves that they wore. The heavier the gloves, the more inefficient they got. Did you look into that aspect at all?

Cardarelli: We are training the subjects with double rubber gloves because the effects that the rubber gloves have on efficiency are probably greater than those from the respirator. We are going to pretrain them and try to remove that variable since this study is looking just at the respirator effect.

FIELD EXPERIENCE WITH REMOTE MONITORING

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ABSTRACT

The Remote Monitoring System (RMS) is a combination of Merlin Gerin detection hardware, digital data communications hardware, and computer software from Bartlett Services, Inc. (BSI) that can improve the conduct of reactor plant operations in several areas. Using the RMS can reduce radiation exposures to radiation protection technicians (RPTs), reduce radiation exposures to plant maintenance and operations personnel, and reduce the time required to complete maintenance and inspections during outages. The number of temporary RPTs required during refueling outages can also be reduced. Data from use of the RMS at a two power plants are presented to illustrate these points.

EQUIPMENT CONFIGURATION

Both sets of data presented here were obtained during outage activities at PWRs. One case involved cutting and removal of RTD piping and the other involved ISI of steam generators. In both cases, one WRM radio receiver was placed in containment. Three of four LEA boosters insured radio coverage of the entire containment. Display terminals were placed at RP control point outside containment. A computer multiplexor in containment connected all of the in-containment devices to the main computer located at the main RP control point. Thus, only a single data cable was required to connect the devices in containment to the outside. The RP control points inside containment were also equipped with closed circuit video equipment and the RPTs had voice radio communication with the workers in the loop areas. The workers wore four or five transmitting dosimeters connected to a multiplexer and transmitter. Dosimeters were also mounted on the steam generator platforms to provide area monitoring for the ISI work.

DOSE SAVINGS TO RPTs

The use of the RMS allowed RPTs to control the workers' radiation doses while working in a low dose area. In the case of RTD pipe cutting, the RPTs were able to work in a 2 mrem/h field while the pipe cutters were in a 500-1000 mrem/h field. Dose saved was estimated to be approximately 30 rem. In the case of ISI, the RPTs were similarly working from a low dose area outside the bioshield wall. The use of telemetry for area monitoring also allowed RPTs to perform routine surveillance from a low dose area.

DOSE SAVINGS TO WORKERS

The use of RMS allows RPTs to accurately monitor dose and dose rate in situations that would not be possible with traditional means of monitoring. Whenever a worker is near collimated sources, point sources, or line sources, monitoring of multiple whole body locations is necessary. If the work activity requires the worker to change position near the source, it is extremely difficult to track multiple whole body locations with time and motion techniques. Therefore, stay time limits are imposed to control dose. Although stay time limits are effective in limiting dose, this technique tends to require multiple trips into the high radiation area to complete the work, because the stay time limit is necessarily conservative. Also, the use of stay times does not, by itself, control the worker's dose rate.

The RMS allows an RPT to control the dose rate during the work activity, even when dose rates are rapidly changing and the worker has multiple dosimeters. The RPT also monitors the dose to each dosimeter, thereby allowing a worker to complete the job with a single entry to the high radiation area in many cases.

Figure 1, for example shows data from RTD pipe cutting. The worker placed his right are close the RTD pipe. The RP technician noticed a dose rate alert at 3000 mrem/h and directed the worker to move his arm away from the pipe. The RPT was located outside the work area, but was able to control the radiological conditions using video, voice radio, and RMS.

Figure 2 shows a more complicated series of events while "Worker T" was cutting RTD pipe. Shortly after beginning the cut, he moved his head and right arm closer to the pipe. The RPT noticed the increased dose rates on the RMS monitor and requested that he move back. The worker complied immediately. He subsequently kept his head back from the pipe, but later his right arm crept back to 1000 mrem/h again. The RPT again requested that he move his right arm back and continued to monitor the right arm closely during the remainder of the cut.

Figure 3 also shows how RMS can control the dose rate during a high dose rate job. Here, "Worker H" was being monitored during RTD pipe cutting. This worker moved both thighs close to the pipe. Again, the RPT, monitoring remotely, observed an unnecessarily high dose rate to the left thigh and called for the worker to move back from the pipe. The worker moved his left thigh back from the pipe for the remainder of the time that he was cutting. However, the right thigh began to edge in closer to the pipe. When the right thigh exceeded 1000 mrem/h, the RPT called for the worker to move the right thigh, which was promptly moved away from the pipe. Then the worker moved his right arm close to the pipe. This movement was also detected and the worker moved his right arm to a lower dose rate.

Figure 4 shows data from monitoring "Worker M" during RTD pipe cutting. This worker would alternately move his right thigh, his left thigh, or both thighs closer to the pipe. The RPT repeatedly monitored the dose rate to all five dosimeters on this worker and adjusted the positions of the worker's thighs to reduce the dose rates. These data demonstrate conclusively that an RPT can reduce a worker's dose by detecting high dose rates during a work activity and reducing the dose rate as the work is in progress. The most important method of reducing a worker's dose is to ensure that the worker does not inadvertently place his or her body in a radiation field that is unnecessarily high. Real time monitoring allows the RPT to reduce the worker's dose because the high dose rates are detected while the worker is in the high radiation area.

In situations that require exposure control via stay time limits, the RMS can constantly monitor dose and dose rate so that workers can complete a task in one entry without exceeding administrative limits. This results in fewer entries to the radiation area and less dose received while entering and exiting. When multiple whole body dosimeters are required, the RMS provides a tool to help the RPT prevent one dosimeter reading significantly higher that the others. This reduced the overall whole body dose.

The RMS can also reduce overexposures due to inadequate surveys. The common ionization chamber survey meter will significantly underestimate the dose rate from point sources at distances less than 15 cm. This occurs due to the well known geometry effect. The RMS' detectors are more accurate under such conditions because the detector is a small silicon chip, similar to the size of a TLD chip.

PREVENTING OVEREXPOSURES

Figures 5 and 6 are examples of how RMS can prevent overexposures. Figure 5 shows the case where a worker on a steam generator platform made an unauthorized partial entry into the steam generator bowl. The unauthorized dose rate to his arm was rapidly detected as the RMS monitor alarmed and the worker was removed from the platform.

In Figure 6, the data demonstrates a case where a worker on a steam generator platform was authorized to make a partial entry (arms only) into the bowl. However, this individual decided to make a full entry into the bowl. Consequently, the detector on his head registered 16,200 mrem/h. The immediate response of the RPTs prevented a possible overexposure.

SAVING OUTAGE TIME

The time required for maintenance or inspection activities can reduced because fewer entries are required into radiation areas. If the radiation area is also contaminated; if respiratory protection is required; or if ALARA briefings are required, the time required for an entry can easily be three or four times greater than the time required to perform the physical work. Reducing multiple entries can remove days from an outage schedule.

Figure 7 gives an example where an individual made two successive full entries into a steam generator bowl. The RMS was used to carefully control the individual's dose to within 200 mrem of the regulatory limit. Having one individual perform two successive entries resulted in a lower overall dose because it saved the dose associated with multiple trips to the steam generator platform. This procedure also saved the dose to support personnel who would have accompanied the worker into the loop area for multiple entries. In addition, the work was completed in less time.

The earlier examples of RTD pipe cutting also demonstrate an outage time savings. In this case, all of the RTD pipe was cut in one work shift. This time savings occurred because the workers were able to complete multiple cuts in each trip to the work location. Since the workers were being monitored with RMS, there was no need to come down from the platform and check "sacrificial" dosimeters after each cut. Also, sacrificial dosimeters can overestimate the TLD reading because a PIC or EDRD on the outside of the protective clothing can hang down from the worker's chest and be much closer to the radiation source than the TLD.

In these examples, the RPTs were working from control points in containment. However, some plants have used RMS to move their control points completely outside containment. This can allow the RPT to work in a relatively quiet location without contamination or respiratory protection controls. This environment can improve coordination between the RPT and other work groups so that work flows without interruption. The productivity of the RPT can also be increased by working from a control point outside containment. Of course, using RMS, one technician may be able to cover work in multiple locations.

CONCLUSION

In addition to the obvious fact that using the RMS can save dose to RPTs, there are significant opportunities to reduce workers' doses, reduce outage time, and reduce outage budgets.

Author Biography

Dr. Arthur E. Desrosiers, CHP has developed computer systems to provide protective action recommendations for the nuclear Regulatory Commission's emergency operations center and he developed the first completely automated radiological access control system. Dr. Desrosiers designed the Dry Active Waste process controllers marketed by Westinghouse in the 1980. Recently, as Vice President of Special Projects at Bartlett Services, Inc., he has developed the Remote Monitoring System. This system collects data via radio transmitters for the control of radiological exposures. The RMS has been selected by 20 commercial nuclear sites and one DOE site.

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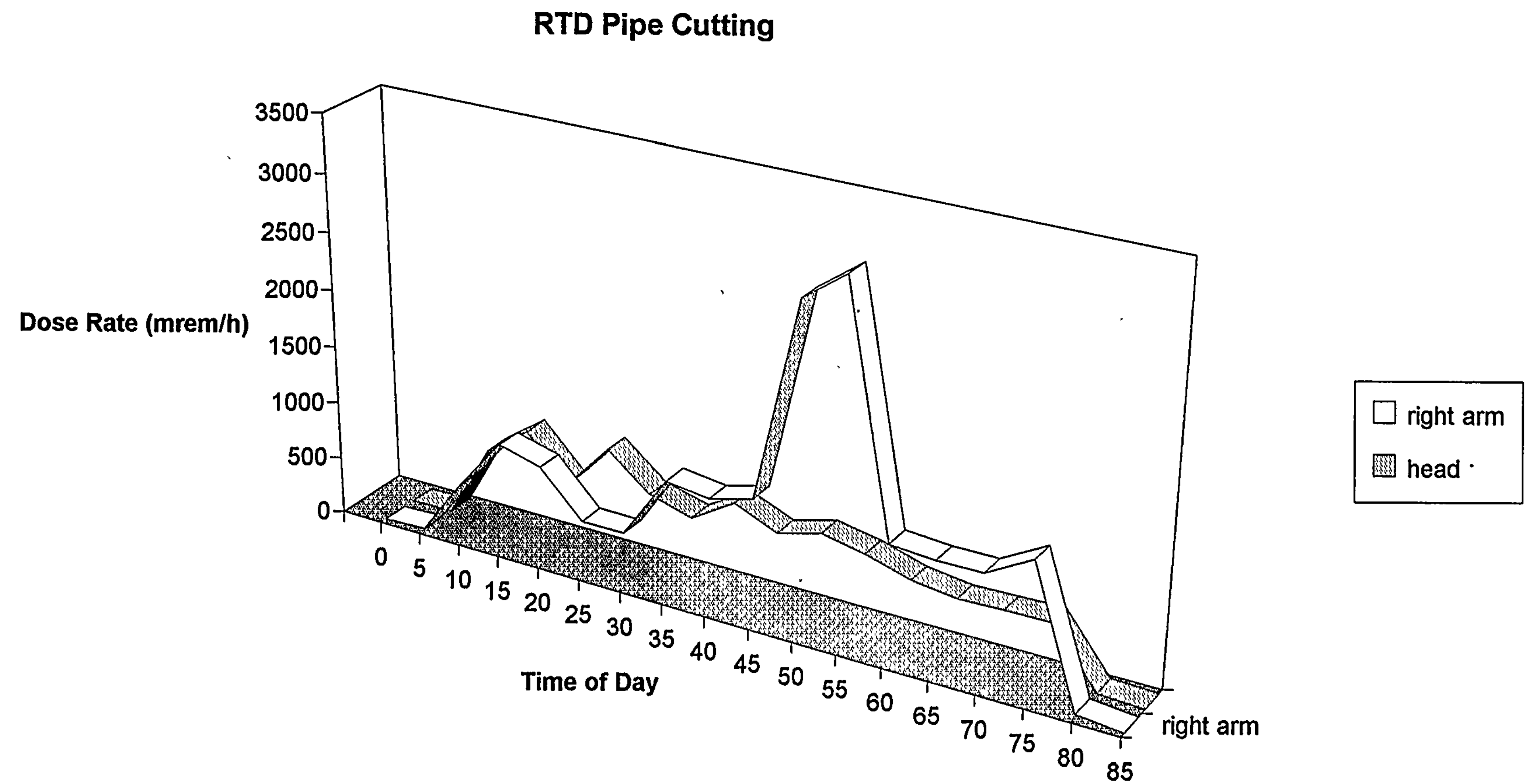

Figure 1. **<u>RTD Pipe Cutting</u>.** This example shows a worker who placed his right arm close to the RTD pipe. The RP technician noticed a dose rate alert at 3000 mrem/h and directed the worker to move his arm away from the pipe. The RP technician was located outside the work area, but was able to control the radiological conditions using video, radio communications and RMS.

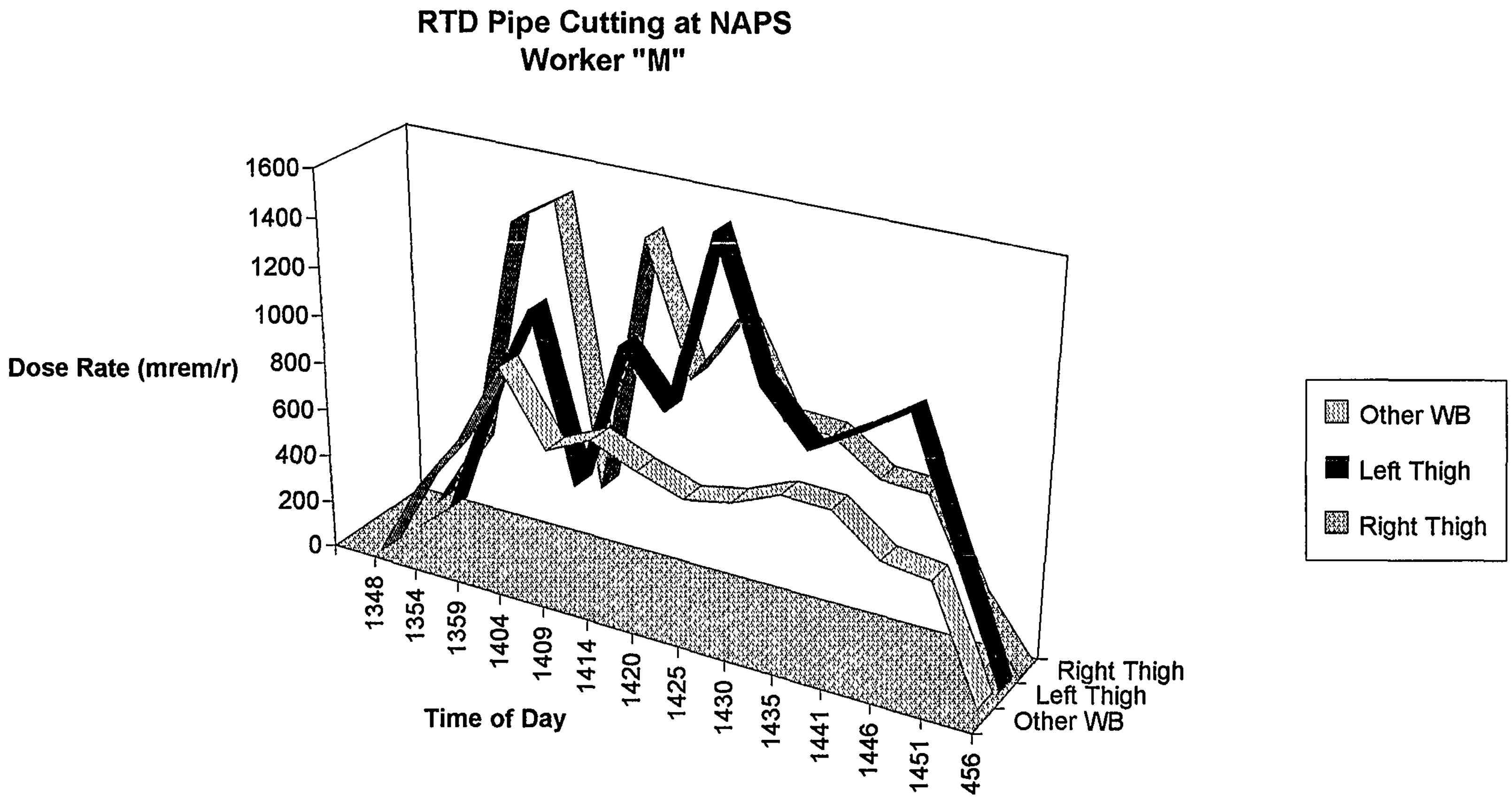

Figure 2. **"Worker M"**. "Worker M" shows a tendency to move his right thigh close to the RTD pipe. After three rounds of requests from the RP technician who was monitoring with RMS, the worker adjusted to a position where the right thigh received the same dose rate as the other whole body locations.

At this point, however, the worker moved his left thigh close to the pipe. The RP technnician subsequently made two requests to lower the dose to the left thigh.

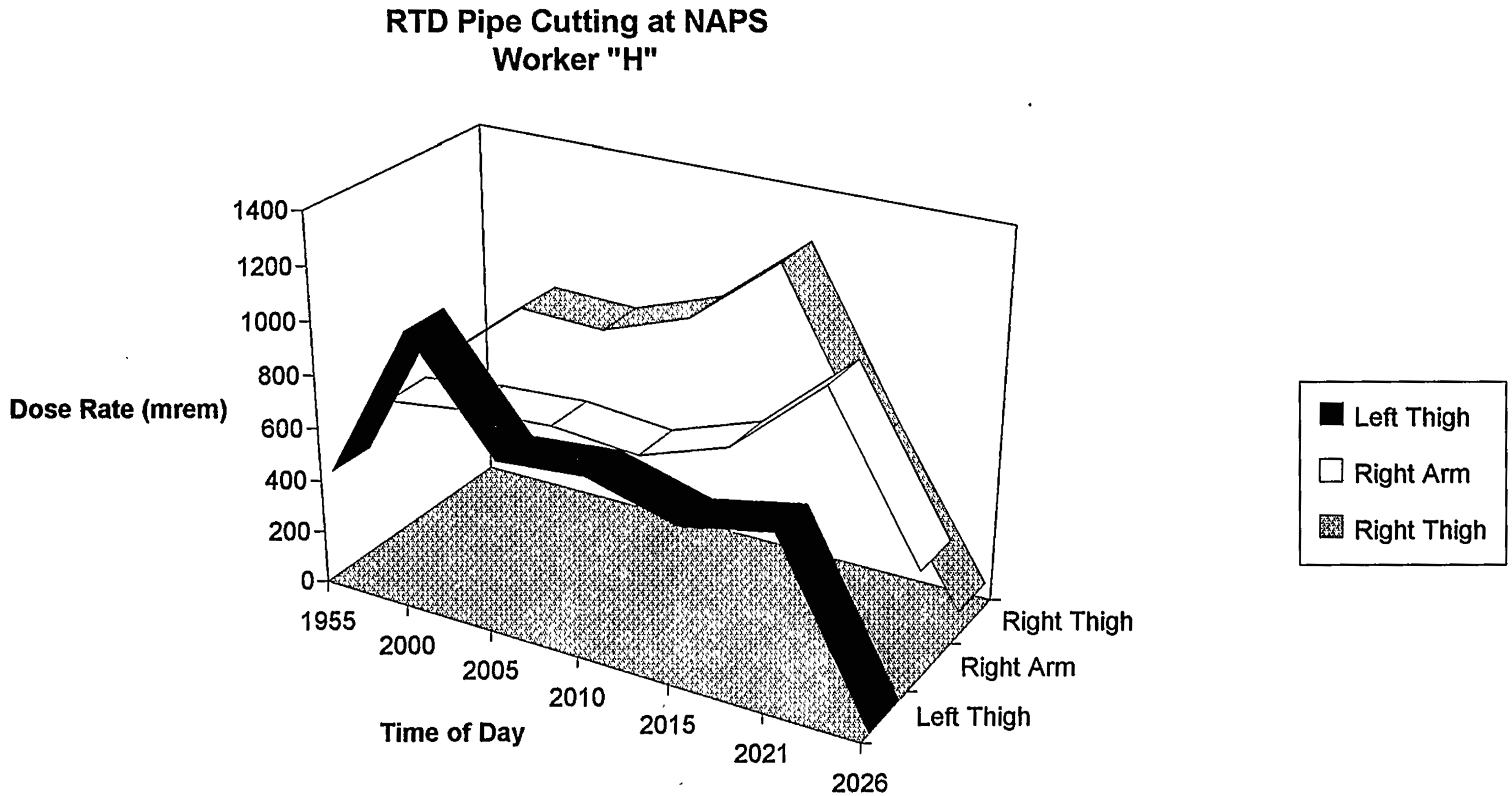

Figure 3. **"Worker H"**. "Worker H" moved his left thigh, right thigh and right arm close to the pipe at different times during the cut. Each time, the RP technician who remotely monitored the job detected the increased dose rate and caused the worker to adjust his position.

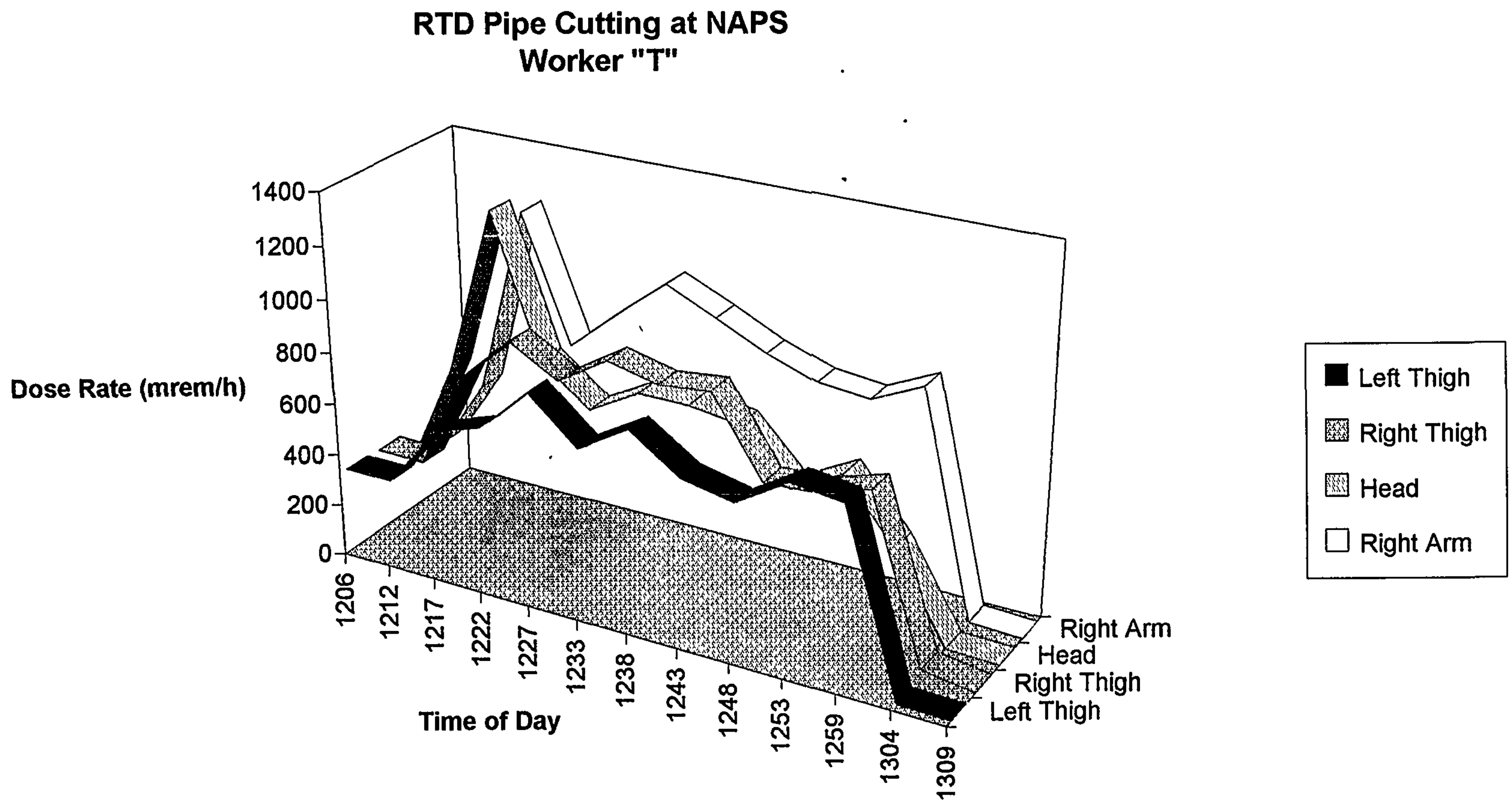

Figure 4. **<u>"Worker T"</u>**. "Worker T" placed his head and right arm near the pipe shortly after the work began. The RP tech requested that he move back. The worker complied immediately and moved his head back. The RP monitored the right arm's dose rate and made two subsequent requests to move the arm back as the dose rate to the arm increased.

This constant vigilance by the RP tech kept all six whole body TLDs at approximately the same dose so that the worker did not "burn out" due to a single body location being much higher than the other locations.

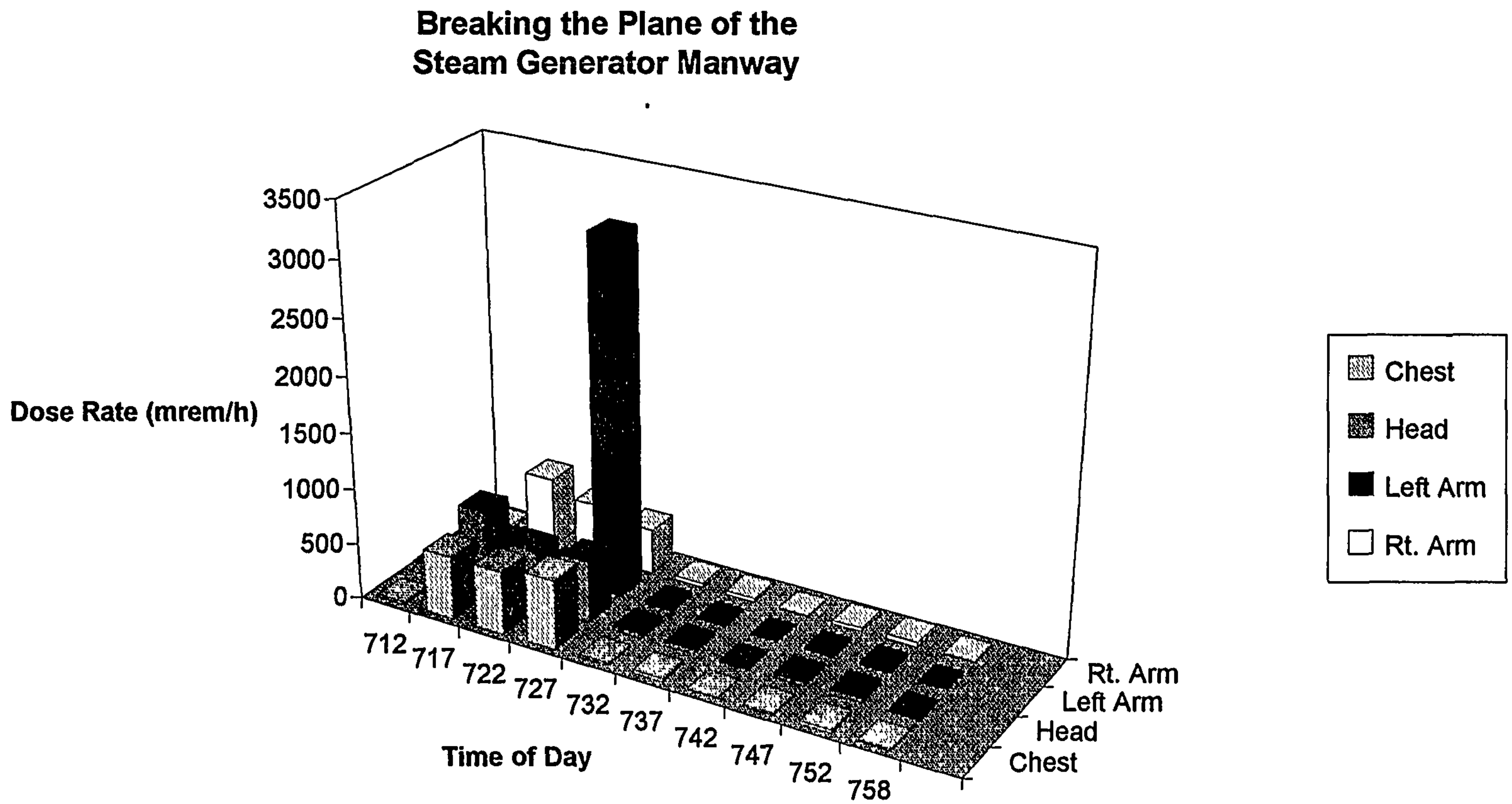

Figure 5. __Unauthorized Partial Entry__. The RWP allowed the worker on the steam generator, but did not allow entry into the steam generator bowl. However, the worker's left arm broke the plane of the generator bowl after about 10 minutes of activity on the platform. This event was rapidly detected by the remote health physics technician and the worker was removed from the platform shortly thereafter.

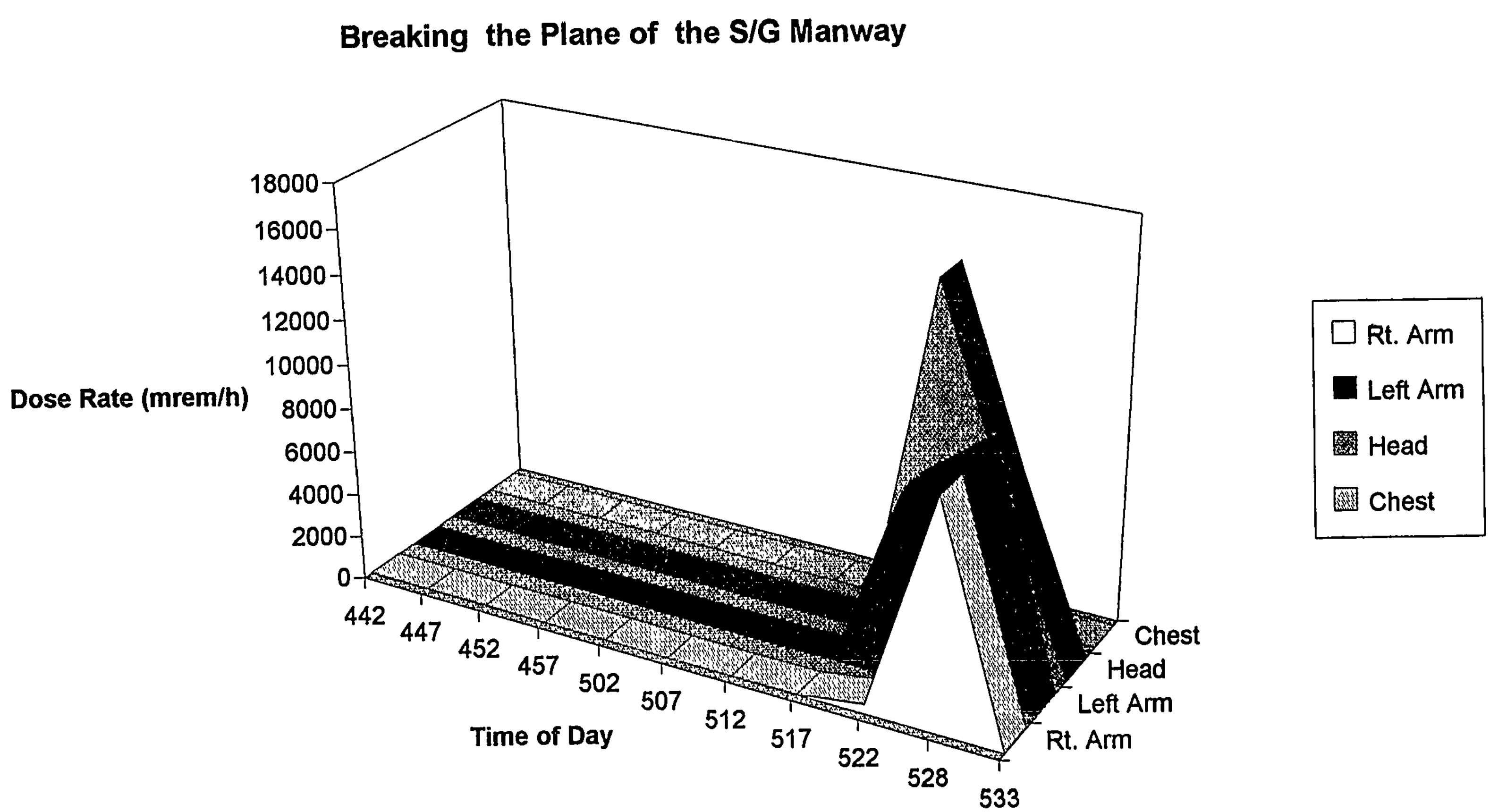

Figure 6. <u>**Unplanned Full Body Entry In A Steam Generator Bowl**</u>. In this case, the worker was scheduled to make a partial (arms only) entry into the steam generator bowl. As you can plainly see (and as the health physics technician also plainly observed), the worker put his head into the bowl first (16.2 R/h). This unplanned event was rapidly detected and the worker removed his head per instruction of the health physics technician. The work was completed as a partial entry. They were able to complete the entry because the dose to the head was being monitored via telemetry. Therefore, the dose due to the unplanned exposure to the head was evaluated as soon as the worker removed his head. Otherwise, additional dose and time would have been expended in descending and reascending the platform.

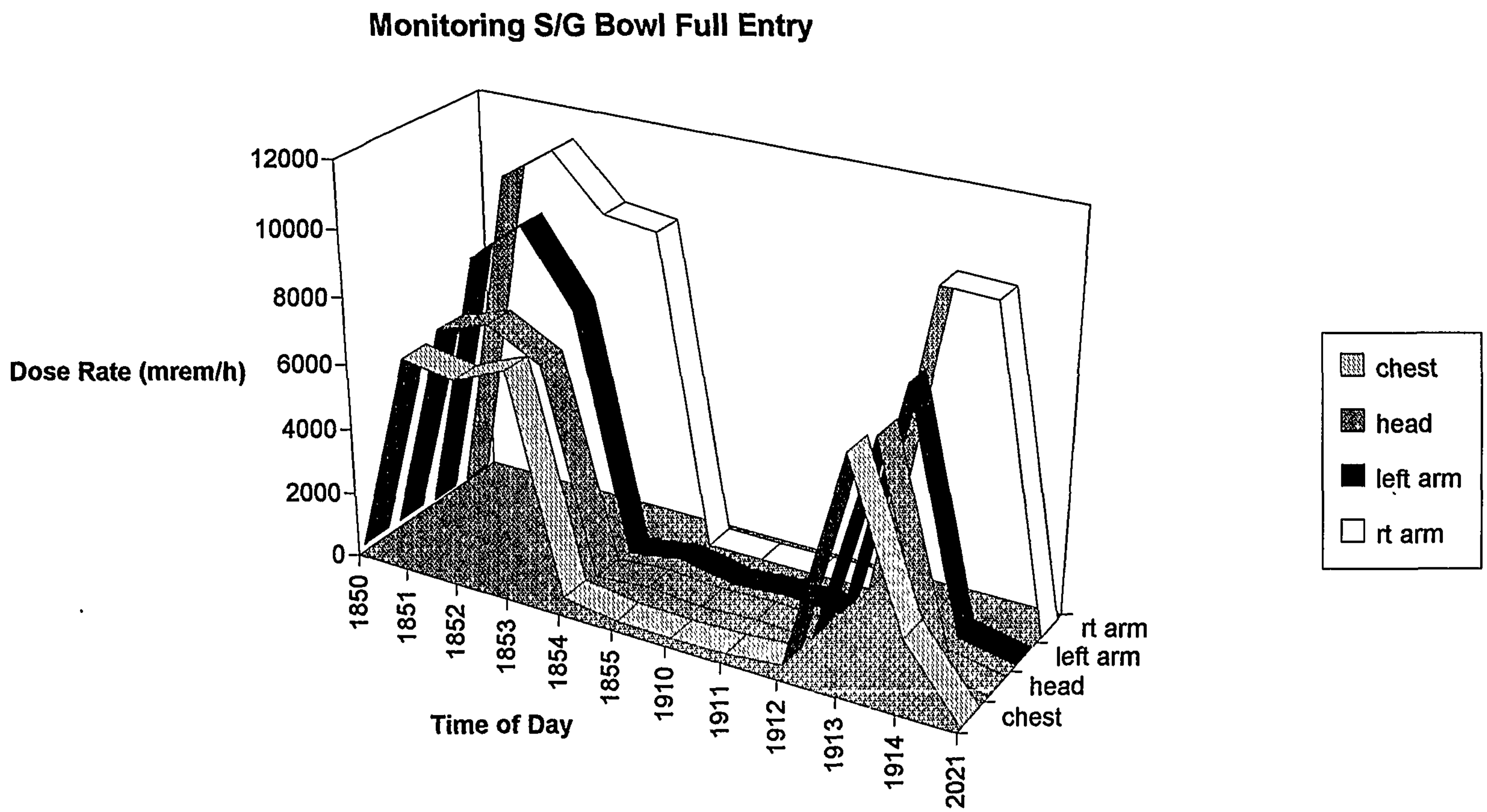

Figure 7. __Two Consecutive Full Body Entries__. The work scope required cleaning of the channel head. In order to minimize the total dose for the job, one person performed all the cleaning in a single evolution. The RMS monitored the worker while he was in the bowl. The dose and dose rate of each dosimeter were reported every 16 seconds. The health physics technicians could read the accumulated dose from the first entry as soon as the worker exited the bowl. In 18 minutes, the worker entered the bowl again to finish the job. There was no overexposure and the two jumps were essentially completed in a total of 24 minutes.

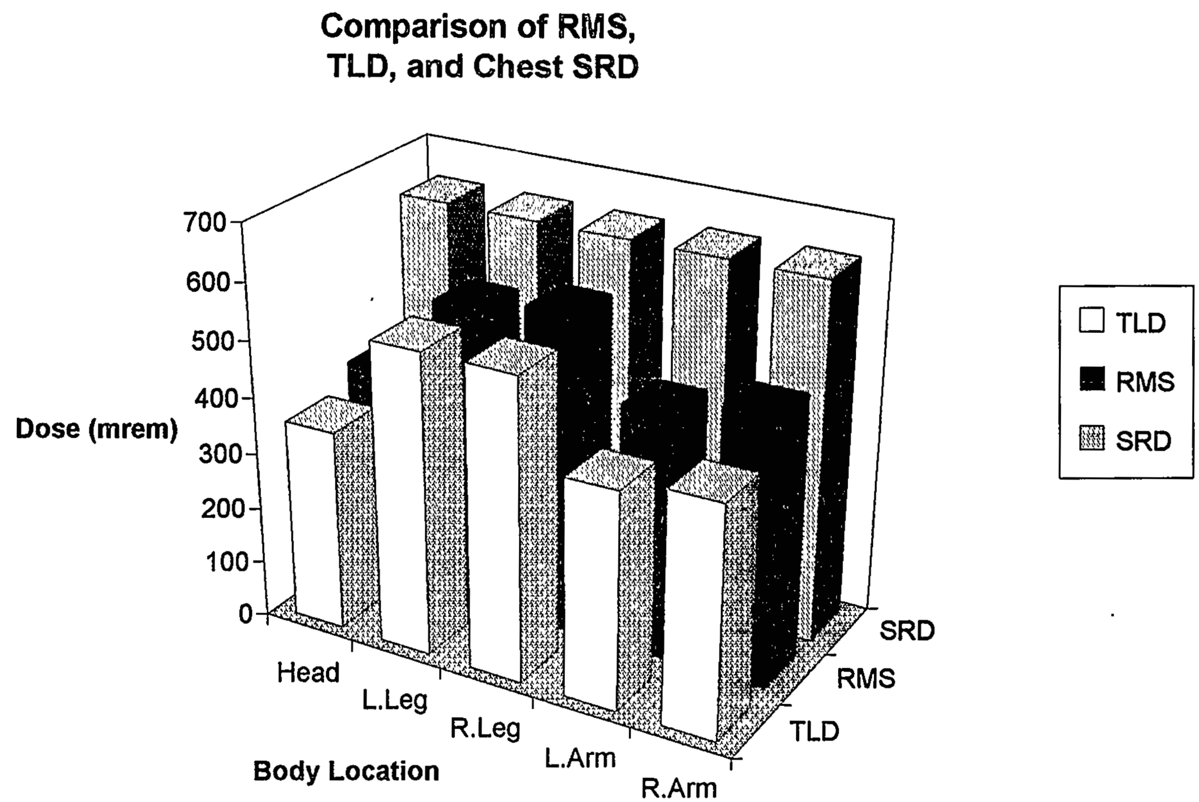
Comparison of RMS,
TLD, and Chest SRD
Dose (mrem)
700
600
500
400
300
200
100
0
Head
L.Leg
R.Leg
L.Arm
R.Arm
Body Location
SRD
RMS
TLD
TLD
RMS
SRD

Figure 8A

Comparison of RMS, TLD, and Chest SRD

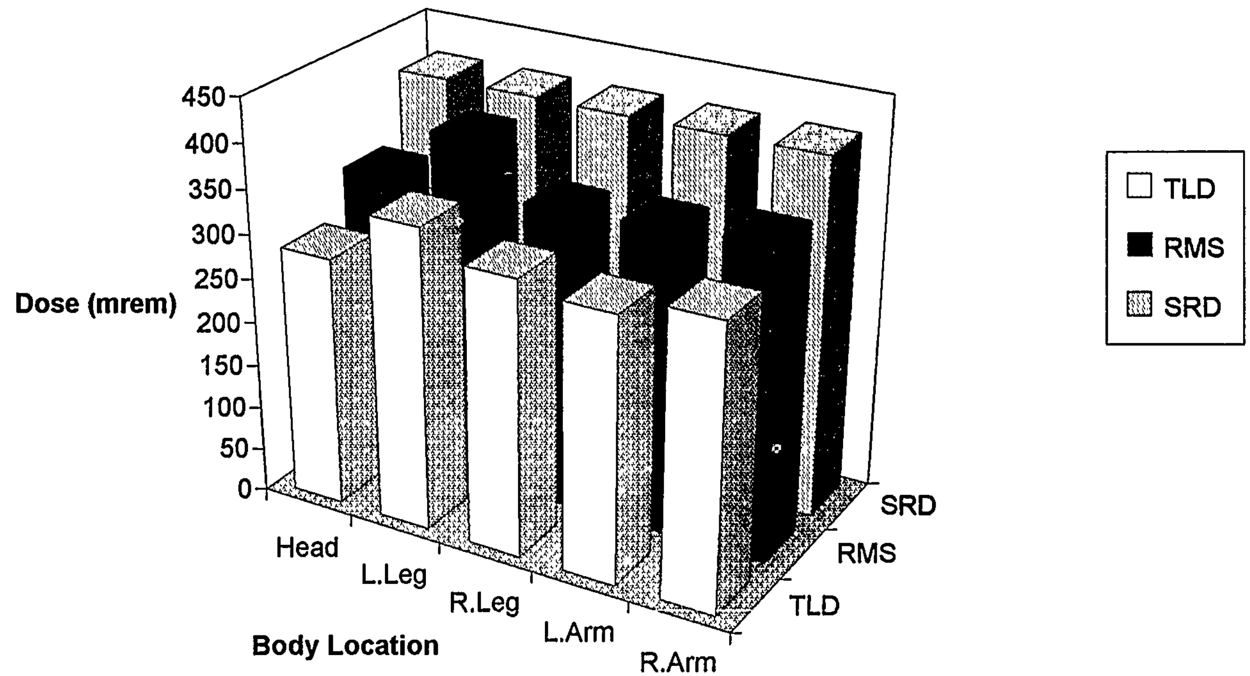

Figure 8B

WESTINGHOUSE CORPORATE DEVELOPMENT OF A DECISION SOFTWARE PROGRAM FOR RADIOLOGICAL EVALUATION DECISION INPUT (REDI)

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ABSTRACT

In December 1992, the Department of Energy (DOE) implemented the DOE Radiological Control Manual[1] (RCM). Westinghouse Idaho Nuclear Company, Inc. (WINCO) submitted an implementation plan showing how compliance with the manual would be achieved. This implementation plan was approved by DOE in November 1992. Although WINCO had already been working under a similar Westinghouse RCM, the DOE RCM[1] brought some new and challenging requirements. One such requirement was that of having procedure writers and job planners create the radiological input in work control procedures. Until this time, that information was being provided by radiological engineering or a radiation safety representative. As a result of this requirement, Westinghouse developed the Radiological Evaluation Decision Input (REDI) program.

INTRODUCTION

During a March 1993 Defense Nuclear Facility Safety Board (DNFSB) visit to the Idaho Chemical Processing Plant (ICPP), WINCO was questioned about how the expertise of other Westinghouse organizations was being utilized in the development of programs required by the DOE RCM.[1] Although some examples were given, WINCO felt that more involvement on a corporate-wide level could provide great savings throughout the Westinghouse organization. Soon after that visit, a corporate-wide committee was organized to develop a program for the development of Radiological Work Packages.

WesTIP Team

Because the DNFSB had questioned the use of Westinghouse expertise from other sites in RCM[1] implementation, WINCO officials began to review items which were to be implemented in the near future for the possibility of corporate involvement. The Radiological Work Package process was a very good possibility, and was eventually chosen for corporate committee review.

WINCO sponsored the committee known as the Westinghouse Technologies to Improve Processes (WesTIP®) Combined Team Review. The goal of the committee was to reduce the costs and time associated with development of Radiological Work Packages while improving quality and consistency. WINCO, West Valley Nuclear Services (WVNS), Westinghouse Savannah River Company (WSRC), and Westinghouse Hanford Company (WHC) chose to participate in the project, and selected members with appropriate expertise for the team. This committee then met in Pittsburgh, Pennsylvania to begin learning the WesTIP® Process and begin applying it to the creation of radiological work packages.

The first step in the WesTIP® Process was to define the current process being used. To do this, the committee members outlined the process being used by their individual facilities and then the processes were combined to get an overview of the basic process being used within the Westinghouse Complex. Problems associated with each step were discussed, and those which were deemed to be significant were then listed on the flow chart model of the process.

Once the overview was completed, each step in the process was reviewed to see how much time it took and how much it cost to complete. Costs for individual steps ranged from $25 to $1,625 and time spent ranged from 0.1 day to 5 days per step. Those steps which cost the most or took the most time were then highlighted on the flow chart and reviewed to see where reductions could be made.

Findings revealed that it was taking an average of 37 days and costing an average of $6,875 to complete one radiological work package. Using the WesTIP® technique, the committee developed a plan which would allow a radiological work package to be completed in 7 days and cost $2,480. This established a time reduction of 30 days and a cost savings of $4,395 per work package (Figure 1). Because of the number of work packages created in a year, the committee determined that significant yearly savings would be realized through the use of the program they proposed. A key point to implementation of the new program was the need for a risk-based decision tree process for creating and completing radiological work packages.

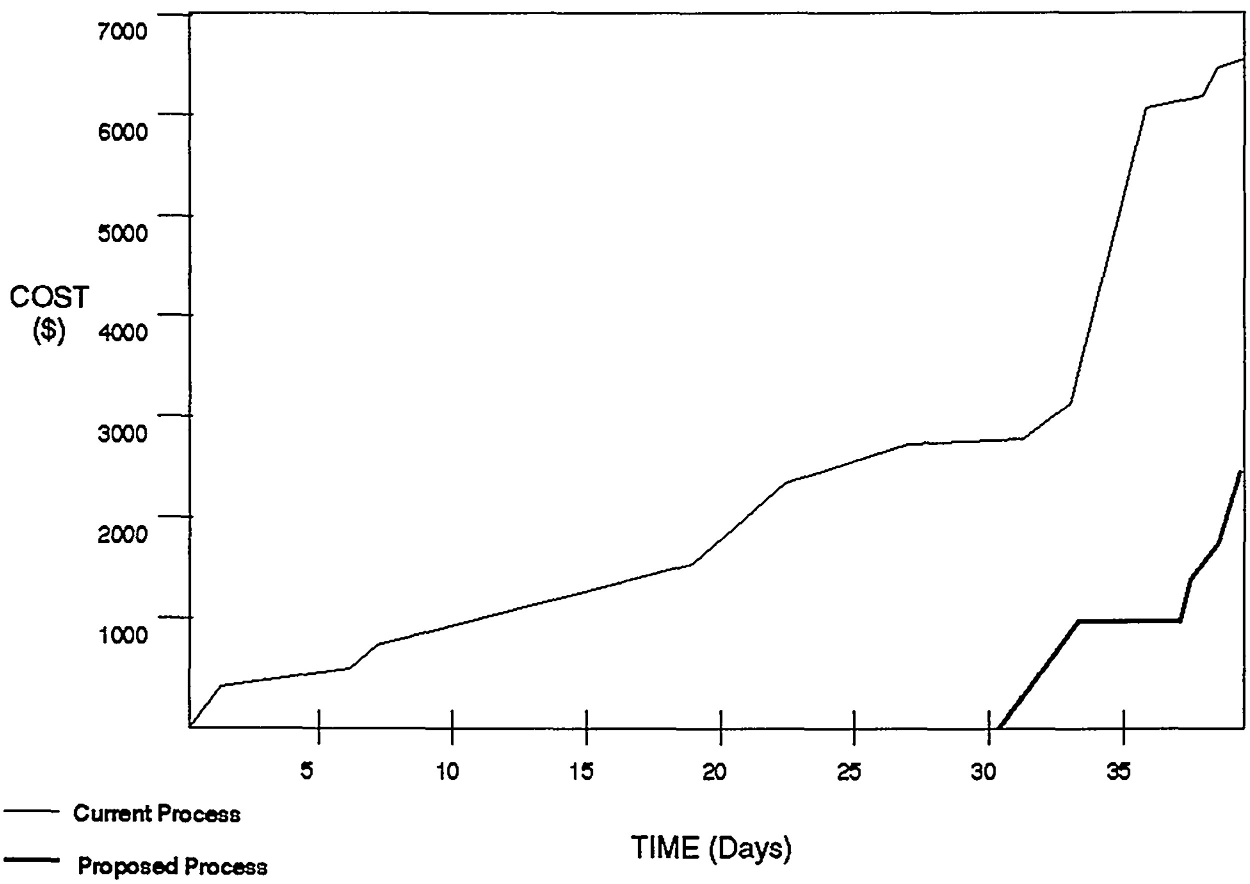

Figure 1 - Work Package Costs Before and After Process Modification

The committee discovered that gathering data from various resources was the largest consumer of both time and money. Much of the data gathered was standard information which was recreated every time a radiological work package was needed. This collection of information was spread over several different steps in the original process.

With the decision made that the goal of the group would be to reduce time and costs associated with gathering information and writing work packages, the work of improving the process could begin. Several changes to the process were proposed, including creating "points of contact" who would function as area experts and creating a decision tree program which would eliminate recreation of information for every work package.

The committee decided that approximately 80 percent of the information recreated every time a radiological work package was developed could be placed into a decision tree software, thus eliminating the need for recreation of that information. This information included Radiological Control Manual[1] requirements, Federal Regulations, common work standards and policies, and local procedural requirements.

Development of REDI

Once the WINCO representatives returned from Pittsburgh, they began to explore hardware/software resources and availability. They made the determination that DClass®, a commercially available software, would be used to develop their decision tree. This determination was based upon applicability, adaptability, and cost. Because DClass® was already being used by WINCO personnel who would be doing the programming, a significant cost savings was realized.

The first step in development of the REDI program was to produce a risk-based decision tree which could be input into DClass®. To accomplish this, a number of radiological packages were reviewed to determine the questions that must be answered in order to create a valid work package. The answers to these questions were not the same in all cases, and were dependent upon the specifics of the work to be completed. With this in mind, multiple choices were designed to cover all probable answers to a question. These choices correspond with precise output devised to be used for final work package details.

Prior to the development of the REDI program, planners and procedure writers would normally solicit information from a number of resources. With the use of the REDI program, the information could be automatically compiled by simply answering area-specific questions. By answering such questions, the requirements could be narrowed for the specific job to be performed. The program was to be designed to automatically create radiological input containing the appropriate guidelines. This input would then be available for use in creating radiological work package portions of procedures.

The REDI program asks the user multiple choice questions. Based upon the answers to those questions, appropriate information is placed into a radiological input package (Figure 2).

Features of REDI

One of the main features of the REDI program is the capability it offers for electronic review and approval of radiological work packages. REDI is installed on a network with access by all individuals in the review and approval process. When the package is ready for review, it is electronically transferred. When the appropriate personnel have reviewed the package, they enter a password which allows the package to be sent back to the originator with comments or approval. If changes are made to the package, prior approvals

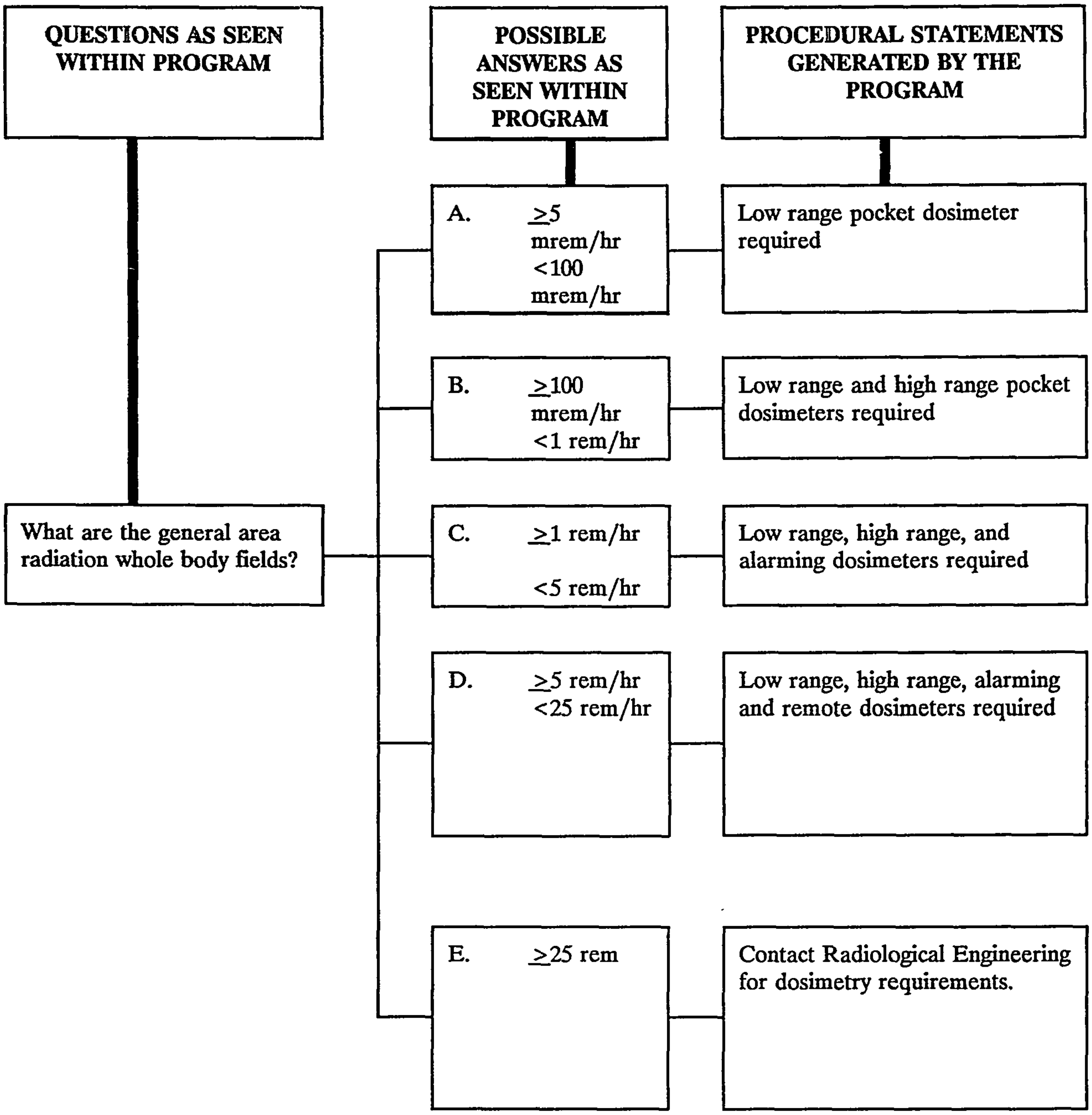

Figure 2 - Sample Decision Tree

are voided and the package is retransmitted for approvals. Any forms associated with the work evolution such as radiological work permits can also be attached and forwarded electronically.

The approved package is returned to the document originator who then places the package in the work control procedure for use in the field. The REDI cycle is not yet complete; however. Post-job critique information is entered as "lessons learned" before the work package can be closed out and considered complete.

All packages are stored electronically. Each package has its own unique file name which allows retrieval for review or modification at any time. For future referencing, packages can be retrieved using the file name assigned to the package. The information contained in this package can be used for future work packages, either in its entirety or by removing pertinent sections. "Lessons learned" will be of great value when planning for future work. Users can refer to previously developed radiological work packages to retrieve pertinent information for the development of the current work package.

CONCLUSION

Development of REDI is an ongoing process. It is currently being used at WINCO, and will be released to other Westinghouse GOCOs when the initial testing phase is complete. It will be customized for each facility, and will likely be released for other DOE facilities shortly after delivery to Westinghouse.

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1. U.S. Department of Energy, *DOE/EH-0256T Radiological Control Manual*, June 1992.

Author Biography

T. Shane Bush is a Sr. Technical Specialist at the Idaho Chemical Processing Plant located on the Idaho National Engineering Laboratory near Idaho Falls, Idaho. Shane currently works in the Radiological Engineering subsection of the Environment, Safety, and Health Department. Before joining Westinghouse in 1984, he worked at numerous commercial nuclear power plants providing health physics support during refueling outages. His primary responsibilities include coordinating implementation of the DOE Radiological Control Manual, Assistant Radiation Safety Officer for radiography operations, and development of the REDI (Radiological Evaluation Decision Input) program which he will be speaking about today. He has an Associate of Applied Science Certificate in Radiation Safety, is a registered radiation protection technologist, and has a Bachelor of Science degree from Idaho State University.

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TASK RELATED DOSES IN SPANISH PRESSURIZED WATER REACTORS OVER THE PERIOD 1988-1992

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ABSTRACT

In order to evaluate in depth the collective dose trend and its correlation with the effectiveness of the practical application of the ALARA principle in Spanish nuclear facilities, and base the different policy lines to promote this criteria, the CSN has fulfilled an analysis of the task related doses data over the period 1988-1992. Previously, the CSN had required to the utilities the compilation of their refuelling outage collective dose from 1988 according with a predeterminate number of tasks, in order to have available a representative and retrospective set of data in an homogeneous way and coherent with the international data banks on occupational exposure in NPP, as the CEC and the NEA ones. The scope of this analysis was the following: first, the collective dose summaries for outage tasks and departments for PWR and for BWR, including the minimum, maximum and average dose (and statistics data) for 18 different refuelling outage tasks and 12 personal departments for each generation of each type of rector, the task and department related collective dose trends in each plant and in each generation, and second, the dose reduction techniques having been used during that period in each plant and the relative level of adoption. In this presentation the main results and conclusions of the first part of the study are reviewed for PWR.

INTRODUCTION

The trend in average occupational collective dose per reactor in Spanish Pressurized Water Reactors (PWR) seems to be stabilising and lightly decreasing during the studied period of time, 1988-1992, within the same range of the average levels in the OECD countries (figure 1).

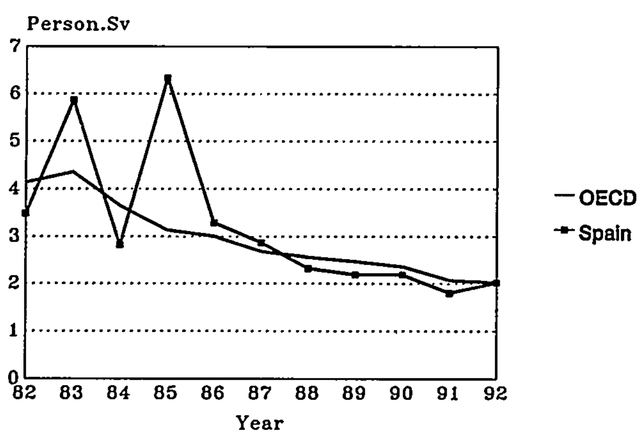

Figure 1. Average annual dose for Spanish and OECD PWR

A tendency towards stability because in 1985 the backfitting operations in J. Cabrera, the first generation reactor, finished and this led to very high peacks in 1983 and 1985. A tendency towards a moderate decrease, because on one hand the second generation reactors broke their fast rise of the early years of operation and, on the other hand, because in 1988 the starting up of the third generation reactors, Vandellós II and Trillo, took place, incorporating more ALARA design characteristics than the formers.

Observing the collective dose per unit of electricity produced evolution, the first generation reactor values remain higher than average both in Spain and in OECD countries, this last being 0.3 Person.mSv/Gwh in 1991*, while the second generation presents average values near the OECD ones and the third has always been under it.

TASK RELATED COLLECTIVE DOSE RESULTS

In order to know the radiological relative weight of each task in which the refuelling outage operations have been distributed, according with the CSN guide 1.5, the average, maximum and minimum for each task has been figured out in this study. In addition, statistics data, as standard deviation and the variation coefficient has also been calculated to have an estimation of the dispersion of the results. Establishing a selection criteria of 5% of the total dose, i.e. around 100 Person.mSv, we have focused this presentation in those tasks whose average contribution exceed this value (table 1).

Table 1. Task related dose results for all PWR generations

Task	Average Person.mSv	%	Maximum Person.mSv	Minimum Person.mSv	Standard Deviation	Variation Coefficient
Steam gen. primary side	588	28	1676	30	406	0.69
General works	274	13	755	44	188	0.68
Refuelling	210	10	497	32	121	0.58
System not listed	205	10	618	1	202	0.98
Valve work	157	8	322	21	81	0.51
Insulation	123	6	384	15	95	0.75
Reactor coolant pumps	104	5	535	2	107	1.03
Routine inspections	97	5	259	20	58	0.6

Depending on the special characteristics of each generation and reactor design, the size (only J. Cabrera is significantly smaller) and the aging effect in the source term, among other factors, the order of this list of relevant tasks is different in each generation (figure 2).

On the other hand, the dispersion of the values for each task within each generation reflects the variety of the scope of the different works included in each one, from one outage to another and from one reactor to another, depending on the inspection requirements, the incidents registered during the operation cycle, the ambient dose rates in working areas, systems and components involved and job procedures. In any case, the variation coefficient for a task can be a rough indication of its relative dose reduction potential (ref. 2).

*This data and the OECD values of figure 1 are taken from reference 1

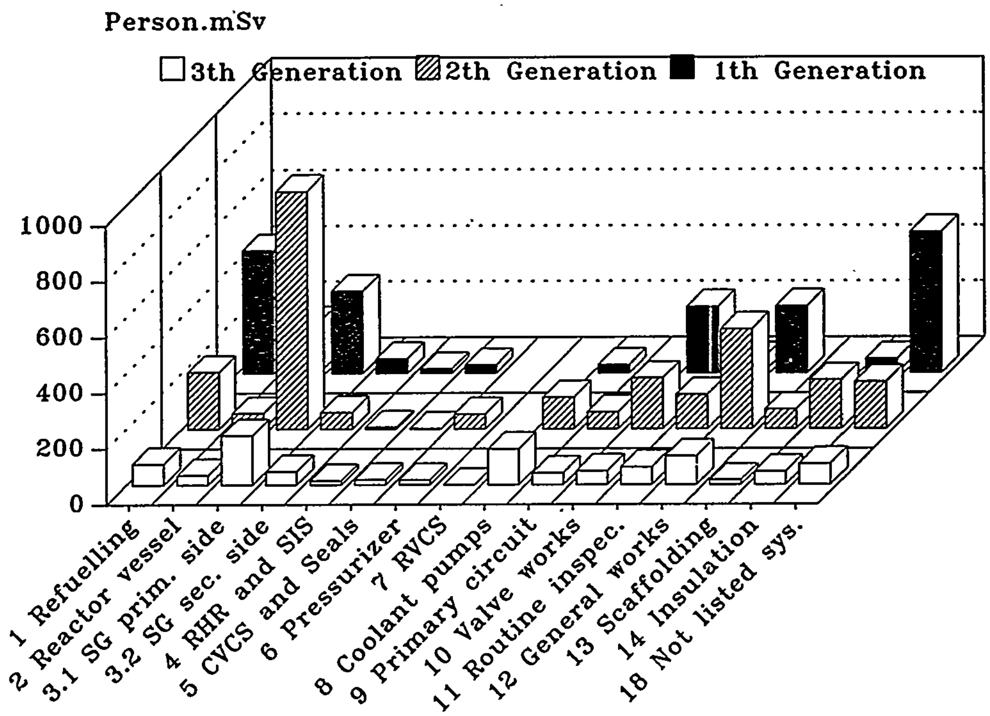

Figure 2. Average collective dose for each task and ech generation

Another factor to take into account is the added difficulties to reassign the collective dose to a new task code from a previous an inhomogeneous one. In addition, some plants do not have computer capability to provide automated the radiation work permit information, at least for part of the studied period, to elaborate the reassignment. We will proceed to comment in some detail each one of the selected tasks.

Steam Generator Primary Side

Steam generator (SG) primary side has been, and still is, the highest contributor to the refuelling outage collective dose in Spanish PWR, both for global and for the last two generation average1. The global average trend of this task presents a progressive slight decrease, except in 1990 in which an important rise took place, owing to the influence of the first and second generation. The second generation reactors represent far and away the major amount to the average collective dose of this task, while the first and third generation results fall rather lower than average. These very different values among generations lead to high standard deviation and variation coefficient.

The singular design of the first generation reactor, Westinghouse with a single coolant system loop and a SG Model 24 with tubes of Inconel 600 MA, the design and material of the second generation SG, Westinghouse Model D3 with tubes of Inconel-600 MA, and the improvements introduced in the third generation, one of which is KWU/Siemens with tubes of Incoloy 800 and the other an advanced Westinghouse Model F, are relevant aspects that explain greatly the mentioned difference and fluctuations of this task.

Table 2. Steam generator primary side results. Comparison among generations

	Average Person.mSv	%	Maximum Person.mSv	Minimum Person.mSv	Standard Deviation	Variation Coefficient
All generations	588	28	1676	30	406	0.69
First generation	295	14	621	100	180	0.60
Second generation	852	34	1676	441	318	0.37
Third generation	177	19	297	30	104	0.59

As we have mentioned above, the contribution of the first generation is significantly lower in this task than that of the second, for similar radiation levels in the SG channel head, maintaining the same relation between respective average collective dose values as the number of loops relation, 1:3. The average collective dose for the first generation would have been even lower if it had not registered an extremely high value in 1991, 621 Person.mSv, due principally to exceptional works (22 PIPs and drill of 8 explosives pluggs). José Cabrera has experienced relatively few problems in the SG tubes, principally thinning at the top of the tubesheet and stress corrosion cracking (SCC), with only 5% of the tubes plugged in 23 years, which have led to a lower scope of this task than in the next generation and consequently lower doses.

The second generation SGs have had substantial problems in their tubes, axially cracks located at support plates and the roll expansion and transition zones, and also some circumferential cracks, that have caused a large scope of inspections, plugging and sleeving works. So much so that utilities have decided to replace the steam generator of both plants in the near future (95-97). These problems joined to the corrective measures taken, as the shotpeening of tubes, has caused a good deal higher collective dose than the average. Nevertheless, the evolution of the collective dose in the second generation shows an inflection point in 1990. Awaiting deeper analysis, we can say in advance that two of the factors which have had a notable influence to break the trend and maintain the collective dose in moderate levels are: the change in the primary chemistry (pH from 6.9 to 7.4) and the systematic use of robots for plugging the tubes (SM-10 y ROSA-III).

The relative relevance of this task in the third generation is nearly exclusively due to Vandellós II. In effect, the SGs of Trillo have not experienced up to now any problem of cracking in their tubes, only one remains plugged due to a loose part. The collective dose registered in Vandellós II is in any case lower than that of the former generation, due principally to the new Westinghouse model of SG mentioned, which has presented only a extensive problem of fretting in the contact zone with the anti-vibration bars (AVB) and with the replacement of the bars for an optimized design in 1992, the need for plugging tubes and the dose associated will be reduced significantly.

Consequently, 1997, the year of the last SG replacement of the second generation PWR, will probably mark a historical date in which the works involving the SG primary side will stop being the critical task for Spanish PWR from the radiological point of view.

General Works

The average collective dose on general works shows a tendency to decrease from 1990, where it was registered a peack owing to a general rise presented in all generations. The higher contribution to this average comes from the second generation and the lower from the third, leading to a high variation coefficient for all generation and within the second and third one. The major contribution to this task is due to decontamination and cleaning works and shielding, and the results depend greatly of the radiological conditions and the scope of the job. These are the aspects where higher efforts should be done in order to reduce unnecesary dose: the source term and the systematic evaluation of the balance between dose saved and resulting dose.

Table 4. General works results. Comparison among generations

	Average Person.mSv	%	Maximum Person.mSv	Minimum Person.mSv	Standard Deviation	Variation Coefficient
All generations	274	13	755	44	188	0.68
First generation	241	11	348	60	63	0.26
Second generation	359	14	755	114	194	0.54
Third generation	103	11	237	44	56	0.55

The first generation average is located within the global range, but a clear increasing trend until 1991 was broken in 1992 where was registered a value under the average. As we have mentioned before, one factor to take into account in this analysis is the difficulties to reassign the collective dose to a new task code from a previous an inhomogeneous one. In this sense, we have found a particular case of the second generation reactors, while Ascó tends to amount all the auxiliary works within the general works task, Almaraz includes it in the system no listed task. So, the values of Almaraz are not very representative of this task. In the other case, Ascó shows a fairly good correlation trend with the evolution of the source term.

In relation with the third generation, the results recorded in Trillo, with an average value of 122 Person.mSv, are higher than those of Vandellós II, with 85 Person.mSv in average. In 1991 the highest value in Trillo, due to both an important increase in the scope of decontamination works, greatly related with the primary coolant pumps works, and a generalized increment in radiation levels, around a 50%, took place.

Refuelling

Stability is the tendency shown by the results of this task, with a progressive decrease from 1989 and remaining around 200 Person.mSv after that date. This value is nearly the same as the second generation average, since the starting up of the third one in 1988 balanced the contribution of J. Cabrera. In effect, the first generation collective doses are significantly higher than those of the rest, with cyclic behaviour in even years, while the third generation plants present much lower values.

These extremes differences are reflected in a moderate high variation coefficient for all generations in comparison with those of each generation. Refuelling includes reactor disassembly and assembly, fuel shuffle and cavity decontamination. A very short variability in its scope and the amounted experience in workers involved in these repetitive works can explain the stability and a small variation coefficient in each generation. Only difference in source terms, tools (i.e. remote reactor vessel head multistud tensioner and spinner out) or the ocurrence of any incident (i.e. stud grips) can cause some fluctuation among the collective values in this task.

The fact that refuelling means higher collective dose for J. Cabrera than for the rest is very significant and characteristic for this reactor and its peculiar design. Effectively, its average value is over the average, both in absolute and in relative terms, and it is the second major contributor to its collective dose, twice the values of the third. Though, last years the influence of a new tool, a triple tensioner, seems to notice in its results, but some gripping problems mask them.

The evolution of this task is very stable for the second generation, with a very low variation coefficient, in which a clear correlation can be established with the diminishing source term from 1991; although, remaining over twice the third generation values.

Table 5. Refuelling results. Comparison among all generations

	Average Person.mSv	%	Maximum Person.mSv	Minimum Person.mSv	Standard Deviation	Variation Coefficient
All generations	210	10	497	32	121	0.58
First generation	441	21	497	359	47	0.11
Second generation	205	8	276	154	37	0.18
Third generation	76	8	115	32	26	0.34

Third generation results show a significant different behaviour between both reactors that make it up. In effect, the average collective dose in Trillo is nearly half the one in Vandellós II, despite the important increase that took place in Trillo in 1991, up to 90 Person.mSv, due to 20 stud grips that had to be finally pulled out manually. This different conduct can be related with the use in Trillo of a remote reactor vessel head multistud tensioner from the beggining, and another similar for straction from 1991. This plant estimates in 50% the dose saved with the use of this tool.

As the impact of this kind of tools has been reflected clearly in the third generation, the accomplishment of an ALARA quantitative analysis to figure out its advantage for the second generation reactors and Vandellós II would be recommendable. Plenty of examples have been made for this particular item and the majority with a positive conclusion (ref. 3).

System not listed

As we have mentioned before, the presence of this task among the higher relative weight in collective dose (10%) is due more to the difficulties in supervising the collective dose associated to the rest of the task or reassigning them to a new task code than to the radiological relevance in any particular system not listed. In this sense, the results of this task can give an rough idea of the validity of the rest of data.

Tabla 6. System not listed. Comparison among generations

	Average Person.mSv	%	Maximum Person.mSv	Minimum Person.mSv	Standard Deviation	Variation Coefficient
All generations	205	10	618	1	202	0.98
First generation	507	24	618	424	70	0.14
Second generation	169	7	461	1	189	1.12
Tercera generación	76	8	162	27	52	0.68

These results show a clear difference among generations, the more modern the generation is the lower collective dose, concuring with more computer capability to provide automated overseeing and assignment of the collective dose according with a predetermine code. In the case of the first generation reactor, J. Cabrera, the lack of an electronic operational dosimetric system during the studied period has greatly risen its difficulties.

Anyway, a general tendency to deminish seems to indicate an improvement with the homogeneity introduced with the publication of the CSN guide 1.5, and can be even further reduced taking advantage of the lesson learned in this analysis.

Valve works

The average collective dose of this task remains stable, around 160 Person.mSv, during the studied period, with a light drop from 1991, despite the cyclic behaviour of the first generation reactor, which contributes in first range, owing principally to the steadiness of the second and third results. Consequently, this is the task with lower variation coefficient for all generations. This is a peculiar conduct taking into account the variety of the scope of this task, according to the number of valves involved and the extent of the work (inspection, maintainance, repair or modification), and the diversity of radiological conditions according to their location.

Table 7. Valve works. Comparison among generations

	Average Person.mSv	%	Maximum Person.mSv	Minimum Person.mSv	Standard Deviation	Variation Coefficient
All generations	157	8	322	21	81	0.51
First generation	239	11	322	139	74	0.31
Second generation	183	7	251	98	45	0.25
Third generation	50	5	74	21	18	0.37

The fact that the first generation reactor presents the highest average collective dose is very significant taking into account that the design is less complex, with a single coolant loop. Less ALARA regards, concerning biological shielding, accessibility and easiness for maintenance, among others, could explain these results.

The evolution of the valve works collective dose in the second generation reflects the same correlation with the source term as we have seen for other tasks before, but remaining in any case three times as high as that of the third values. More inspection requirements, less ALARA design characteristics and the effect of aging on source term can be related with this fact.

Insulation

The behaviour of the insulation average collective dose for all the generations is cyclic, showing a profile with picks in even years, owing essentially to the second generation contribution. In 1991, the third generation suffered an important increase, but it was counteracted by the drop of the results of the first one. The disparity among the respective values of each generation leads to a high variation coefficient for the array and for the second and third one.

Insulation removal and replacement greatly depends of the scope of other works, specially in-service inspection, complexity and design of the plant and the type of insulation, principally if it comprises quick connect tabs.

These considerations can make intelligible the fact that J. Cabrera, with a single loop, registers the lower collective doses, while the second generation presents the highest. On the other hand, problems related with coordination and the distribution of responsability have been detected as the cause of rework and an excessive extent of the task that explain, at least partially, some of the higher values recorded.

Tabla 8. Insulation removal and replacement results. Comparison among generations

	Average Person.mSv	%	Maximum Person.mSv	Minimum Person.mSv	Standard Deviation	Variation Coefficient
All generations	123	6	384	15	95	0.75
First generation	52	2	80	38	15	0.29
Second generation	176	7	384	76	92	0.52
Third generation	48	5	119	15	33	0.68

Reactor coolant pumps

Coolant pump works present an increasing trend until 1992, date in which a remarkable fall is registered. The main contribution comes from the third generation, especially in 1990 and 1991, and from the second one, while the first generation has not registered relevant collective dose during the studied period. Important fluctuations among both different plants and different outages have led to a very high variation coefficient for the array of PWR and for each generation.

These fluctuations reflect principally the great dependence of this task of the scope of the work, where inspection, maintenance and repair of the pump internals involves the main radiological risk. Such is the case of the profile shown by the second generation, where, despite the specific shielding and decontamination performanced, the pump internals inspections have been responsible of the important peaks registered in 1989, Ascó II and Almaraz II, and in 1990, Almaraz I.

Problems found in reactor coolant pumps of Trillo in 1989 have become in this plant the main contributor to the PWR average collective dose for this task and this task represents the first range for the global collective dose during the refuelling outages. In 1991, where a maximum value of 535 Person.mSv was registered, the internals of the three coolant pumps were removed, inspected, repaired or modified and, parallel, the source term of the plant increased significantly.

Table 9. Reactor coolant pump results. Comparison among generations

	Average Person.mSv	%	Maximum Person.mSv	Minimum Person.mSv	Standard Deviation	Variation Coefficient
All generations	104	5	535	2	107	1.03
First generation	31	2	104	2	39	1.27
Second generation	114	5	302	32	76	0.67
Third generation	129	14	535	27	162	1.26

Routine inspections

The profile of the collective dose evolution for this task presents a slight increase until 1990, diminishing in 1991, and rising again in 1992. The major contribution comes from the second generation, although the first one also contributed to the increment of 1992 in a relevant way, but remaining before that in lower values than the average and meaning this task less than selection level (5%) of the total collective dose for this generation.

The results of this task are very dependent of the scope of the testing program (that includes in-service inspection not listed in other tasks and snnuber, hanger and anchor bolt inspections) and the location of the components to inspect, which determines the radiological conditions associated to the work. The first factor is defined by safety codes (ASME codes and IE bulletins) and only a relief on high collective dose inspections by the regulatoy body or ASME can modify it. In relation with the second factor, the plant design has a strong influence but can be modified by shielding the local hot spots, testing in a low dose rate area and delating the high dose rate components inspections as much as possible.

Table 10. Routine inspections results. Comparison among generations

	Average Person.mSv	%	Maximum Person.mSv	Minimum Person.mSv	Standard Deviation	Variation Coefficient
All generations	97	5	259	20	58	0.60
First generation	51	2	85	35	18	0.35
Second generation	123	5	259	20	59	0.48
Third generation	64	7	102	22	30	0.68

Both, first and third generation show a cyclic behaviour with alternant peaks that counteract among them. Any way, relevant fluctuations have been registered for the PWR average leading to high variation coefficient for the global array and the generation coefficient of each one. The close relation with insulation makes their respective evolution profiles very analogous.

CONCLUSIONS

Although tentatively, on one hand because this is the first approach of this kind in Spain, and because a parallel analysis on dose reduction thecnics evolution, now under way, has to be finished, the following conclusions can be said:

The trend in average occupational collective dose per reactor in Spanish Pressurized Water Reactors (PWR) seems to be stabilising and lightly decreasing during the studied period of time, 1988-1992, within the same range of the average level in the OECD countries. In these global terms, each respective ALARA regards in design have been determinant in the results of each generation.

The evolution of the collective dose in the second generation shows an inflection point in 1990 for many tasks. Awaiting deeper analysis, we can say in advance that some of the factors which have had a notable influence to break the trend and maintain the collective dose in moderate levels are: the change in both plants of the primary chemistry (from pH 6.9 to 7.4); a 100% fuel inspection, a fuel no damage policy and a global plan for hot spot shielding in Ascó.

Steam generator (SG) primary side has been, and still is, the highest contributor to the refuelling outage collective dose in Spanish PWR, both for global and for the last two generation average. Nevertheless, 1997, the year of the last SG replacement of the second generation PWR, will probably mark a historical date in which the works involving the SG primary side will stop being the critical task for Spanish PWR from the radiological point of view.

As the impact of remote reactor vessel head multistud tensioner has been reflected clearly in the third generation, the accomplishment of an ALARA quantitative analysis to figure out its advantage for the second generation reactors and Vandellós II would be recommendable.

Several tasks, as valve works, insulation and general works, have shown to be very sensitive to problems related with a lack of coordination and an ambiguous distribution of responsability, causing rework and an excessive extent of the task that explain, at least partially, some of the higher values recorded. Efforts to avoid these problems would be among the priorities to reduce the collective dose of these tasks.

Finally, in spite of other analysis results, we can affirm that a progressive development of ALARA culture is having a positive impact in Spanish PWR and we have to develope even further.

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Author Biography

Patricio O'Donnell is an Official from the Special Nuclear Safety and Radiolocal Protection Group at the Nuclear Safety Council (CSN) of Spain and Head of the Occupational Radiation Protection of the Radiolocal Protection Division. Among his responsabilities are: ALARA concerns in the licensing and dismantling process of new fuel cycle facilities and Nuclear Power Plants (NPP) (and the renewal of their permission) in the reviewing of the major design modification, i.e. the Steam Generator Replacement in Ascó and Almaraz plants, ALARA inspections during refuelling outages, and the promotion of ALARA principle in NPP. Apart from his work in the CSN, he is Vice-chairman of the Information System of Occupational Exposure (ISOE), established by the NEA-OECD and teaches in the Energetic, Environmental and Technological Investigations Center (CIEMAT) in the Advanced Course on Radiological Protection. He has a B.Sc in Physics from the Complutense University of Madrid (Spain) and belongs to the Spanish Radiation Protection Society.

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SESSION 8B

BWR AND GAS COOLED PRESENTATIONS

Co-chairs:

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SCALE MODELS: A PROVEN COST-EFFECTIVE TOOL FOR OUTAGE PLANNING

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ABSTRACT

As generation costs for operating nuclear stations have risen, more nuclear utilities have initiated efforts to improve cost effectiveness. Nuclear plant owners are also being challenged with lower radiation exposure limits and new revised radiation protection related regulations (10 CFR 20), which places further stress on their budgets. As source term reduction activities continue to lower radiation fields, reducing the amount of time spent in radiation fields becomes one of the most cost-effective ways of reducing radiation exposure. An effective approach for minimizing time spent in radiation areas is to use a physical scale model for worker orientation planning and monitoring maintenance, modifications, and outage activities. To meet the challenge of continued reduction in the annual cumulative radiation exposures, new cost-effective tools are required. One field-tested and proven tool is the physical scale model.

INTRODUCTION

This article details how three drywell area models for Commonwealth Edison Company (CECo) were deployed to improve outage performance through improved planning and contractor employee plant orientation, resulting in a considerable reduction of both CECo and contractor radiation exposure.

Cost-Effective Tool

The model is a tool. And like a tool, it saves time; specifically, it reduces time that people spend in containment, saving both dose and money. By adding physical models to their operation and maintenance programs, nuclear stations can improve the quality of their maintenance work, lower personnel radiation exposures, reduce design interferences, lower modification costs, and improve the quality of backfit design.

The complexity of a particular model is determined by its intended use. If the model is to be used for equipment lay-down only, a simple block model may be sufficient. If it is to be used to support maintenance, modification, and outage planning, a detailed model including equipment, piping, raceways, ducts, supports, etc., may be required. The benefits of the physical model for design and modification work are well known.

Because a physical model is uniquely capable of presenting total visual information, it can provide at a glance what would otherwise require a thorough examination of many drawings. Models can show a plant system or area in its full three dimensions, easily allowing station personnel to jointly review and evaluate the area.

A detailed model permits rapid comparisons of alternate designs, facilitating the choice of a constructible, operable, and maintainable design. From a standpoint of keeping personnel radiation exposure as low as reasonably achievable (ALARA), using a model can minimize the time needed for design personnel to conduct field verification activities in high-radiation areas.

Before planned outages, station personnel can use a model to collectively view the area and discuss the scope of work. The model can be used to plan manpower allocation and to identify restricting or interfering work assignments, which can save valuable outage time, reduce personnel radiation exposure, and help with future outage planning. As a result, a model can easily pay for itself.

Maintenance-Outage Planning Tool (Part 1)

Our first drywell area application, a 1:16 physical scale model of an operating BWR Mark II for LaSalle Unit 2, was completed and successfully used by the CECo project team as an outage and planning tool. The area model, built with a ribbed containment and bioshield structure, also allows ongoing continuous model development to support design use for the life of the plant.

Station outage personnel used the scale model to collectively view the work areas and plan the scope of work away from the drywell. The model was also used to plan manpower allocation and identify restricting or interfering work assignments. The planning and review of work locations for maintenance activities were easily visualized and demonstrated on the model, creating a significant reduction of personnel radiation exposure, improved lead shielding placement, and improved prejob briefings before entering the drywell.

Economic Justification

In the future, the availability of the area model will greatly improve the long and short range planning capabilities while the drywell area is not accessible. The model was completed in nine weeks at a cost of approximately $80,000. The benefits indicated show that the cost of the model was easily recovered. For the economic justification, CECo used a factor of approximately $5,000 for one person-rem. Since then, the client has increased the value to approximately $10,000 per person-rem.

The following benefits were noted by the utility through a post-outage evaluation on the use of the model.

- An estimated 6 person-rem was saved by using the model to improve ALARA briefings by Inservice Inspection engineers (ISI), modification work locations, temporary lighting planning, prejob crew allocation and prearranging the flow path of equipment to and from work locations. **Estimated Savings: $30,000**

- From the model, improved engineering planning benefitted 9 person-rem in estimated savings by prearranging temporary lead shielding placements and demonstrating to craft personnel the location and method to hang the shielding. **Estimated Savings: $45,000**

- An estimated 5 person-rem were saved through improved utilization of scaffold and platform layout and identification of the locations for valves, snubbers, and welds. **Estimated Savings: $25,000**

- The orientation of new personnel, such as craft and/or station personnel to the general layout of the drywell, was accomplished through the use of the model, reducing the expected orientation exposure accumulation to almost zero, saving an estimated 5 person-rem. **Estimated Savings: $25,000**

  Outages are overwhelmingly the source of exposures in nuclear plants. According to NRC staff calculations, outage dose accounts for 31 times the nonoutage dose on an industry-wide average.

In our most recent drywell areas application for Dresden 2 and 3, two 3/4"=1'-0" physical scale models were completed and successfully used by the CECo project team as an outage planning and management tool. To explain in detail the purpose of the models, the daily planning activities with ISI engineers and craftsmen, project benefits and economic justification, I would like to introduce Mr. Roy Lee.

Dresden 3 - A Case Study (Part 2)

The scale model that you are viewing has already been used in one outage. It paid for itself by saving 57 person-rem through reduced man-hours and radiation exposure. During the outage, the model was set up in the craft supervisory area where maintenance foremen had their offices to make it accessible. It took about two weeks for people to realize the model's benefits and, after they did, there was so much activity around it that it was nicknamed the "ALARA coffee pot." There were as many as 25 to 30 people at a time around the model planning their daily activity.

As Bob said in Part 1 of this talk, the model is a tool. Well, what does this tool do? It saves time. Specifically, it reduces the time that people spend in containment and that saves dose and money.

Orientation Benefit

One of the major time savings comes from orienting people. One of the common problems during an outage is that new people who are not familiar with the containment layout have a difficult time finding their way to work locations. Well, if a picture is worth a thousand words, a model is worth a thousand pictures. For most of these people, it is easier to orient themselves when they can see the entire drywell in 3D than they do when they are given verbal instructions, marked-up drawings, even photographs. Of course, it helps that the model has colored see through floors, and can be viewed from several different angles. But the fact remains that almost everyone can relate to a scale model and remember what they saw.

The model has all of the major landmarks so it was easy to show a team exactly where to find a snubber, valve, ISI point, or any other work location. ISI points were shown on the model using small white "post-it" labels or "stickeys." It was also easy to show team members the correct "low dose" route to take to work. This meant they could get to work faster. Even more importantly, they did not need to waste time and dose walking around the containment, sometimes in the wrong direction, looking for their work. In summary, the model makes it easy for people to get to the right place quickly the first time. To make this even easier, a book located at the drywell entry had photos of the containment location next to photos of the model. People could see exactly where to go, how to get there, and what to look for when they arrived.

Planning Benefit

A second timesaver is simply keeping people out of the containment. It was easy to plan major equipment moves, staging areas, and work sequences from the model so the need for physical walkdowns was greatly reduced. Every person kept out of containment means a saving of time, money, and dose. In addition, it is easier to discuss something standing around a model in an office than it is standing under a valve with full plastics and a respirator.

Shielding Benefit

The model really helped when it came to installing temporary lead shielding. The shield installers were shown on the model exactly where to put the lead blankets. Since everything is to scale, the correct location and number of blankets could be put on the model. Red plastic blankets were cut to scale to show the exact blanket locations. Shield installers were then given a shielding package that had a photo of the model (with the shielding on it) and a plant photo marked up to show where the shielding was to be placed.

There was a real time saving on shift turnovers. For example, the insulators could indicate exactly where insulation was partially removed so that the next crew knew exactly where to continue removal. The ISI crews benefitted too because they could schedule their work without having to do a walkdown in the containment to check progress.

With all of this information shown on the model, people began to use it to coordinate the flow of equipment and personnel. The real value here is that since work in progress is there for everyone to see, the situation can be avoided where one crew is sitting around waiting for another crew to finish so they can start.

Since the model is always available, it was then used for postoutage reviews and prejob planning.

Economic Justification

To pay for the model during the last outage, any one of the following had to occur

- reducing time spent in the drywell by 1 percent.
- improving lead blanket handling efficiency by 30 percent.
- improving ISI efficiency by 0.5 percent.
- preventing just one overexposure.
- preventing disassembly of one wrong component.

The final result of the outage was an ISI exposure saving of 12 person-rem, a lead installation exposure saving of 25 person-rem and a total saving of 57 person-rem. The model cost was approximately $100,000 and resulted in a financial savings of approximately $540,000 for one outage (one person-rem = $9500).

Since the model paid for itself during the first outage, its use is essentially free from now on. But, is the model sturdy enough to last for more than one outage with all of the use it sees? The answer is a qualified yes. These are plastic creations with a lot of small parts, and some people do have heavy hands. There was some minor damage but it was easily repaired, and the model is ready for another outage. Furthermore, it can be turned into a living model by installing the mods as they occur -- after they have been planned out using the model, of course.

Dresden 2 - Continued Outage Development (Part 3)

Our third drywell area model was used last year as a planning tool for a major outage. The model for Dresden 2 was purchased by Engineering and Construction, buoyed by the success of the Dresden 3 outage application.

The person rem savings attributed to the model was 90 person rem with a savings of 4000 ISI hours. The largest saving, 45 person-rem, and benefits came from the Inservice Inspection Program (ISI). The area model was used daily to identify and describe the ISI team program and locate ID points. Implementation of the program through team meetings and briefings with the ISI team, craft, management, and prejob craft meetings, was a big part of the savings. The model cost was approximately $120,000 and resulted in a financial savings of approximately $900,000 for one outage (one person-rem = $10,000).

Outage Team Survey

Finally, I would like to give you some quotes from surveys that we made following the outages. These quotes are from two stations that used BWR drywell models.

> "It shows techs exactly where to take surveys."

"Instead of hunt and find in the drywell, now you hunt and find on the model and go directly to the area in the drywell."

"I think the model was as useful as any other tool used in the outage." (From drywell coordinator)

"Not as much time was spent discussing the job at the jobsite."

"It saved us from confusion."

"Showed where to put tents."

"It was faster and easier to show men where the work was located rather than having to explain."

"Used model for planning before access to containment was available."

"The model encouraged a lot of conversation about jobs that normally would not have taken place."

Most of the comments were positive but, like everything else, the model setup was not perfect.

"Need someone to keep the model clean and make minor repairs."

"Would like some electrical details shown."

"Put it on a higher table with a guard rail."

"Would like a step stool to see the top."

And one comment from an iron worker that I really like, "It's a good start."

SUMMARY

In the past, when working with the craft, we often felt that they viewed ALARA as being just the opposite of donuts: donuts taste good on the way down but, in the end, they are not very good for you. However, ALARA can taste pretty bad on the way down but be great for you in the end. One of the ALARA pills that is often difficult for the craft and their supervisors to swallow is planning, planning, and more planning. For the craft, the model is the sugar coating on the ALARA planning pill.

To meet the challenge of continued reduction in the annual cumulative radiation exposure for nuclear workers, new cost-effective tools are required for improving the quality of time spent in radiation fields and assisting utilities in operating safe, reliable, and efficient nuclear stations that are competitive on individual and industry basis. Optimization of ALARA practices does not rely on any single ALARA tool, but rather on a combination of tools. One effective tool is the physical scale model.

Authors' Biographies

Roy Lee is an ALARA Engineer for the Corporate Health Physics Support Group of Commonwealth Edison Company. He has 22 years of combined experience in radiation protection and plan operations for both boiling water and pressurized water design nuclear power plants.

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PAPER 8B-1
DISCUSSION

Unknown: Where did you locate the model?

Lee: Actually, it was moved around quite a bit. First, it was in the construction management office. Right now it is in the ALARA review trailer where the contractors go in. We had one similar model at the drywell.

Cybul: Are any of the components removable? If they are, how did you pick out which ones?

Lee: The components in this particular model are not removable, except for the reactor vessel. It is not intended to be what you may call a working model. Other models have been developed where you may do so.

DOSE REDUCTION AND COST-BENEFIT ANALYSIS AT JAPAN'S TOKAI NO. 2 PLANT

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ABSTRACT

In the Tokai No. 2 power plant of the Japan Atomic Power Company, about 80% of the annual dose equivalent is received during periodic maintenance outages. A project group for dose reduction was organized at the company's headquarters in 1986; in 1988, they proposed a five-year program to reduce by half the collective dose of 4 person-Sv per normal outage work. To achieve the target dose value, some dose-reduction measures were undertaken, namely, permanent radiation shielding, decontamination, automatic operating machines, and ALARA organization. As the result, the collective dose from normal outage work was 1.6 person-Sv in 1992, which was less than the initial target value.

INTRODUCTION

In 1957, the Japan Atomic Power Company (JAPC) was established by the electric power companies of Japan. JAPC now operates four reactors of three different types at two sites: one GCR and one BWR at the Tokai site, and one BWR and one PWR at the Tsuruga site. Tokai No. 2 plant, a BWR Mark-II, type-2 with 1100MWe of electric output, started commercial operation in November 1978. As a consequence of its safe and stable operation, the station holds Japan's record for power generation for a single unit. In August 1992, a world record for BWRs of 100 billion KWH was achieved. For several years after start of commercial operation, the annual collective dose equivalent was about 4 person-Sv, but later, by 1985, this value had increased to about 7 person-Sv. About 80% of this annual dose equivalent was due to the periodic maintenance work. A project organization on dose reduction was set up to halve the collective dose for regular outage work.

ORGANIZATION FOR DOSE REDUCTION

ALARA Coordination Committee

The Tokai No.2 plant, which was planned and constructed on the basis of experience from the Tsuruga No.1 plant (BWR Mark-I, Type-2, 357MWe), started commercial operation in March 1970. For several years, the dose rate in the working environments was maintained at a low level; but the surface dose rate of major lines increased year by year. As a consequence, by 1985, the annual collective dose equivalent had reached to 7 person-Sv. Figures 1 and 2, respectively, show the annual collective doses at Tokai No. 2 and the trend of surface dose rate of the primary liquid recovery (PLR) lines.

At the seventh outage (1985), the CUW piping had to be replaced as a countermeasure to SCC; the collective dose from this outage was anticipated to be more than 10 person-Sv. Therefore, a special organization for the dose reduction was set up on site, and an ALARA program was instituted before the outage. This activity is referred to as the INPO program.[1]

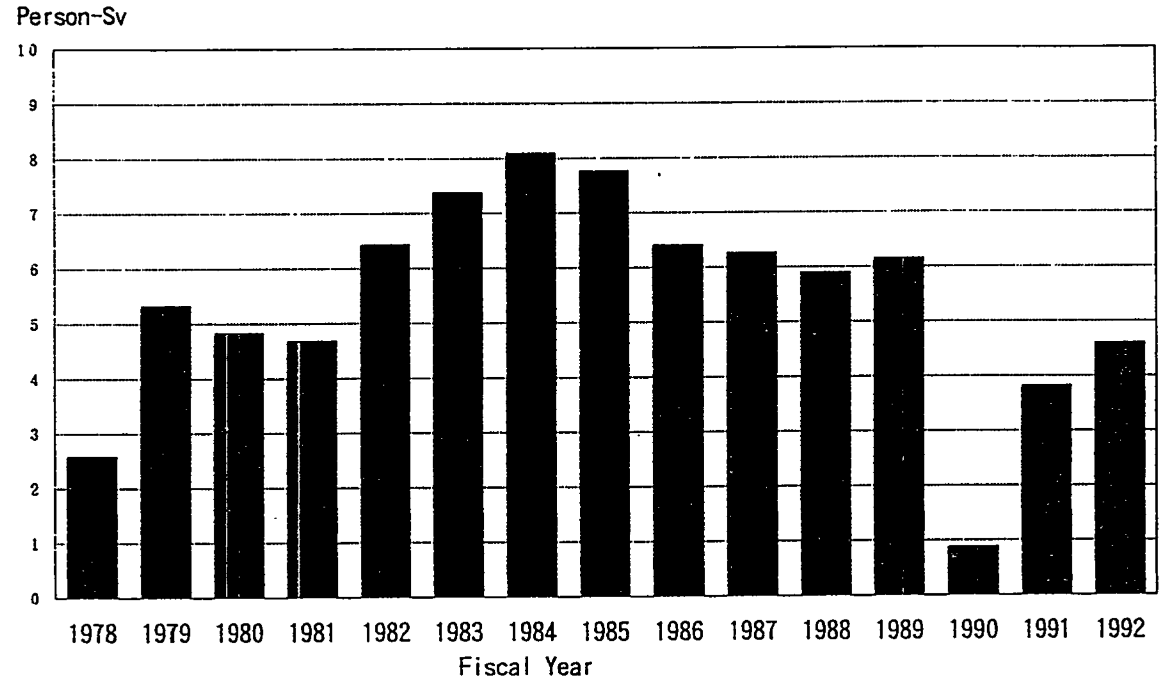


Fig. 1 Collective Occupational Dose

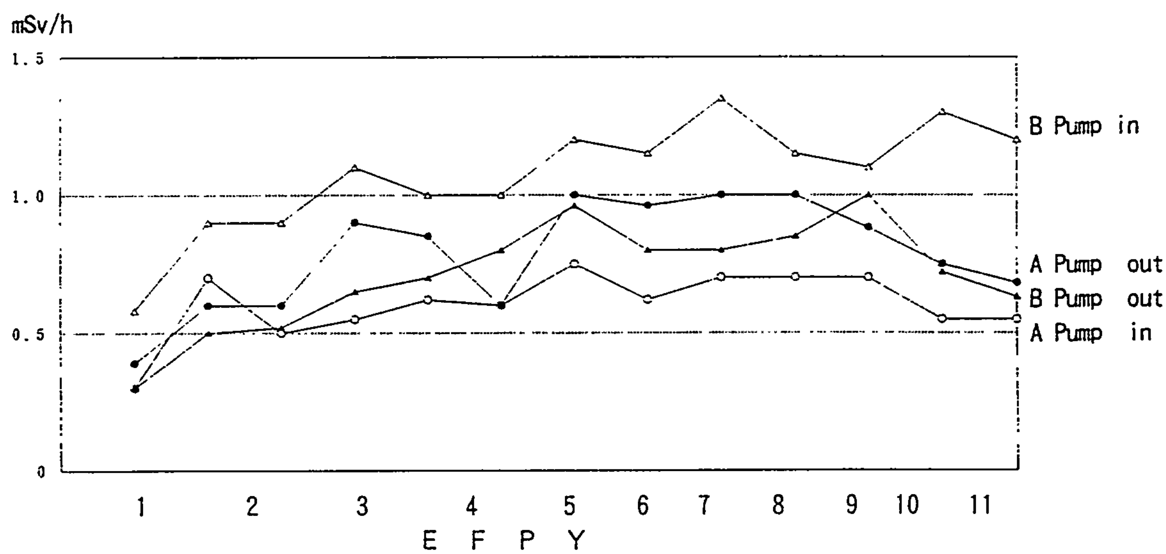


Fig. 2 Surface Dose rate of PLR Line

The organization, called the ALARA coordination committee, meets on a regular basis to review the status of the ALARA program. The committee is composed of about 20 members drawn from the major functional departments on site, deputy managers of each section of JAPC, and radiation control managers of the main contracting companies. The deputy superintendent of JAPC is the chairman, and both the deputy superintendent of outage contractors and the manager of the radiation control section of JAPC are the sub-chairmen.

The activity of this committee starts six months before the beginning of a periodic maintenance outage. First, the work items, work contents, and work places are recognized and initial estimate of exposure is made for each job step. Subsequently, work and job steps that involve high doses are identified and dose reduction measures are sought for them. Possible countermeasures that could be carried out in an outage are considered. Figure 3 shows the flow chart of ALARA program in JAPC. This committee meets once a week to check dose-reduction measures and give advice accordingly if the doses are higher by 10% compared to the estimates. Further, the group discusses selection of lines for flushing or additional shielding, the transfer of high-radiation equipment, and changing of the working area. The weekly report of the committee is sent to the outage schedule management committee that compares the results and the plans presented for ALARA. After the outage period, the ALARA activity, radiation data, survey data, and related photographs are compiled and a report with an evaluation of the yearly trend is written and edited. The results of this report is reflected to the next outage works. Figure 4 shows the results of outage dose. This ALARA coordination committee has been continuously managing outages from the seventh one and still is working effectively in the thirteenth outage this year.

The Organization of Dose Reduction for Whole Company

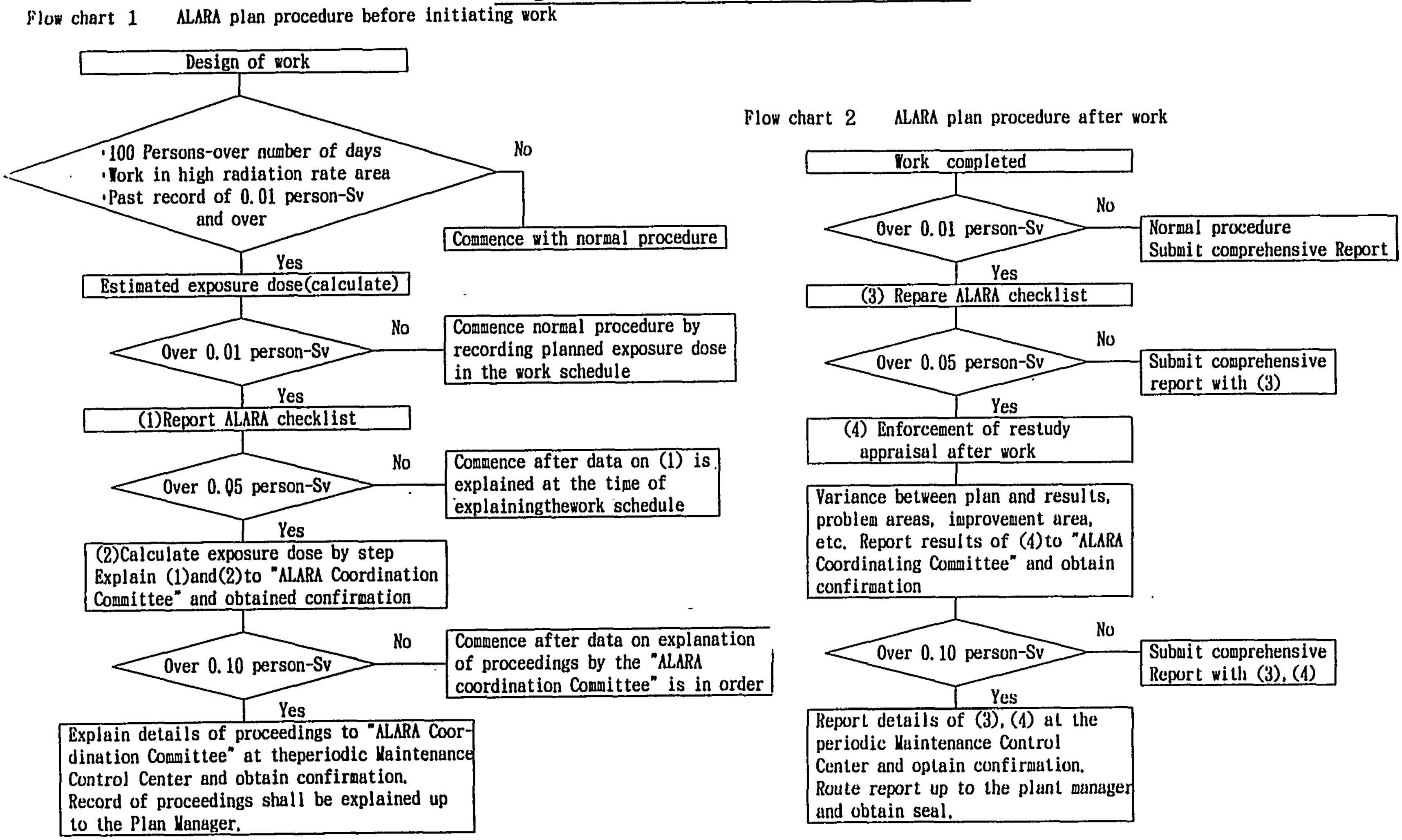
In the Japan Atomic Power Company, dose reduction activities were carried out individually in each section. To unify these programs, in June 1986 a dose reduction promoting group (DRPG) was organized at headquarters, and a dose reduction countermeasures committee (DRCC) was organized at the site. The DRPG is composed of the following members: executive general manager (managing director), who is the chairman of this group, the general managers at headquarters, and the superintendent of the sit. It meets about twice a year. The main objectives of this group are include the following: 1) planning a target for dose reduction; 2) assessing the needs of long and semi-long term planning and evaluating the results of dose reduction, 3) examining techniques for dose reduction, and 4) considering ALARA information for domestic and foreign nuclear plants.

The DRCC is composed of the deputy superintendent of the site, who is the chairman, and the managers of each sections and meets about twice a year. The main objectives of this group include the following: 1) planning methods to achieve the target of dose reduction, 2) examining the difference between the initial estimated dose and the actual dose for the report to the of ALARA coordination committee, 3) analyzing factors that might disturb dose reduction, and 4) reporting to the DRPG.

PROPOSING THE FIRST 5-YEAR PLAN

In 1988, the long and semi-long term planning for dose reduction at Tokai No. 2 was discussed by both the DRPG and the DRCC. Recognizing recent dose conditions, they analyzed the collective dose and effective dose rate of each work and each job-step, and the dose ratio of different sources. Consequently, items for dose reduction were selected in each field of work and each work place. In planning the dose-reduction targets, only the dose targets for regular outage work were discussed, and it was proposed to reduce the dose to 2.5 person-Sv per outage after five years from the average 4 person-Sv per outage of the sixth or seventh outage. Adoption of the techniques for dose reduction was finally decided upon, according to the results of a cost-benefit analysis.

Fig. 3 ALARA Plan Procedures



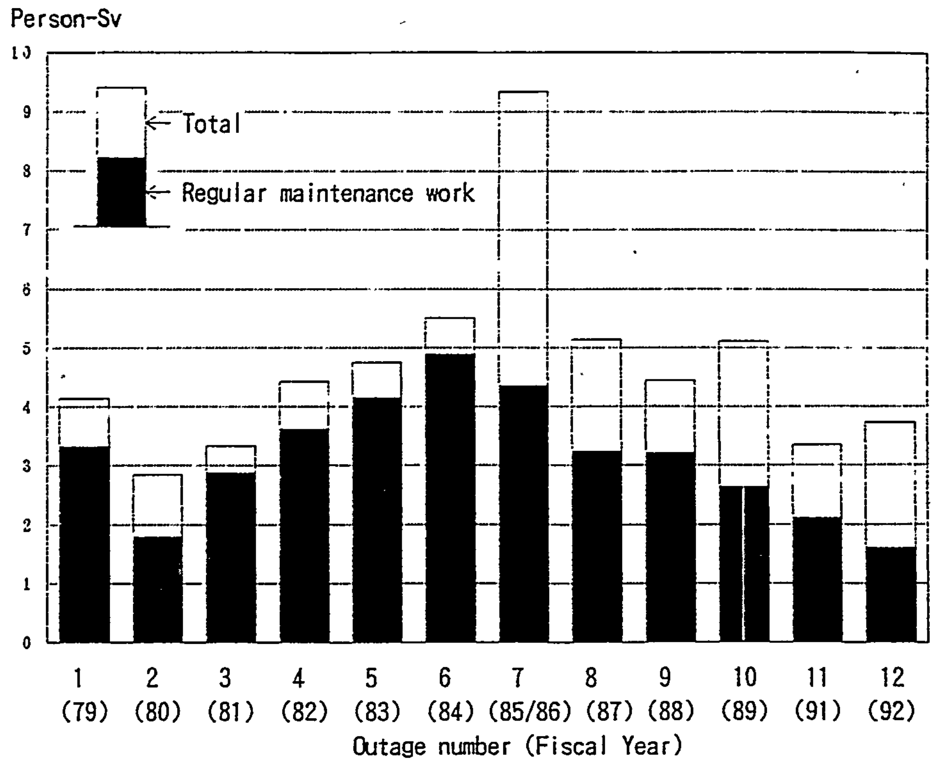


Fig. 4 Outage Dose

Table 1. **Results of Dose-Reduction Measures at Tokai No. 2**

Years (outage number)	Collective Dose (Person-Sv)
1988 (8th)	4.0

Reduction Measures	Dose Reduction Effect	Cost Benefit	
	Person-Sv/outage	($/person-mSv)	($/person-rem)
1. Permanent radiation shield	1.23	400	4000
• Piping in PCV (PLR, RHR, CUW sys)	1.00	---	---
• Piping out of PCV	0.23	---	---
2. Dose reduction by target dose management	1.00	400	4000
3. Automation and remote control of the work	0.31	4200	42000
• Automatic operation by advanced CRD exchanger	0.19	---	---
• Remote operation by reactor well decontamination equipment	0.06	---	---
• Others	0.06	---	---
4. Decontamination of reactor wall and piping	0.13	800	8000
5. Others	0.03	---	---
Total	2.60	---	---

1992 (12th)	1.6

Table 2. Exposure of Periodic Regular Maintenance Work
(The 8th Maintenance Outage at Tokai No. 2)

Work Area		Ratio (%)
PCV	PLR sys	24.3
	RHR sys	9.8
	CUW sys	6.6
	Others	19.4
	Subtotal	57.1
R/B (Out of PCV)	6FL (operating floor)	
	RHR sys	3.6
	CUW sys	5.1
	Others	11.6
	Subtotal	39.2
RW/B, T/B		3.7
Total		100

CARRYING OUT THE 5-YEAR PLAN

Some dose-reduction measures were undertaken, starting at the ninth outage (1988). In these 5 years since 3.3×10^9 yen (\$$3.3 \times 10^7$) has been invested for permanent shielding, decontamination, automatic operating machines, and so on. In consequence, the dose from periodic maintenance work in the twelfth outage (1992) was reduced to 1.6 person-Sv per outage, so achieving the initial target of reducing exposure from 4.0 person-Sv per outage to 2.5 person-Sv per outage. Furthermore, in the decontamination of reactor in the twelfth outage, 300 Ci CRUD was removed. This radioactivity was not directly removed from exposed sources such as the PLR piping, but from the steam separator and reactor annulus. The effect of this decontamination is expected to reduce gradually the dose rate of the PLR piping, although the effect of this countermeasure has not been evaluated yet. The following are the main countermeasures carried out for dose reduction:

1. permanent radiation shielding for the piping in the PCV,
2. decontamination of the reactor wall and shroud,
3. application of automatic operation and remote control,
4. renewal of construction materials,
5. enhancement of worker's consciousness for ALARA.

Tables 1 and 2 show the results of these measures. The results of cost-benefit analyses also are shown in same tables, in which the cost per person-Sv was calculated, considering the cost for each countermeasure and the accumulated dose in the effective working period for each task. The addition of permanent shielding and the enhancement of ALARA consciousness were the most effective of all the measures and their costs were low; their cost to effect amounted to 4×10^4 yen/person-mSv (\$$4 \times 10^3$/person-rem), the lowest of all. The cost to effect other countermeasures were between 4×10^4 and 4.3×10^5 yen/person-mSv (\$$4 \times 10^3$-\$$4.3 \times 10^4$ /person-rem).

SETTING UP STANDARDS FOR COST-BENEFIT ANALYSIS

The process of promoting of dose reduction is as follows;

1. recognition of dose-reduction needs
2. setting of target values
3. selection of items for dose reduction
4. evaluation of cost-benefit analysis
5. recognition of countermeasures for dose reduction
6. decision on countermeasures
7. development of the plan in each year
8. design and operation of countermeasures
9. evaluation of reduced dose

In this process, the most difficult matters to optimize are setting target values and evaluating the cost-benefit analyses. In setting a target value, other factors are considered, such as comparisons to other plants, the economic scope for dose reduction, the balance of dose limits of each worker, and management of persons working at other plants. An objectively useful cost-benefit (dose-reduction cost) index is necessary to promote an active ALARA program. But it is difficult to define this index because of 1) changes of social concepts on radiation exposure, 2) the variety of evaluation methods, 3) and changes in the economy of nuclear power plants.

At Tokai No. 2 plant, we are directing the contractors to plan individual outage doses of less than 1 mSv/day, unless a special plan is proposed. Following this direction, contractors plan individual doses of 0.3-0.5 mSv/day and schedule an effective working time of several hours in a high radiation area. If the dose rate in this working area can be decreased by half, then a person can work for twice as long at the same dose. Considering the relationship between labor cost per increased working time and reduced dose, the price per person-mSv becomes 1×10^5 - 3×10^5 yen (\$1 x 10^3 - 3×10^3). A report[2] of the BNL ALARA Center gives the same value. Considering of all of these factors, we have established an index of cost-benefit of 3×10^5 yen/person-mSv (\$3 x 10^4/person-rem) in the Tokai No. 2 plant.

CONCLUSION

In the Tokai No. 2 plant, the project groups were organized to consider dose reduction and some countermeasures were carried out. Setting up an index of cost-benefit of 3×10^5 yen/person-mSv (\$3 x 10^4/person-rem), several measures were adopted, such as permanent shielding, decontamination, automatic or remote operation, and renewal of equipment. As a consequence, the dose from regular maintenance work decreased 4.0 person-Sv to 1.6 person-Sv. The results of cost-benefit analysis of these measures were estimated roughly as 4×10^4 - 4.3×10^5 yen/person-mSv (\$4 x 10^3 - 4.3×10^4/person-rem) and were evaluated to be in the reasonable range. In 1993, a second five-year program was proposed to reduce to 1 person-Sv the dose from regular maintenance work. This program was started in 1994.

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Author Biography

Kazufumi Taniguchi is a deputy manager at the radiation control office of the Japan Atomic Power Company. Presently, he is primarily engaged in managing and developing the computer system for radiation control (mainly occupational dose management). From 1981-84, he worked at the radiation control section of Tsuruga Power Station, where his work focussed on environmental control. Before coming to the radiation control office of JAPC, he was temporarily transferred to the Japan Nuclear Fuel Service Company (JNFS), which is now the Japan Nuclear Fuel Company (JNF). For the three years that he was there, he was engaged in developing an in-line monitor for radioisotopes.

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PAPER 8B-2
DISCUSSION

Unknown: I am interested in the decontamination that you did inside the reactor on the walls and the steam separator. You cleaned the reactor vessel walls and the separator to remove the 300 Ci. What recontamination rate have you seen or do you expect? Do you expect it to build back up to that level or have you done something to keep it from reoccurring and having to perform another decontamination in future?

Taniguchi: The recontamination rate for this equipment and the effects of the removal of 300 Ci are now being evaluated, including the implementation of the continual decontamination of inner reactor.

Cody: You mentioned that the reactor well decontaminator saved three days of outage time. Could you explain how that was possible? Usually refuel floor are on the critical path, which makes that very attractive.

Taniguchi: For manual decontamination in former times, as you know, the scaffolding had to be set up. Through the introduction of the automatic decontaminator of reactor well, the construction of footing is now not necessary, and the removal of reactor water and the decontamination of reactor well are possible to be done at the same time. It took about 8 days to decontaminate before using this automatic decontaminate.

Cody: The name of the manufacturer of the reactor well decontaminator was ATOX, Inc. Is that a Japanese company?

Taniguchi: Yes, ATOX is a Japanese company. The telephone number is 0471-45-8801. The fax number is 0471-45-3649.

RADIOLOGICAL CONTROLS INTEGRATED INTO DESIGN

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ABSTRACT

Radiological controls are required by law in the design of commercial nuclear power reactor facilities. These controls can be relatively minor or significant, relative to cost. To ensure that radiological controls are designed into a project, the health physicist (radiological engineer) must be involved from the beginning. This is especially true regarding keeping costs down.

For every radiological engineer at a nuclear power plant there must be fifty engineers of other disciplines. The radiological engineer cannot be an expert on every discipline of engineering. However, he must be knowledgeable to the degree of how a design will impact the facility from a radiological perspective. This paper will address how to effectively perform radiological analyses with the goal of radiological controls integrated into the design package.

INTRODUCTION

Every health physicist, radiological engineer, ALARA coordinator, or whatever title one may hold, has a primary objective to maintain exposure to ionizing radiation As Low As Reasonably Achievable. Many hours have been focused on achieving reduction of radiation exposure in the commercial nuclear power industry. Why is this? Regulatory Guide 8.10[1] says it well:

> "Even though current occupational exposure limits provide a very low risk to injury, it is prudent to avoid unnecessary exposure to radiation. The objective is thus to reduce occupational exposures as far below the specified limits as is reasonably achievable by means of good radiation protection planning and practice..."

NUREG-1272[2] says:

> "Because the economics of operating a plant creates a strong impetus to lower exposures and achieve ALARA (As Low As Reasonably Achievable) objectives.. the vast majority of nuclear power plant personnel have annual exposures far below NRC regulatory limits ..."

NUREG-1272[2] further brought out that the average radiation worker received 1.9 rem twenty years ago compared to 0.4 rem in 1991.[3] Between 1983 and 1992, the average collective dose per U.S. reactor dropped by 65%. The report concludes that the reduction is believed to be primarily the result of the licensees' extensive dose-reduction efforts.

Many good practices that reduce dose have been implemented and communicated throughout the industry. Data collected by INPO (the Institute of Nuclear Power Operations) clearly indicates a strong commitment to achieving lower doses. One extremely important area that can be tapped to further reduce dose is the design change process. Granted, our plants have been built and all have some inherent designs that have proven to have not been in the best interest of ALARA. However, all power plants are getting older, new technology is constantly being developed, and changes are being made. The time for improvements is now. The health physicist must take advantage of the opportunities to include ALARA principles. However, one must not overlook the R, which stands for "reasonably," in the acronym ALARA. The purpose of this presentation is not to define reasonably, but to clearly present a methodology of obtaining dose reduction at a reasonable cost from a design change perspective.

CONCEPTUAL DESIGN INPUT

The design engineer must meet with the radiological engineer during the conceptual phase of the project. The initial communication should include basic information to enable the radiological engineer to determine the frequency of interface. The following three designations are recommended:

- **Category:** Insignificant/Low Risk

  Example: An emergency service water flow indicator is being replaced with a different more reliable type. Work is to be performed within the radiologically restricted area; however, dose-rates are only <0.5 mrem/hr (0.005 mSv/hr).

  Frequency: The need for radiological interface subsequent to the conceptual review is infrequent. Certainly, the radiological engineering group should review the final package as part of the approval process; however, there is no need for a periodic interface during the interim.

  Reason: The radiation levels in the area are very low, the proposed activity has no potential to cause a change in radiological conditions, and there is no change to the operability of the system from a radiological perspective.

- **Category:** Periodic/Moderate Risk

  Example: A design change is determined to be necessary regarding the installation of several temperature detectors on the residual heat removal system. Radiation levels are 100 - 300 mrem/hr (1 - 3 mSv/hr) and the work is expected to require 25 person-hours in the work area.

  Frequency: The need for interface for a job such as this would be periodic. The radiological engineer would determine exactly the frequency period. It may be weekly, monthly, or whatever he should feel appropriate.

  Reason: There is typically a degree of latitude when installing temperature detectors. A distance of a few feet may be quite substantial regarding dose rates. The radiological engineer could provide guidance. Additionally, conduit has to be installed for the wiring. Selection of where the conduit traverses is also important. Another consideration should be to compare the dose required for permanent installation and subsequent upkeep compared to temporary installation.

- **Category:** High Risk/Impact

  Example: The reactor water clean-up line in the drywell has sustained significant erosion/corrosion wear and is in need of replacement. Radiation levels in the work areas range from 20 - 2500 mrem/hr (0.2 - 25 mSv/hr).

Frequency: Radiological engineering would want to work every step of the way with the design engineer on a project such as this. Design changes in this category have a high risk of dose from the implementation and/or as a result of the implementation.

Reason: Many facets of the job have the potential to have a significant radiological impact. This includes not only the methodology of removing the old pipe and installing the new, but also includes material selection, and processes such as electropolishing and passivation. Additionally, other support activities, such as chemical decontamination prior to removal of the old line, or determination of permanent shielding as part of the design change, need to be considered.

Why break down the frequency of radiological engineering interface into three categories? Early, clear definition of radiological engineering involvement will help to ensure the modification is designed correctly from an ALARA perspective the first time. The identified frequency will ensure the proper emphasis is directed to design changes as appropriate, resulting in more bang for the buck. Thus, resources can be spent more efficiently.

INTERFACE OBJECTIVES

What are the objectives of the radiological engineer when interfacing with the design engineer? Simply, the radiological engineer would want to clearly communicate with the design engineer the radiological risk, impact, and benefit from the proposed change, along with potential dose reduction opportunities. By doing so, proper input can be provided at the appropriate time.

The radiological engineer should include the appropriate dose reduction methods relative to the cost/benefit. For example, a new type of gauge is being installed in the reactor water cleanup (RWCU) system. There are two options presented for review. 1) the first gauge cost $10,000, has a service life of 10 years, and requires calibration once every 12 months, and 2) the other gauge cost $14,000, has a service life of 12 years, requires calibration once every 18 months. The dose rate in the area is 50 mrem/hr (0.5 mSv/hr), calibration requires 2 person-hours in the area, entrance to the area requires a dress-out for contamination control.

Additional facets with replacing the gauge, regardless of which model, are; plant operations takes reading from this gauge shiftly (2 minutes), surveillances are performed quarterly (1 person-hour in the area), and two feet of tubing will be replaced.

Considering the given information, one can perform a comparison to determine which gauge is the best from a dollar versus dose perspective. However, the radiological engineer should not stop there. Since the RWCU system is a highly contaminated system that has the potential to cause a CRUD build-up within the tubing which could result in an increase in area radiation levels, one must take into consideration the selection of materials.

Consideration should be given to whether the gauge can be relocated. Perhaps by moving the gauge only a few feet radiation levels may be significantly lower. Possibly, by moving the gauge only a few feet dress-out for contamination control may be avoided.

Granted the additional cost of relocating will have to be weighed against the benefit. But the point is, DO NOT STOP AT THE SURFACE! Do some digging, be creative.

BENEFIT FROM INTERFACING

What is the benefit of integrating radiological controls in the design change package, especially from the conceptual stage?

Less cost, less dose.

How? If a design engineer spends time to formulate an idea and does not involve the radiological engineer, ALARA principles may not be included. When the radiological engineer does review the package, he may feel it appropriate to request (require) the design engineer to "go back to the drawing board." Thus a portion of the time (money) spent on the project would be lost.

Furthermore, additional ALARA benefit can be realized by the opportunity afforded during some design changes. For example, let us use the example of the gauge replacement discussed earlier. Relocating the gauge just for ALARA reasons in this case would not be an option. However, since the gauge is being replaced along with some tubing, the extra work might be acceptable. Along with the lower dose rates at the proposed location, time and cost savings associated with not having to dress-out should be factored in.

CONCLUSION

In conclusion, integration of radiological controls into design is an opportunity to further reduce radiation exposures now. ALARA actions can be implemented as part of a design change that would not have been an option otherwise. Collective dose is trending downward; use this viable resource to reduceccupational exposures as far below the specified limits as is reasonably achievable.

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8B-4

PC BASED TEMPORARY SHIELDING ADMINISTRATIVE PROCEDURE (TSAP)

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ABSTRACT

A completely new Administrative Procedure for temporary shielding was developed for use at Commonwealth Edison's six nuclear stations. This procedure promotes the use of shielding, and addresses industry requirements for the use and control of temporary shielding. The importance of an effective procedure has increased since more temporary shielding is being used as ALARA goals become more ambitious.

To help implement the administrative procedure, a personal computer software program was written to incorporate the procedural requirements. This software incorporates the useability of a Windows graphical user interface with extensive help and database features. This combination of a comprehensive administrative procedure and user friendly software promotes the effective use and management of temporary shielding while ensuring that industry requirements are met.

INTRODUCTION

Each of Commonwealth Edison Company's (CECo) six nuclear power stations (a total of six boiling water reactor and six pressurized water reactor units) had in place site specific administrative procedures for temporary shielding. A project was implemented to consolidate the best features of each of these procedures and develop an enhanced method for dealing with temporary shielding. A PC-based Temporary Shielding Administrative Procedure (TSAP) was developed for use at all CECo nuclear stations.

An objective of developing this procedure was to implement a consistent approach to shielding use and control at all CECo stations. An additional objective was to ensure compliance with industry requirements. The TSAP and software incorporate numerous enhanced features that benefit the promotion of temporary shielding use and implementation of administrative controls.

This procedure incorporates all applicable Industry requirements for controlling the use of temporary shielding. Requirements and guidelines issued by the U.S. Nuclear Regulatory Commission (NRC) and the Institute of Nuclear Power Operations (INPO) have been included. included. Use of the procedure was enhanced through the development of PC-based software. This software serves as a front end to a database, thereby enabling automatic updates and storage of shielding data, along with automated database queries and look-ups. The software includes an interview feature that asks the user applicable questions, and, based on the responses, fills out the appropriate forms. An option to directly fill out the forms is also available. Look-up libraries, hypertext, help features, and auto fills speed completion of the procedural requirements. Mandatory and non-mandatory responses ensure that the requirements are followed.

To assist with the development of the TSAP, existing procedures from numerous stations were obtained. The best features from these procedures were used along with input from personnel experienced in the use of temporary shielding. Applicable industry requirements were also reviewed and incorporated. An oversight committee consisting of Radiation Protection personnel from each of the CECo stations was set up. This committee periodically met to review and provide input to the development of the TSAP and the TSAP software. This oversight committee was a valuable source of input. Additionally the participation of the committee members, who will be the users of the procedure and software, was an effective means of ensuring that the resulting procedure and software would be relevant to each of the stations needs and be readily accepted by the intended users.

INDUSTRY REQUIREMENTS

There are several industry requirements that govern the use of temporary shielding. These requirements address shielding control, tracking, and evaluations necessary to assess the effects that shielding placement has on the design basis response of affected systems, structures and components. NRC IE Information Notice No. 83-64, *Lead Shielding Attached to Safety-Related Systems Without 10 CFR 50.59 Evaluations*, requires the evaluation of shielding effects on piping and the tracking of shielding installations. This Information Notice requires that the effects resulting from the temporary placement of shielding be evaluated in accordance with the requirements of the Code of Federal Regulations, Title 10, Paragraph 50.59. NRC I.E. Circular No. 80-18, *10 CFR 50.59 Safety Evaluations for Changes to Radioactive Waste Treatment Systems*, is referrenced by the Information Notice as containing guidelines for completing the necessary evaluations.

To address these requirements, INPO issued Good Practice TS-411, INPO 86-006, *Temporary Lead Shielding*, February 1986. This document provides a recommended program for the control of temporary lead shielding, including shielding evaluations, installation and tracking. These NRC requirements and INPO guidelines have been addressed by the TSAP.

PROCEDURAL REQUIREMENTS

The purpose of the TSAP is to further the use of temporary shielding in accordance with ALARA goals and to put in place administrative controls that will ensure compliance with industry requirements. Procedural steps are included to verify that the installation of temporary shielding does not adversely affect the plant's design basis. Additional steps are included to track and document the use, installation and storage of temporary shielding.

The procedure was written to include a number of forms. Each form addresses a specific aspect of the shielding requirements - both mandatory and optional forms are included. Completion of the mandatory forms is required to meet industry and station specific requirements. Completion of the optional forms will improve the documentation of the shielding installations. The TSAP was developed so that when the applicable forms have been completed, all the requirements of the procedure have been met. A listing of the forms along with a description of their objectives are provided below.

Form A - Temporary Shielding Request (TSR). This form is the first step in the shielding process. To promote the use of shielding, it can be completed by anybody to request the installation of shielding.

Form B - Temporary Shielding Justification. This form is completed by Radiation Protection personal to demonstrate that the shielding installation will result in a net dose savings.

Form C - Evaluation of Temporary Shielding. Evaluations completed to assess the shielding installation effects on plant systems, structures and components (SSCs) are documented here.

Form D - Field Installation of Temporary Shielding. This form provides shielding installers with installation instructions, precautions and a sketch of the shielding installation.

Form E - TSR Closeout. This form documents that the shielding has been removed and the affected SSCs have been restored to their required configurations.

Optional Form 1 - Dose Benefit Analysis. Calculations completed to determine dose savings specific to a given shielding installation are documented in this form.

Optional Form 2 - Temporary Shielding Sketch. This form documents the sketch(es) used to request, evaluate and install the shielding.

Optional Form 3 - Temporary Shielding Inspection Requirements. Requirements for any inspections that may be needed to verify that the shielding has been installed and remains installed according to specifications are documented here.

Optional Form 4 - Temporary Shielding Tracking Log. This log tracks the location of the temporary shielding along with the required installation and removal dates.

Optional Form 5 - Dose Information. This form provides for documenting the actual received dose for a particular shielding installation. This information can be used to verify the accuracy of pre-shielding calculations and to increase the accuracy of the documented dose savings resulting from shielding use.

Shielding Summary Form. This form provides a one page summary of all the necessary shielding documentation. The form is automatically completed (by the software) based on user input to the procedural forms.

TSAP SOFTWARE

Objectives

As is demonstrated by the number of forms described above, completing the applicable requirements can involve a significant effort. Therefore the TSAP was computerized to minimize the paperwork, expedite completion of the procedural steps, and ensure compliance with the applicable requirements. The TSAP software was designed to meet the following objectives:

- Reduce the time required to learn the procedure.
- Reduce the time and effort needed to complete the procedural requirements.
- Strengthen compliance with the procedural requirements.
- Enhance compliance with Industry requirements.
- Improve control and documentation of temporary shielding usage.
- Automate data storage and retrieval through use a database.
- Standardize the lead shielding administrative procedure used at all stations.
- Increase acceptance of the TSAP at all stations, through ease of use and fast turnaround of shielding requests.
- Improve accuracy and legibility of the documentation.

Software Features

The TSAP software provides two methods of completing the procedural forms; an interactive interview and a form filler. The features of the software are depicted in Figure 1. The user is able to choose the method of completion or can alternate between the interview and the form filler features. User input is linked to and saved in a database. The database saves all information input to the procedure forms. The user has the option to print completed forms.

Following the flow depicted in Figure 1, the user first chooses a method of completion. The form filler feature is chosen when the user is familiar with the procedural requirements and cognizant of the responses and forms that need be completed. If the form filler option is picked, then the user is presented with a menu of the procedural forms. The user picks the applicable form and then fills in an electronic version of the form . Look-up libraries are available at appropriate form fields to expedite completion of the procedure and provide for consistent input. Repeated information, e.g., TSR numbers, is automatically transferred to the other forms, so that it need only be entered once. The user has the option to switch to the interview feature to continue with the procedural requirements.

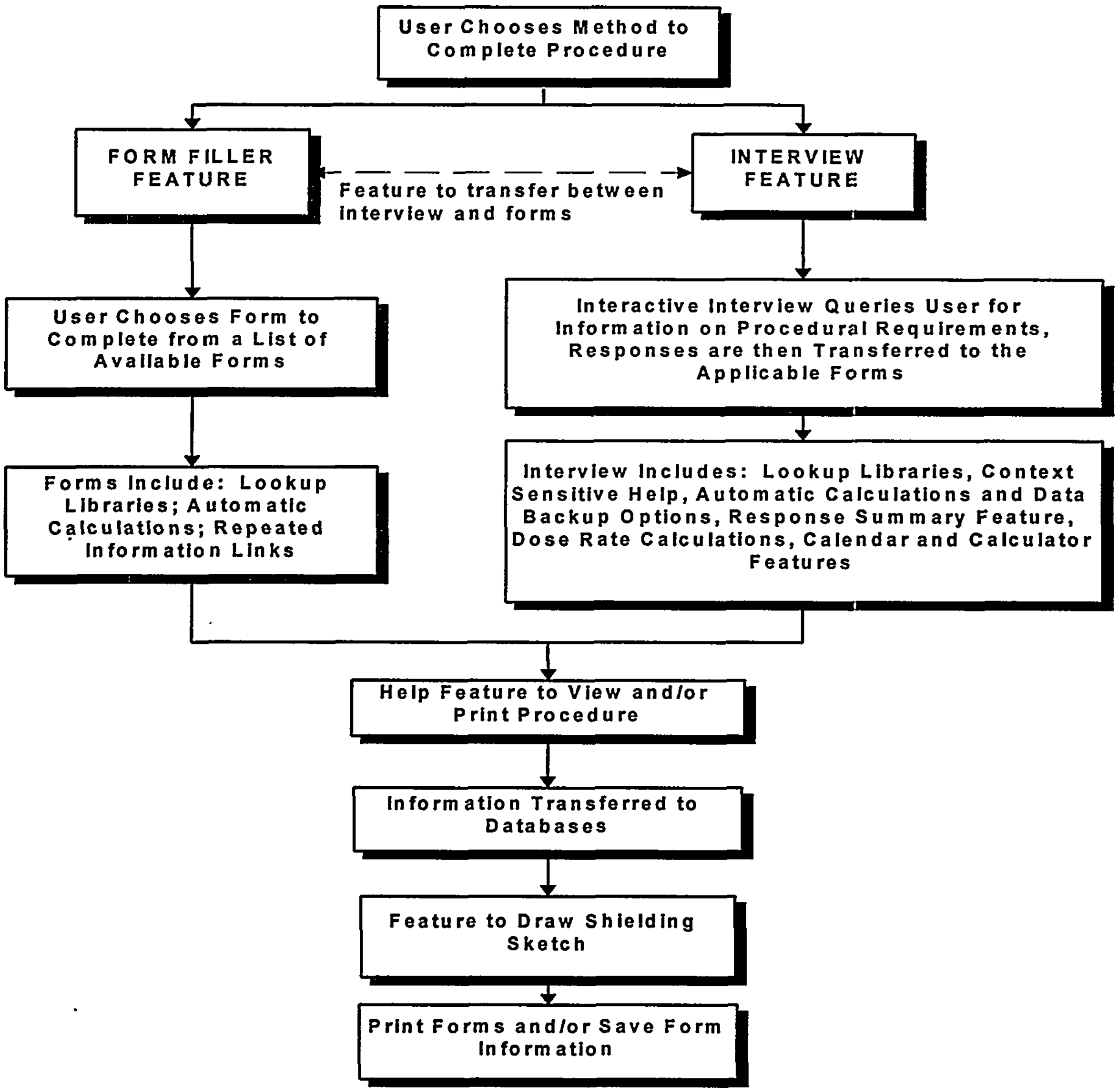


Figure 1

If the interview feature is chosen, the user is then led through a series of questions that walk through the steps of the procedure. Responses to the questions are transferred to the applicable forms through dynamic data exchange (DDE) links. The interview feature leads a user through the successful completion of all the applicable procedural requirements. Additionally, the interview includes a data backup feature with a range of specifiable backup times. If the user input responses are not saved for any reason, the backup files are automatically detected when the TSAP software is reloaded. The user is then given the option to load the responses from the backup file.

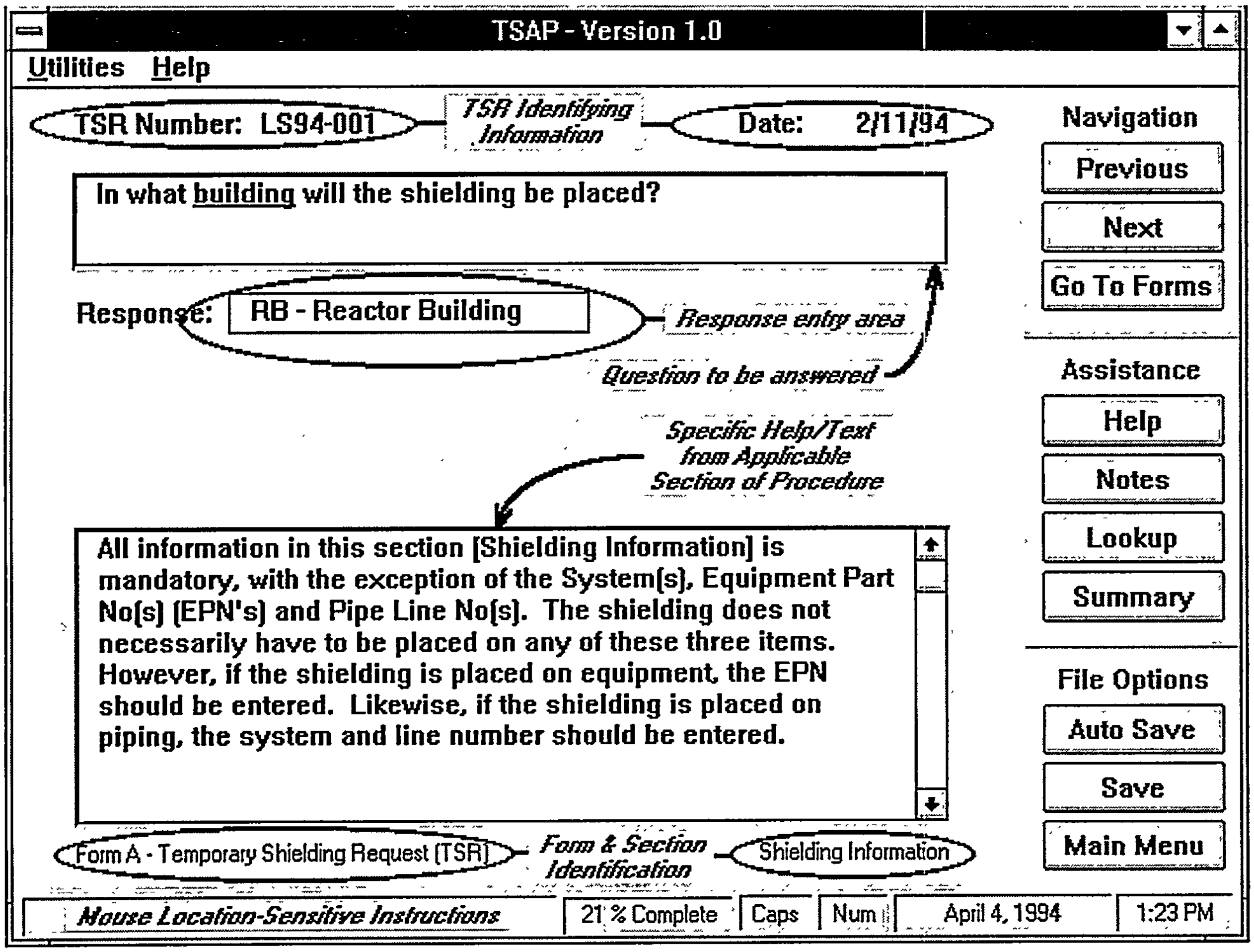
The interview feature provides the most assistance in completing the procedure. This feature enables a user with minimal familiarity with the procedure to quickly and accurately implement the procedure and complete the applicable forms. The interview feature asks questions appropriate to the specific shielding application and provides extensive user help. To ensure that the mandatory requirements are completed, mandatory responses must be entered before the user can proceed. The section of the TSAP applicable to a question is displayed directly below the question.

The interview feature also includes comprehensive help and look-up libraries to assist the user and expedite completion of the procedure. Extensive user help includes: providing applicable procedure sections that correspond to the questions, look-up libraries, automatic calculations, calculator and calendar features, a search routine, dose rate calculations, and a help feature containing a copy of the procedure and a glossary of definitions. A help feature enables the user to interactively review the procedure or use a glossary to search for definitions; a search routine is included to locate terms used in the procedure. The look-up library information is also available to the form filler feature. The interview includes the capability to review the procedure and print an uncontrolled copy. A help glossary of definitions is also available.

Included with the interview feature is a summary of all the user responses. This summary is contained in a single table that lists the interview questions and user input responses. From this table the user can go directly to any question by doubleclicking on the question with the mouse. The user may also enter additional notes concerning any of the responses; these notes are then saved together with user input and can be printed along with the forms.

When a form or forms are completed, the user is given the choice to save the information in the database and/or print the form(s). The shielding sketch form is also linked with computer aided drawing (CAD) software, which can be used to complete and electronically save a shielding sketch.

A sample screen from the TSAP software is shown in Figure 2. This screen shows an example of the user interface and options available in the interview feature of the program. This screen has been annotated with explanations of the information fields. Descriptions of various features are also given.



Menus:

Utilities	Help
Calendar	View Procedure Text
Calculator	Glossary
Dose Rate Calculation	About TSAP...
Shielding Types/Descriptions	

Buttons

Previous, Next	-	Used to navigate through the questions and responses
Go To Forms	-	Sends user to the Main Menu of the form filler feature
Help	-	Displays context-sensitive help for a given question.
Notes	-	Allows the user to enter notes specific to the TSR
Lookup	-	Brings up a Lookup Library containing all the expected responses to the question, e.g. lists of Buildings, Elevations, System Titles, etc.
Summary	-	Brings up a Summary Table which briefly states the Questions and Responses which have been entered.
Auto Save	-	Allows the user to set a timed backup of the data.
Save	-	Saves the data immediately.
Main Menu	-	Returns the user to the Main Menu.

Figure 2

Data Communication Structure

The data communication structure used by the TSAP software is depicted in Figure 3. The software serves as a front end to databases that store the field entries of the forms. The databases initially will be queried through the interview feature to select a previously completed Temporary Shielding Request (TSR). The user may then continue with and edit previously entered TSR's, or begin a new TSR.

When the interview feature is used, user responses are transferred to the applicable form fields via DDE links. From the form filler feature, responses entered in the form fields are saved in the databases.

The databases are an important feature of the software. Use of these databases enable effective tracking of current shielding installations and help make efficient use of historical data on past shielding applications.

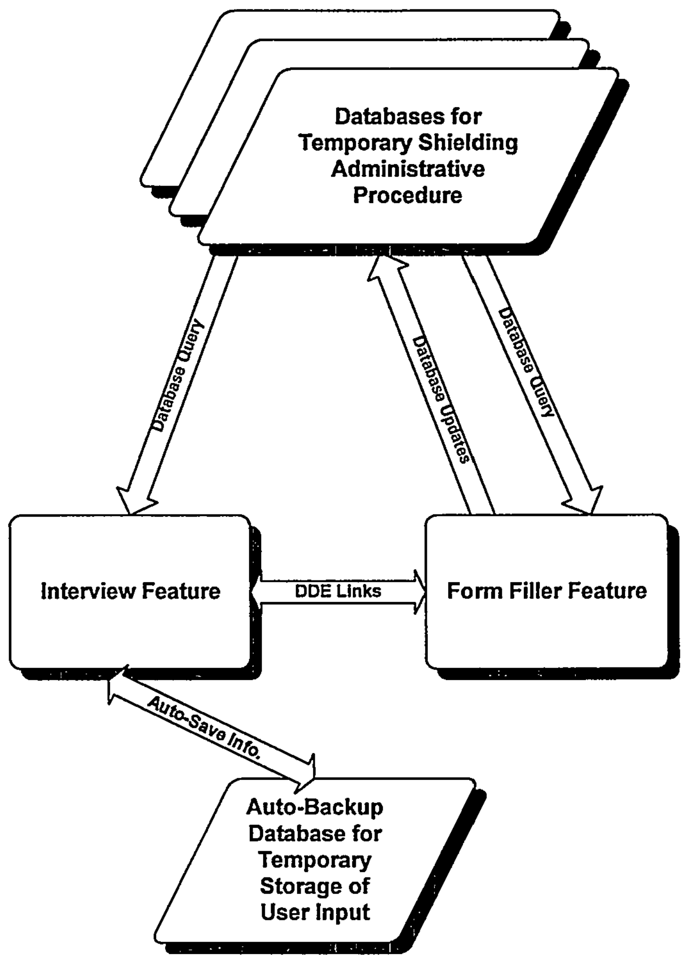


Figure 3

SUMMARY

A completely new Temporary Shielding Administrative Procedure was developed to promote the use of temporary shielding at Commonwealth Edison Company's six nuclear stations. This procedure also effectively addresses industry requirements for the use and control of shielding. This comprehensive procedure is implemented through the completion of a number of forms.

The procedure was computerized to expedite completion of the forms and to minimize the time and effort expended in completing the paperwork. The software was completed using an easy to use Windows interface and an innovative interview feature, with extensive help features to speed completion of the procedural requirements. This results in numerous benefits, including enhanced compliance with, and fast implementation of the procedure. Additional benefits are improved quality of the necessary paperwork and record keeping. This software has been well received by the Radiation Protection personnel responsible for implementing the shielding programs.

Author Biography

David Olson is a Senior Project Engineer at Sargent & Lundy. His responsibilities include the design, analysis, and in-plant testing of piping systems and associated structures. As part of this, he has been actively involved in the use and evaluation of temporary shielding. This includes the development of software for the control and evaluation of shielding. He has also been responsible for the development of administrative procedures for temporary shielding. He has a BS in mechanical/structural engineering from the University of Illinois and an MBA from the University of Chicago.

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PAPER 8B-4
DISCUSSION

Baum: What is the availability of this software? Is it something that you sell or is it provided as a service?

Olson: It is something that we sell. The status of the program is that we have a copy and one at Commonwealth Edison stations for beta testing. We are putting the finishing touches on it now, and the plan is to install it at all six stations this summer.

TSAP
PC Based Temporary Shielding Administrative Procedure

David E. Olson
Glenn E. Pederson
Sargent & Lundy
Peter N. Hamby
Commonwealth Edison Company

Objectives

- Promote Temporary Shielding Use
- Address Industry Requirements
- Utilize Advantages of PC
- Reduce Paperwork Burden
- Improve Turnaround Time

Industry Requirements

- NRC Requirements
  - NRC IE Information Notice No. 83-64, *Lead Shielding Attached to Safety-Related Systems Without 10 CFR 50.59 Evaluations*
  - NRC IE Information Notice No. 80-18, *10 CFR 50.59 Safety Evaluations to Radioactive Waste Treatment Systems*
- INPO Good Practice, *Temporary Lead Shielding*

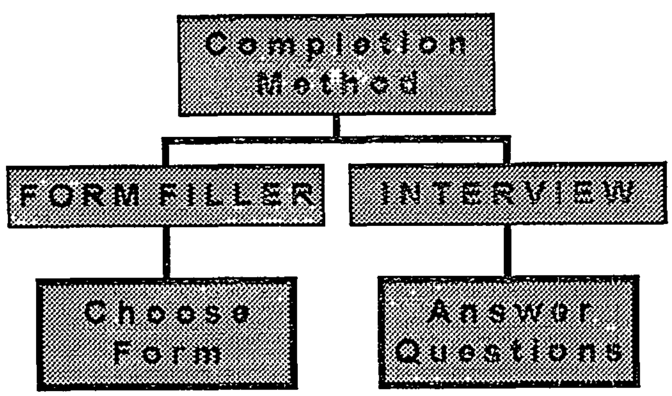
Procedural Requirements

- Temporary Shielding Request (TSR)
- Temporary Shielding Justification
- Evaluation of Temporary Shielding
- Field Installation of Temporary Shielding
- TSR Closeout

Optional Requirements

- Dose Benefit Analysis
- Temporary Shielding Sketch
- Temporary Shielding Inspection Requirements
- Temporary Shielding Tracking Log
- Dose Information
- Shielding Summary Form (Automatic)

TSAP Software

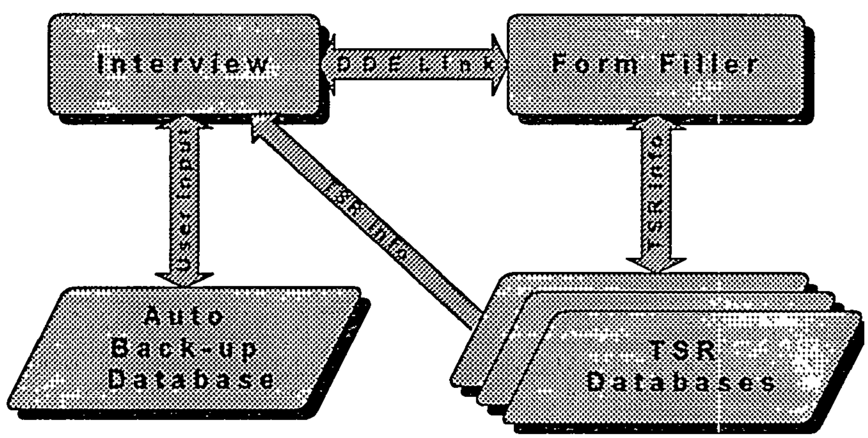


User Help

AVAILABLE HELP

- Lookup Libraries
- Context-Sensitive Help
- Info Links & Data Backup
- User Input Summary
- On-Line Procedure
- Auto & Dose Rate Calcs
- Calculator and Calendar

TSAP Database Structure



TSAP Benefits

- Effective Use of PC
  - Reduced Time to Learn Procedure
  - Automated Data Storage and Retrieval
  - Improved Control and Documentation
  - Reduced Completion Time and Effort
  - Strengthened Compliance with Procedure
  - Enhanced Compliance with Industry Requirements
- Increased Use of Temporary Shielding

ANI/MAELU ENGINEERING INSPECTION CRITERIA
8.3 ALARA

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ABSTRACT

The purpose of this criteria section is to provide guidelines for programs whose intent is to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA).

The success that has been achieved by applying ALARA concepts at nuclear power plants is clearly illustrated by the major reductions in the annual cumulative dose to workers at many sites over the last few years. This success is the combined result of the general maturity of the nuclear industry, the intensive study of dose reduction practices by industry groups, and the successful sharing of experience and practices among plants.

Source term reduction should be used as a primary ALARA mechanism. Methods which should be considered include: stellite and cobalt reduction, chemistry control, decontamination, submicron filters, zinc addition, hot spot reduction and permanent or temporary shielding.

Certain chemical control practices and operational activities can help to minimize the buildup of radioactivity on primary system piping and components. Specific ALARA practices should include the following:
failed-fuel action plans; consideration of increased exposure levels from hydrogen water chemistry (HWC) implementation; evaluations of "soft" shutdown techniques; and robotics.

The criteria presented in this section are derived by considering the parameters that affect dose, the variables that exist in station design and operation and the functions that can be addressed by station administrative actions. The Criteria are organized by the general areas or activities where ALARA practices can be effectively applied. These include:

- Management Commitment;
- ALARA Organization;
- Procedures;
- Goals;
- Work Planning;
- Feedback;
- Training;
- Engineering and Design;
- Source Term Reduction; and
- Specific Practices.

DISCLAIMER

BACKGROUND

The purpose of this criteria section is to provide guidelines for programs whose intent is to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA).

An effective ALARA program is, at its core, the practice of good health physics by all station workers and management. The health physics community has long recognized the prudence of avoiding unnecessary exposure to radiation and maintaining personnel exposures ALARA. The ALARA concept has evolved over the years. The concept during the early 1980's was at many sites a review function, separate from major plant activities. The relevant NRC Regulatory Guides 8.8[1] and 8.10[2] were generally prescriptive in nature, and were used in a "cook book" manner to determine the necessary elements, from a regulatory point of view, of an acceptable ALARA program.

The success that has been achieved by applying ALARA concepts at nuclear power plants is clearly illustrated by the major reductions in the annual cumulative dose to workers at many sites over the last few years. This success is the combined result of the general maturity of the nuclear industry, the intensive study of dose reduction practices by industry groups, and the successful sharing of experience and practices among plants.

The newly revised 10 CFR 20 Standards for Protection Against Radiation; Final Rule[3] published May 21, 1991, in the Federal Register established a requirement to formulate a program that would address ALARA for occupational doses and doses to the public. This requirement is stated in 10 CFR 20.1101 Radiation Protection Programs (b):

The licensee shall use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA).

The details of these procedures and engineering controls are left to the individual utility to develop and implement.

The new Part 20 defines ALARA in 20.1003 Definitions:

ALARA (acronym for "as low as is reasonably achievable") means making every reasonable effort to maintain exposures to radiation as far below the dose limits in this part as is practical consistent with the purpose for which the licensed activity is undertaken, taking into account the state of technology, the economics of improvements in relation to state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to utilization of nuclear energy and licensed materials in the public interest.

The "Total Effective Dose Equivalent (TEDE) ALARA" concept is one of the more significant new features introduced in the new Part 20. For example, a prospective "TEDE ALARA" evaluation should compare the additional DDE (deep-dose equivalent) estimated to be received because of the assumed reduced work efficiency due to wearing a respirator versus the estimated CEDE (committed effective dose equivalent) that is likely to result from an assumed intake if a respirator is not worn. Other nonradiological work place conditions should be considered. Documentation of prospective evaluations should include quantitative guidelines for each task and job which exceeds the individual or collective DDE guidelines established by each plant.

The primary ALARA objective is to reduce occupational exposures as far below the specified limits as is reasonably achievable. "Reasonably achievable" should be judged by considering the state of technology and the economics of improvements in relation to the projected benefits from these improvements. "Reasonable cost" has not been determined fully in terms of dollars per person-rem: Appendix I of 10 CFR 50,[4] codified in 1975, established, the values of $1000 per total body man-rem and $1000 per man-thyroid-rem (or such lesser values as may be demonstrated to be suitable in a particular case) as a figure to be used in cost-benefit analyses. It should be noted that the Appendix I criteria are applicable to doses to the general population. In practice, individual utilities are defining their own cost-benefit analysis values. Cost-benefit and optimization analyses are effective ALARA decision making tools, but they are highly dependent on assumptions, including the magnitude of the critical dollars-per-person-rem value used by nuclear power plants.

Recent ICRP publications have focused increased attention on the quantitative aspects of ALARA. ICRP 37[5] deals in detail with cost-benefit analysis and the optimization of radiation protection. The ICRP refers to the ALARA concept as the "optimization principle", which states that "all exposure shall be kept as low as reasonably achievable, economic and social factors being taken into account." ICRP 55[6] starts from the general concept of optimization of protection and shows how it can be implemented at different levels of decision and in different contexts using appropriate techniques, one of which is cost-benefit analysis. Optimization is the ICRP's term for ALARA. Some argue that they don't mean exactly the same thing, but ICRP disagrees:

It is now clear that *"keeping all exposures as low as reasonably achievable", "optimization of protection"* and *"ALARA"* are identical concepts within the ICRP system, even though the acronym 'ALARA' is not used by the Commission [the ICRP].[7]

The principle of reduction of exposures to levels that are ALARA is normally implemented in two ways. First, it may be applied to the design of facilities and modifications so as to reduce, prospectively, the anticipated exposure to workers. Second, it may be applied to actual operations so that work practices are designed and carried out to reduce the exposure of workers.

NUREG/CR-4254[8] addresses an examination of high-dose jobs, reliable equipment selection, radioactive waste handling improvements, and ALARA incentives at nuclear power plants.

EPRI's Radiation-Field Control Manual - 1991 Revision[9] identifies the following techniques among others, performed during various phases of a nuclear power plant cycle, as highly cost-effective in reducing occupational radiation exposure:

- Before power raising (startup): For BWRs, replace cobalt alloys in control blades, electropolish recirculation piping surfaces, and control oxygen during hot functional tests. For PWRs, electropolish channel heads and control chemistry during hot functional testing.

- During operation: For BWRs, minimize feedwater iron input, improve reactor water quality, and use zinc injection. For PWRs, control pH and use early boration or peroxide addition at shutdown.

- Refueling: Use low-cobalt materials in replacement fuel and cobalt-free pins and rollers in BWR control blades; use Zircaloy grids in replacement fuel for both types of reactors.

- Maintenance: For BWRs and PWRs, replace valves being refurbished with cobalt-free hardfacing alternatives and improve valve maintenance procedures to maintain cleanliness and remove debris.

- Special maintenance and repairs: For BWRs, decontaminate old piping, electropolish new piping, and pre-condition or water prefilm replacement piping. For PWRs, decontaminate, electropolish channel heads, and use low-cobalt Inconel for replacement steam generator tubing.

Plant life extension: For BWRs and PWRs, decontaminate the complete primary system.

The criteria presented in this section are derived by considering the parameters that affect dose, the variables that exist in station design and operation and the functions that can be addressed by station administrative actions. The Criteria are organized by the general areas or activities where ALARA practices can be effectively applied. These include:

- Management Commitment;
- ALARA Organization;
- Procedures;
- Goals;
- Work Planning and Scheduling;
- Feedback;
- Training;
- Engineering and Design;
- Source Term Reduction; and
- Specific Practices.

MANAGEMENT COMMITMENT

Management's commitment to ALARA should be formally stated in Corporate policy and evident in the resources provided and in the day-to-day conduct of plant operations. Senior Corporate and Plant Management should strongly support the ALARA program by becoming personally involved in monitoring radiation protection performance and holding workers, supervisors, and line managers accountable for their ALARA performance. ALARA outage coordination should be organizationally established.

ALARA ORGANIZATION

An organization with sufficient responsibility and staffing should be established to ensure that management's ALARA commitment is met. ALARA reporting responsibilities should be clearly delineated and assigned to a qualified professional with the authority to coordinate the ALARA program development and implementation. An ALARA committee (or equivalent) to coordinate activities is suggested.

PROCEDURES

The ALARA program including all important ALARA functions and methods should be described in plant policies and procedures. Procedures should describe those ALARA records, including ALARA reviews, audits, and documentation, which should be retained in accordance with ANI/MAELU Bulletin 80-1A.[10]

GOALS

Goals are established and used to determine the degree of success and to ensure consistent performance to the ALARA philosophy. Management should establish and review the achievement of challenging quantitative goals for collective radiation exposure per year, for outages, and for major or repetitive jobs. Significant variations from goals should be evaluated by management. A summary of plant performance relative to the ALARA goals should be provided to plant and corporate management on an annual basis. Periodically, goals should be re-evaluated in light of developing industry experience.

WORK PLANNING AND SCHEDULING

Appropriate planning for ALARA should be conducted for all work activities. Planning activities which should be considered include: surrogate (computer visual) tours, system walkdowns, pre-job briefings, shielding considerations, and interdepartmental interfacing with maintenance scheduling and planning, outage planning, and design and modification packages. Threshold levels should be established for cumulative and per-job exposures, above which ALARA planning should be conducted.

FEEDBACK

Feedback mechanisms should be in place to ensure that reviewing, commenting on and recommending changes to jobs and procedures based on lessons learned are routine station practices. Typical mechanisms are: ALARA program effectiveness and compliance audits, plan-of-the-day meetings and ALARA reports. Effective ALARA feedback provides the opportunity to benefit from experience in improving future performance. Job specific reviews include the identification and documentation of successes, problems, lessons learned, and specific recommendations for improvements identified in post-job debriefs.

TRAINING

Plant personnel should receive ALARA training appropriate to their position or job to ensure that management's commitment is met and that ALARA policies and procedures are implemented. Groups receiving targeted training should include, General Employees, HP technicians, operations and maintenance personnel, engineering and management. ALARA personnel should participate in industry ALARA meetings, as appropriate. ALARA training should include the use of dry runs and mockups as appropriate.

ALARA training is covered in Section 2 of ANI/MAELU Engineering Inspection Criteria.[11] The ALARA provisions which have been incorporated into the various sections of the ANI/MAELU training criteria are summarized below.

General Employee Training

- ALARA objectives and station ALARA policy
- Basic exposure reduction methods
- The manner in which the ALARA program is implemented at the station
- The importance and responsibility of individual workers to maintain their doses ALARA

HP Technician Training

- ALARA concepts
- ALARA program and procedures
- ALARA responsibilities
- Job review and coverage
- Dose reduction techniques and job/craft specific instructions to reduce dose

Operation and Maintenance Technical Training

- The ALARA concept and its purpose
- ALARA responsibility and incentives
- ALARA techniques such as planning and briefings, use of mock-ups and decontamination
- Importance of utilizing the technical services and advice of the radiation protection staff

Engineering ALARA Training

- Design review and design considerations including radiation exposure considerations in selection and placement of equipment (reliability, maintainability, accessibility)
- Dose reduction techniques including radiation exposure considerations in building layout, radiation zone control, ventilation control, operation and maintenance
- Cost effectiveness evaluation methods

Management Staff

- Importance and overall justification of the ALARA program
- Procedures for evaluating ALARA performance
- Importance of utilizing the technical services and advice of the radiation protection staff
- Management system for ALARA program implementation, goal development and measurement, and performance evaluation

ENGINEERING AND DESIGN

Design, both initial and in the form of plant modifications, plays an important role in maintaining personnel exposures ALARA. USNRC Regulatory Guide 8.8[1] provides guidance on design features which can contribute to ALARA.

USNRC Regulatory Guide 8.19[12] provides a mechanism to project, during the design stage, the estimated annual radiation exposure to station personnel. Person-rem estimation is also an important part of the overall, ongoing radiation protection review of dose-causing activities to reduce exposures. The dose estimation process requires a working knowledge of the principal factors contributing to exposure, and methods and techniques for dose reduction.

Programmatically, ALARA involvement in the design process should take two principal forms. ALARA specialists should be involved in the design/modification review process for the purpose of applying their specialized expertise towards the identification of design features which can be cost effectively incorporated and result in significant dose savings. Their involvement in this process sets up an ALARA advocacy. ANI/MAELU also believes that personnel with ALARA responsibility should take an active position with respect to design by proposing modifications and design changes which can result in cost effective dose savings.

Certain PWR plant modifications, such as main coolant RTD bypass piping removal, have been effective at removing so-called crud traps which increase radiation fields by allowing activated corrosion products to accumulate.

SOURCE TERM REDUCTION

Source term reduction should be used as a primary ALARA mechanism. Methods which should be considered include: stellite and cobalt reduction, chemistry control, decontamination, submicron filters, zinc addition, hot spot reduction and permanent or temporary shielding.

Cobalt Reduction

An aggressive cobalt reduction program involves minimizing cobalt impurity in structural materials; eliminating cobalt from in-core materials, particularly fuel and control rod drive mechanisms; and substituting cobalt-free hardfacing alloys for the "Stellites" typically used in nuclear plant valves. There have been renewed efforts to develop low-cobalt or cobalt-free alloys as alternatives to the cobalt-based alloys in order to reduce exposure of maintenance personnel to Co-60.[16]

EPRI NP-6708[16] addresses recent advances in radiation-field control techniques which contribute to reduced occupational radiation exposures. There are four aspects of radiation field control which correspond to the fundamental processes involved in activation, transport, and deposition of wear and corrosion products in primary systems. The most successful control programs feature at least three of the four methods listed below:

- Controlling the source,
- Controlling transport and activation,
- Controlling out-of-core deposition, and
- Decontamination

Chemistry Control

Water chemistry control has an indirect, multi-faceted and important impact on radiation exposures to workers. Good water chemistry is the key to minimizing the formation and release of corrosion products into reactor water. It is notable that there is a strong correlation between absence of radiation hot spots in crud traps in BWRs and good water chemistry. Minimizing impurities in BWR reactor water via feedwater, pH control, condensate polishing and Reactor Water Cleanup operation is essential to controlling radiation field buildup. For PWRs, pH control is crucial to minimize shutdown radiation fields.[16]

PWR shutdown radiation fields are primarily the result of cobalt isotopes 58 and 60. Co-60 is formed by neutron capture of naturally-occurring Co-59 which is found throughout PWR materials as impurities (steam generator tubes) and hardening alloys (valves). Nickel is a major component of alloy 600 steam generator tubes and, when activated, becomes Co-58. These elements are released to the primary coolant by wear and corrosion processes. As the coolant is transported into the core the elements are activated and then deposited throughout the reactor coolant system.

The higher the coolant pH in PWRs can be maintained the lower corrosion product release and transport rates will be. Lithium hydroxide is added to increase pH, but its effect is limited by boric acid used for reactivity control and restricted by the potential for accelerated fuel clad corrosion. In particular, raising the pH from the previously typical 6.9 to 7.1-7.4 (by increasing lithium concentration) has been effective. Swings in pH due to lithium fluctuations have also been observed to contribute to radiation field buildup, so precise controls are important.

During PWR plant startup and shutdown, coolant chemistry can also be controlled to enhance radioactivity release and removal. Hydrogen peroxide or other oxygenating agents are added to speed up the release of cobalt from the core. Without doing so, the release would occur when the vessel head is removed, leading to increased dose rates during refueling operations. Reactor coolant and purification system flow should be maximized to enhance corrosion product removal. Boron concentration should be increased to refueling concentrations as soon as possible to achieve acidic, reducing conditions which dissolves nickel and cobalt from non-core surfaces.

Close coordination at all times between operations, chemistry, radiation protection and outage planning personnel is necessary to ensure the success of chemistry regimes designed to reduce radiation fields. Radiation Protection personnel need to be especially aware of planned chemical injections which may change dose rates significantly until released corrosion products are removed.

EPRI has coordinated development of two consensus guidelines (*PWR Primary Water Chemistry* and *PWR Primary Shutdown and Startup Chemistry*) which incorporate the latest research results and plant experience necessary to develop a plant specific program.

Decontamination

Decontamination is gaining increased acceptance within the nuclear industry as a cost effective method to reduce out-of-core radiation fields. Considerable resources have been directed toward utilizing chemical decontamination techniques to reduce radiation exposures during special maintenance and ISI work. Decontamination technology has been successfully used in BWR recirculation system piping repair, PWR steam generator maintenance, and other partial system applications.

Beyond partial system application is chemical decontamination of the entire primary system. A full-system decontamination (FSD) qualification program is being developed by EPRI and several U.S. utilities. Once FSD has been qualified and demonstrated, it should help utilities achieve lower exposures.

Mechanical decontamination methods are applied when the exposed surfaces are easily accessible. Some examples are:

- High pressure jet spalling;
- Abrasive blasting (e.g. ice, carbon dioxide, plastic bead);
- Ultrasonic cleaning;
- Liquid honing;
- Strippable coatings; and
- Vacuuming (wet or dry).

Submicron Filters

The use of fine, absolute rated filters to reduce corrosion product transport in the primary system can be effective in the decrease of out of core radiation fields. A substantial fraction of activated corrosion products exist as sub-micron particles. Submicron filtration increases the removal of radioactive corrosion products or the removal of corrosion products prior to activation. The 20 micron (nominal) pore size of early days of PWR operation has given way to pore sizes as low as 0.2 micron in systems such as Chemical Volume Control System, Seal Water, Boric Acid, Spent Fuel Pool/Reactor Water Purification Systems and Radwaste.

Zinc Addition

The addition of zinc to the primary system of BWRs may reduce the cobalt plate- out on primary piping and, therefore, reduces dose rates from the piping.

Specific Practices

Certain chemical control practices and operational activities can help to minimize the buildup of radioactivity on primary system piping and components. Specific ALARA practices should include the following:

failed-fuel action plans; consideration of increased exposure levels from hydrogen water chemistry (HWC) implementation; evaluations of "soft" shutdown techniques; and robotics.

Failed Fuel

Care should be taken to prevent fuel damage by proper primary chemistry and by an aggressive program to prohibit any loose parts or foreign objects from getting into the primary system during maintenance and refueling operations.

HWC

Several BWRs have implemented HWC to protect reactor piping and vessel internals from stress corrosion cracking with a subsequent increase in radiation field dose rates. An evaluation should be performed to address the increased radiation field intensity resulting from HWC implementation.

"Soft" Shutdown

The nature and magnitude of activated corrosion product spiking following shutdown of a BWR are functions of a plant's specific water chemistry as well as shutdown procedures and subsequent steps during an outage. The U.S. BWR industry experience has shown that a "soft", controlled shutdown which minimizes boiling in the core and maximizes cleanup is effective in reducing plant radiation levels. Guidelines for reducing the concentration of activated corrosion products in reactor water during refueling outages have been suggested. These corrosion products and associated activities can affect outage work adversely in several ways. For example, they can:

- Increase dose rates on the primary system and the refueling floor, increasing personnel exposure;
- Delay refueling activities by reducing water clarity and, therefore, visibility; and
- Contaminate reactor cavity walls and/or other components in the pool, requiring decontamination at the end of the refueling outage.

To reduce the exposure to maintenance personnel during outages, it is suggested that to the extent practicable, the U.S. BWR industry consider the suggested guidelines when planning shutdowns.

Robotics

Robotics, automated equipment, and remote reading instrumentation should be used where practical such as for steam generator channel head entries, reactor coolant pipe welding, decontamination, and surveillance in high radiation areas.

CRITERIA

Management Commitment

8.3.1 Management's commitment to ALARA should be formally stated in Corporate policy and evident in the resources provided and in the day-to-day conduct of plant operations.

ALARA Organization

8.3.2 An organization with sufficient responsibility and staffing should be established to ensure that management's ALARA commitment is met.
8.3.2.1 ALARA reporting responsibilities should be clearly delineated.
8.3.2.2 An ALARA committee (or equivalent) to coordinate activities is recommended.
8.3.2.3 Outage coordination should be organizationally established.

Procedures

8.3.3 The ALARA program including all important ALARA functions and methods should be described in plant policies and procedures.

8.3.3.1 Procedures should describe those ALARA records, including all ALARA reviews, which should be retained in accordance with ANI/MAELU Bulletin 80-1A.[10]

Goals

8.3.4 Management should establish and review the achievement of challenging quantitative goals for collective radiation exposure per year, for outages, and for major or repetitive jobs. Significant variations from goals should be evaluated by management.

Work Planning

8.3.5 Appropriate planning for ALARA should be conducted for all work activities.

8.3.5.1 Threshold levels should be established for individual, cumulative, and per-job exposures, above which ALARA planning should be conducted.

Feedback

8.3.6 Feedback mechanisms should be in place to ensure that reviewing, commenting on and recommending changes to jobs and procedures based on lessons learned are routine station practices.

8.3.6.1 Post-job debriefs should be conducted whenever ALARA planning was conducted or whenever an unexpected exposure or a significant problem was encountered.

8.3.6.2 ALARA program effectiveness and compliance audits should be conducted.

Training

8.3.7 Plant personnel should receive ALARA training appropriate to their position or job to ensure that management's commitment is met and that ALARA policies and procedures are implemented.

8.3.7.1 ALARA personnel should participate in industry ALARA meetings.

8.3.7.2 ALARA training should include the use of dry runs and mockups as appropriate.

Engineering and Design

8.3.8 Engineering designs and modifications should be reviewed for the purpose of identifying cost-effective changes which would result in reduced personnel exposures.

Source Term Reduction

8.3.9 Source term reduction should be used as a primary ALARA mechanism.

8.3.9.1 Source term reduction methods should include: chemistry control, stellite and cobalt reduction, hot spot reduction, and chemical decontamination of systems.

8.3.9.2 An operational reactor coolant chemistry control program should be developed which minimizes the release, activation, transport and deposition of corrosion products which lead to increased dose rates.

8.3.9.3 A plant-specific shutdown and startup primary chemistry control program should be developed which maximizes the release and removal of radioactivity.

8.3.9.4 Primary system filtration and ion exchange system improvements to enhance radioactivity removal should be evaluated.

- Ensure purification systems are reliable and maintained on line to the fullest extent within operational constraints.
- Evaluate the use of improved filter designs, submicron filters and ion exchange to enhance the removal of corrosion products.

8.3.9.5 Radiation Protection should be notified before performing chemical transients which could change dose rates.

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Leland Schneider is a Principal Health Physicist in the Engineering Department at American Nuclear Insurers with 24 years of nuclear power related experience. During his present 14 years at ANI, Leland has conducted 118 plant inspections including 72 BWR inspections at 27 different BWRs and 46 PWR inspections at 26 PWRs. ANI's inspection activities focus on radiation protection, radioactive waste, effluent, and radioactive environmental monitoring programs at U.S. nuclear power plants. Before joining ANI, he spent 2 years as Radiation Protection Supervisor at Illinois Power Company's Clinton Power Station and 8 years at Nebraska Public Power District's Cooper Nuclear Station, where he started as one of the original five Health Physics Technicians and was promoted to Plant Health Physicist. He has a B.Sci. in Education and taught high school Physics and Chemistry in Nebraska for 3 1/2 years before starting at Cooper in 1970.

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PAPER 8B-5
DISCUSSION

Uchida: The career and personal exposure record, along with health inspection record, are they available using a person's social security number or something like that?

Schneider: ANI has an engineering information bulletin (80-1A) that deals with records retention. Before the new part 20, all records - Form 4, "Occupational External Radiation Exposure History; Form 5, "Current Occupational External Radiation Exposure;" and termination reports were obtainable by request from the utilities or the NRC. Under the new part 20, plants are not required, except for planned special exposures, to obtain the total previous lifetime exposure of individuals, although ANI still suggests that it be done and maintained. We have a bulletin that really addresses all the things that we want there, and I will make that available to you.

Crouail: Did you break any contract of insurance with utilities because of nonobservance of ALARA guidelines?

Schneider: No, we have not since the implementation of the pools in 1957. In the U.S., to do that under the Price-Anderson Act and the federal regulations (10 CFR part 140), the utility would have to shutdown and look at either self-insuring or getting some other avenue of insurance. Right now, we provide the only nuclear liability insurance. To terminate a policy would result in a highly political situation, and we have had good success when needed to really achieve what we wanted without having to do that. That avenue would be available.

Baum: A related question -- how are the statistics on litigation? Are we getting many cases per year, and how many are being fought?

Schneider: We just received our 166th claim. I'm sure you've heard about the claim at San Onofre, what we call the "Tang case." A female NRC inspector brought a suit against our insured, and that was the 165th claim. What I'm going to say on this is that those of you who can make it to the joint BWR-PWR Owner's Group Meeting in July in Denver can hear Jerre Forbes, who is our Technical Director of Liability Claims make a presentation on litigation claims. I'm not authorized to say a lot about the claims ends of things. Roughly all but 30 of the 166 claims have been closed since the inception of the Pools in 1957. Actually, we had fewer than 40 claims through TMI in 1979, and we've had this rise to 166 since. The following does not represent ANI's position, but I personally feel that it's going to get tougher, and that we are going to see more claims without a lot of technical merit, but we will have to see what emerges.

Miller: Thank you, Leland. I would say even though his criteria are only guidelines, they are very helpful for the ALARA person in the field because you can reference those guidelines as requiring RP to be notified when major changes occur. Although I would also clarify that he normally would not cancel you, but as a customer, he can add a surcharge above the average $500,000 base premium for a typical unit plant, based on nine engineering parameters. We call them penalties. They can raise the premium approximately $50,000 or $60,000 per year. Eight of the nine subfactors are radiological or dose-related items.

ALARA DEVELOPMENT IN MEXICO

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ABSTRACT

Even though the ALARA philosophy was formally implemented in the early 1980s, to some extent, ALARA considerations already had been incorporated into the design of most commercial equipment and facilities based on experience and engineering development. In Mexico, the design of medical and industrial facilities were based on international recommendations containing those considerations. With the construction of Laguna Verde Nuclear Power Station, formal ALARA groups were created to review some parts of its design, and to prepare the ALARA Program and related procedures necessary for its commercial operation. This paper begins with a brief historical description of ALARA development in Mexico, and then goes on to discuss our regulatory frame in Radiation Protection, some aspects of the ALARA Program, efforts in controlling and reducing of sources of radiation, and finally, future perspectives in the ALARA field.

INTRODUCTION

The peaceful uses of atomic energy in Mexico include medical, industrial, research projects, and supplying electrical power across the country. The main users are the Comision Federal de Electricidad (CFE) with the Laguna Verde Nuclear Power Station; the Instituto Nacional de Investigaciones Nucleares (ININ), at the Nuclear Center; the Instituto Mexicano del Petroleo (IMP), and many major hospitals, located in big cities like Mexico City, Guadalajara, and Monterrey.

Among those institutions, CFE at the Laguna Verde Station is the only one that has a formal ALARA program. For this reason, this paper mainly focuses on the development, current work, and future perspectives of ALARA topics there.

Laguna Verde has two 675 MWe BWR-5 units from General Electric, with turbine-generators supplied by Mitsubishi. Construction started by 1976 in the Laguna Verde area, 70 km north of Veracruz City, on the Gulf of Mexico. The original architect-engineer, Burns and Roe later was replaced by Ebasco, but the CFE was directly responsible for construction since the beginning. Unit 1 has been connected to the grid since April 1989, and entered commercial service in July 1990. In March 1994, its third refueling outage ended. Unit 2 is in the pre-operational test stage, and it is planned to start operations by the middle of 1995.

REGULATIONS

Based on the regulatory law on nuclear affairs, (published in February 1985) as a part of article 27 of the Mexican Constitution, the National Nuclear Safety and Safeguards Commission (Comision Nacional de Seguridad Nuclear y Salvaguardias, CNSNS) as the regulatory body in Mexico, is responsible for issuing licenses to use radioactive and nuclear materials; for assuring the implementation of radiological safety systems in all activities involving nuclear energy, including planning, operations, waste management, and decommissioning; and for inspecting all licensed facilities during normal operations and abnormal situations.

In November 1988, Mexican government issued the General Nuclear Safety Regulation (Reglamento General de Seguridad Nuclear[1]), that shall be observed by those persons and companies working with radioactive materials in Mexico, in accordance with specific requirements contained in guidelines and rules.

The requirements for radiation protection contained in Mexican Regulations, conform entirely with the recommendations of the International Commission on Radiation Protection (ICRP) and the Basic Safety Standars for Radiation Protection of the International Atomic Energy Agency (IAEA). The full dose limitation system of justification, optimization and setting of individual limits is included in the regulation. In dose limitation, optimization is the most complicated task because is greatly influenced by personal interpretation, social concerns and technical and economic resources. Because of limited applicability of quantitative methods, this kind of optimization is not required by law, however optimization requierements are complied with through design and operation in conformity with existing international standards.

In the case of Laguna Verde Station, since the licencing stage, Mexican government decided to use US Nuclear Regulatory Commission standards and guidelines, specifically radiation protection and ALARA practices are based on 10 CFR parts 20, 50 and related regulatory guides.

OPTIMIZATION

Optimization process shall be directed to achieve reasonably low individual and collective doses on the conditions that individual dose equivalent limits are not exceeded. This is a logical step if it is assumed that every exposure, irrespective of the dose, has a harmfull effect.

Since Laguna Verde was designed by 1975, when the ALARA philosophy was not implemented yet, its design was based on mayor radiation sources separation and shielding, but without almost any specific ALARA analysis. However since 1984 the Engineering office of CFE through its Applied Physics group performed some limited ALARA reviews of systems and areas inside the plant. In adition, since 1987 the ALARA group of the Operations Team started to work in the confirmatory ALARA review of design, and the preparation of the ALARA program and related procedures.

Quantitative Optimization

Differential cost-benefit analysis proposed in ICRP publication 37[2] can be used as a tool in the process of quantitative optimization, this method commonly is presented as a system of mathematical expressions, relating to the function of optimization and also to constraining or limiting functions. In such equations are involved the parameters alpha and beta, which are numerical constants related with the monetary value of the collective doses of the population exposed as a result of certain practice. The alpha parameter should be unique and invariable in a given country, while the appropiate selection of the multiple values of the beta parameter makes it possible to take into account sociological considerations, such as the type of persons exposed (workers, the public, nationals or foreigners), the dose levels (high or low), the spatial and temporal distributions of exposure and the normal and random nature of events.

Alpha and Beta Values at Laguna Verde

In the case of Mexico, to date our regulatory body has not established a national value for the alpha parameter. For that reason, Laguna Verde has established and used its own values. During design stage we used a value of US$ 100,000/Person-Sv (US$ 1000/Person-rem) in all analyzed cases. Recently, we changed that value to US$ 530,000/Person-Sv (US$ 5300/Person-rem)[3] for all groups of dose levels. In the future we are planning to implement a basic alpha value for Laguna Verde, consistent with the value fixed by the CNSNS (when they do), and different beta values depending on the dose level group of the workers involved in the assessment.

Qualitative Optimization

Due to difficulties in implementing these quantitative methods, we considered that the use of qualitative optimization, based on knowledge, experience, and judgement is a powerful tool. Hence, at Laguna Verde Station we have been developing data bases to identify the main irradiation sources, the mistakes in design, critical groups, and related information. Even this process is very difficult for us at present, because we do not have enough historical information because Laguna Verde is the first nuclear power plant in operation in Mexico. Most of the radiological conditions are new for us, so we have been using information supplied directly from other nuclear plants, or organizations like the Institute of Nuclear Power Operations (INPO), and the IAEA, together with advice from experts.

DOSE REDUCTION AT LAGUNA VERDE

At Laguna Verde, the work of the Radiation Protection group focuses on preventing workers receiving doses higher than reasonable values, and ones that always lower than administrative limits (1 Rem/quarter and 4 Rem/year whole body dose). Efforts in collective dose reduction are directed by the RP Analysis (ALARA) group, and include two kinds of activities: Implementation of the ALARA Program (related to work practices), and the Source Reduction program (related to controlling radioactive sources).

ALARA Program

The ALARA Program at Laguna Verde is focused on maintaining doses at optimum levels, based on workers' training and work practices mainly in high radiation and contamination areas, or long-stay jobs. Implementation of the program is one of the biggest challenges at Laguna Verde because workers have not yet had enough "ALARA Culture," even recognizing that important advances have been achieved. To help in solving this problem, our periodic ALARA meetings have been very helpful for plant workers. At those meetings, we discuss the health impact of radiation, the benefits of work planning, and we resolve any doubts they may have about our dose-reduction activities. For contractor workers, it is more complicated to control their doses because, most often, they are not well trained in radiation protection practices. This problem becomes critical during refueling outages when the number of contractor workers increases significantly.

Another important part of the ALARA program is dose tracking by work groups and high dose job. For example, during 1992 we established annual goals for normal operation (2 person-Sv) and refueling outage (3 person-Sv). In addition, the goals for work groups during normal operation were distributed as follows: Maintenance (with 60 % of the total estimated dose), Radiation Protection (25%), Operations (7%), and Others (8%). We note that Radiation Protection group at Laguna Verde includes groups concerned with decontamination and with handling radioactive waste.

As an example of dose tracking by job during 1991, Laguna Verde personnel got about 0.5 person-Sv in a collective dose during repairs to leaks in the Turbine building. Therefore in 1992, General Management decided to implement a priority program to attack only the most significant leaks, based on collective doses, increases in water inventory, increases in environmental temperature around vital areas, and repair costs. In addition, we budgeted a maximum dose for repairing leaks in that year, and established monthly goals to control this job. Consequently, the 1992 total collective dose for such repairs was near 0.4 person-Sv. This value was not as low as we wished. Therefore, since our second refueling outage, at the end of 1992, Laguna Verde started a valve life load seals replacement program for valve load seals that decreased the total number of leaks and the 1993 total collective dose to about 0.2 person-Sv.

Other aspects of our ALARA program are the ALARA planning and ALARA post-job review, undertaken when estimated and/or obtained collective doses are higher than 0.01 person-Sv (1 person-rem), because this allows us to plan in more detail the activities involved in particular jobs, and to fix individual and collective doses for each part of the job. Doses and dose-reduction devices or techniques for all jobs subject to ALARA planning are tracked and documented in data bases to have the information for future jobs.

Source Reduction Program

Source reduction activities at Laguna Verde are directed to controlling and reducing radiation levels inside the plant, that are due mainly to activated corrosion products generated by the wear-out of components inside pipes, valves, and other equipment. Our source reduction program in preparation) has three main parts: materials control, hot spots reduction, and systems decontamination.

Materials Control

Materials control at Laguna Verde is currently focused on reactor coolant pressure systems (RCPS) and vessel internals because they are the main contributors in the increase in cobalt inventory. The following activities are included:

a. Review of bid specifications, and lists of spare parts to specify components with low cobalt alloys.

b. Replacement of valves, or seats of valves, in RCPS depending on their location, kind of valves, diameter, and costs.

c. Replacement of pins and rollers in control rod drives (CRD) in the vessel of Unit-2 (not in operation yet), and initial studies to replace them in the vessel of Unit-1.

Hot Spot Reduction

At present, we are preparing the hot spot reduction program. It will be based on the following criteria:

a. Identification and classification of hot spots inside the plant, depending on dose-rate readings, and estimated collective doses for jobs near each hot spot.

b. Selection of the best way to remove hot spots: by operational actions, like systems operations maneuvers or flushing; by mechanical removal using brushes; by engineering modifications, to suppress dead legs, or relocate valves or components; or by temporary shielding.

c. Preparation of data bases to identify each hot spot, the actions taken to eliminate them, and their recurrence if they appear periodically.

In addition to that program, during our third refueling outage (January-March 1994) we replaced about 35 small drain valves (kerotest valves) connected to the recirculation (RRC) and reactor water cleanup (RWCU) systems, removing hot spots with dose-rate values in the order of 3 to 150 R/hr.

Systems Decontamination

Systems decontamination is focused on primary systems that show important rates of increase in radiation levels around them, due mainly to the deposition of corrosion products inside pipes. The criteria used to decide whether to decontaminate a system are based on measurements of contact dose rates around systems, trends with time in those measurements, and the collective dose projected from jobs around those systems.

Unit-1 has followed the BRAC points program for primary systems described in NEDC-12688[4] since the beginning of commercial operation, showing about 95 mR/hr/FPYO[5] after two cycles of operation. Even though this value is good, during an SCRAM in March 1993, we found very high dose rates in the primary systems, specially in the reactor water cleanup system (RWCU) where dose rate values were in the order of

R/h. For this reason, we decided to undertake a chemical decontamination of that system; preliminary decontamination factors are about 6.

In addition to chemical decontamination, we have been mechanically decontaminating (by brushing) sections of equipment and floor systems, successfully removing hot spots.

NEW 10CFR20 IMPLEMENTATION

Our regulatory body and Laguna Verde recently started to review the impact that the eventual implementation of Regulations contained in new 10CFR20 on the Laguna Verde Radiation Protection Plan. Also, since the second semester of 1993, the Radiation Protection group at Laguna Verde started a plan to review their procedures and equipment to comply with these new requirements. As a part of this plan, we started a program to teach workers about the benefits in allowing small internal contamination in situations when the total dose could be reduced.

FUTURE PERSPECTIVES AND GOALS

One of the reasons that our annual collective doses (about 0.5 person-Sv) are higher than the U.S. plant average is because our workers (mainly contractors) have not enough "ALARA Culture," due to the reduction of nuclear applications in Mexico, and the resulting lack of experienced workers.

In the author's opinion, the following might be the further targets for Laguna Verde:

a. Implement widely our ALARA program among plant and contractor workers.

b. Keep doses quite below the dose limits to keep the average radiological risk at low levels (reduce or maintain dose-rate levels inside the plant).

c. Provide a good working environment, similar to that in modern industries where radiation is not involved (reduce or keep the levels of contamination low inside the plant).

d. Ensure that the dose rates, concentrations, and collective doses achieved compare favorably to similar plants elsewhere.

e. Reduce our collective annual doses to levels near the average in similar plants.

ACKNOWLEDGMENT

This work presents the opinion of the author, and does not necessarily represent the position of Laguna Verde in any of these matters. Nevertheless, the author appreciates the support received from the Laguna Verde Radiation Protection group, especially from its manager, Sergio Zorrilla.

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SESSION 9

DECOMMISSIONING OF NUCLEAR POWER PLANTS

Chair:

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SHIPPINGPORT STATION DECOMMISSIONING PROJECT ALARA PROGRAM

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ABSTRACT

Properly planned and implemented ALARA programs help to maintain nuclear worker radiation exposures "As Low As Reasonably Achievable."

This paper describes the ALARA program developed and implemented for the decontamination and decommissioning (D&D) of the Shippingport Atomic Power Station. The elements required for a successful ALARA program are discussed along with examples of good ALARA practices.

The Shippingport Atomic Power Station (SAPS) was the first commercial nuclear power plant to be built in the United States. It was located 35 miles northwest of Pittsburgh, PA on the south bank of the Ohio river. The reactor plant achieved initial criticality in December 1959. During its 25-year life, it produced 7.5 billion kilowatts of electricity. The SAPS was shut down in October 1982 and was the first large-scale U.S. nuclear power plant to be totally decommissioned and the site released for unrestricted use. The Decommission Project was estimated to take 1,007 man-rem of radiation exposure and $98.3 million to complete. Physical decommissioning commenced in September 1985 and was completed in September 1989. The actual man-rem of exposure was 155. The project was completed 6 months ahead of schedule at a cost of $91.3 million.

Key lessons learned in the application of ALARA at the Shippingport Station Decommission Project were:

- Incorporate sound ALARA practices and detailed man-rem estimates into the initial D&D planning.
- Include the requirement for ALARA practices and detailed man-rem estimates in all work activities.
- Monitor and enforce effective radiation control work practices.
- Establish an aggressive man-rem reduction program.
- Obtain the endorsement and continued full support of top management.

INTRODUCTION

ALARA planning for the Shippingport Station Decommissioning Project (SSDP) was integrated into all phases of the project. It began with the development of a detailed, twelve-volume SSDP Decommission Plan completed in 1983. A two-volume detailed cost and man-rem exposure estimate was also produced. The man-rem estimate was based on the estimated man-hours required to decommission the Shippingport Atomic Power Station (SAPS) and release the site for unrestricted use. The plan formed the basic requirement document used by the U.S. Department of Energy for the selection of the SSDP Decommissioning Operations Contractor (DOC).

The total man-rem estimate for the Shippingport Station Decommissioning Project was 1,007 and included a one-year surveillance and maintenance period prior to the start of physical decommissioning work. Figure 1 represents the estimated man-rem exposure versus time. The estimated man-rem was based on the measured radiation dose rates that existed at the end of nuclear plant operations in 1982. Figure 2 is a typical radiation survey showing dose rates in the reactor compartment and equipment (boiler) chambers. The low dose rates are attributed to good water chemistry control and maintenance practices performed by the plant operator, Duquesne Light Company, during the 25 years of commercial nuclear power plant operation.

The actual SSDP total man-rem exposure during the Decommissioning Project (September 1984 through September 1989) was 155 man-rem which was less than 20% of the Decommissioning Plan estimate of 1,007 man-rem. This achievement was the result of an aggressive ALARA program which included good planning, management involvement, and worker endorsement. Figure 3 presents the cumulative personnel radiation exposure received during the Decommissioning Project.

THE ALARA PROGRAM

At Shippingport, management's policy was to maintain personnel radiation exposures at the lowest possible levels. To accomplish this, an aggressive ALARA program was implemented. Initially, management established a challenge goal to keep personnel radiation exposures two times lower than the Decommissioning Plan estimate of 1,007 man-rem. (See Figure 3.) The main elements of the ALARA program were:

- Positive management leadership and involvement at all levels
- Detailed ALARA planning and integration into all work procedures
- Effective ALARA training
- Strict procedure compliance
- Close monitoring of work practices and accumulated personnel exposures
- Program endorsement by the workers, management and supervision

ALARA was emphasized throughout the entire decommission process. It was considered during the initial development of the Decommissioning Plan. Good ALARA work techniques were specified during the preparation and review of detailed work procedures. ALARA practices were monitored during work performance by the Radiation Control Technicians, supervision, and management evaluation teams. At the completion of work activities, post-work critiques were conducted with the workers to identify improvements for future work activities. Frequent, formal reviews were performed of actual personnel radiation exposure data to measure the ALARA Program effectiveness.

The DOC ALARA Program requirements were applied to all subcontractors working in a radiation area. After contract award, but prior to the start of physical work, each subcontractor was required to develop detailed work procedures which included personnel exposure estimates. These procedures and man-rem estimates were reviewed and approved by the DOC. If necessary, the subcontractor revised the procedures prior to receipt of DOC approval if significant reductions in personnel radiation exposure could be realized.

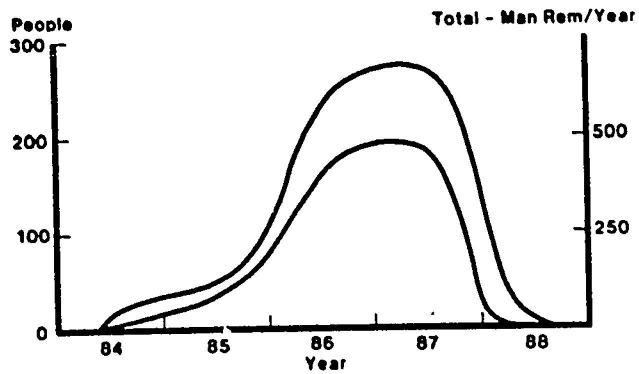

Figure 1. Manpower Loading and Exposure

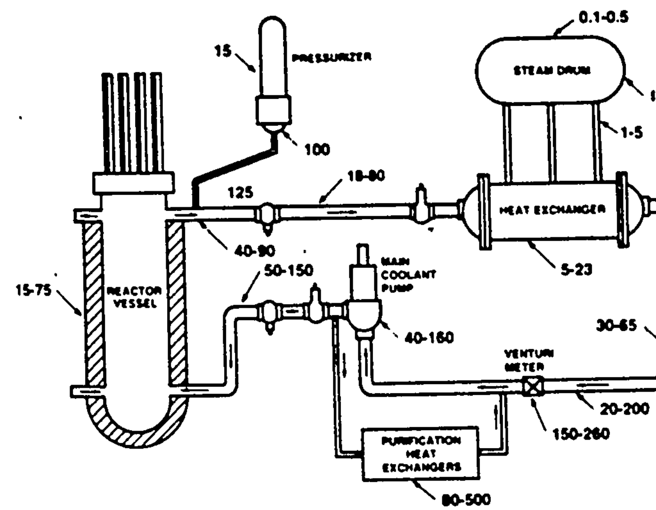

Figure 2. Shippingport Plant Primary System Radiation Survey - MR/HR

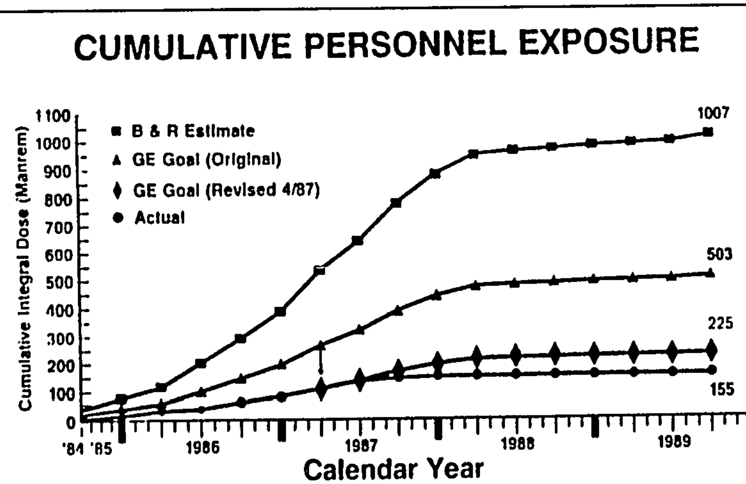

Figure 3. Shippingport Station Decommissioning Project

Detailed work procedures involving significant personnel radiation exposure were submitted to the DOC's Management Safety Committee (MSC) chaired by the DOC Project Manager. During these reviews man-rem estimates frequently were challenged as being too high. These procedures and estimates were revised based on technical guidance received from the MSC. When the procedure and new man-rem estimate was found acceptable, they were approved for use. During the performance of work, changing conditions were promptly addressed. If required, procedures were revised before work could continue. Frequent use was made of classroom and mockup training. The training included a review of the detailed work procedure followed by use of the procedure during mock-up training. Mock-up training helped ensure worker familiarity with the procedure, and the incorporation of good ALARA work practices.

SOME EXAMPLES OF GOOD ALARA PRACTICES

Removal of Radioactive Piping System

The original Decommissioning Plan required the removal of all radioactive piping systems in total containment. The practice of using total containment was simplified by the Decommissioning Operations Contractor's use of a vacuum system equipped with a HEPA filter. The first cut into the radioactive piping was made in total containment. All future cuts were made with the piping system under a negative pressure and exhausted through a HEPA filter. Therefore, containment was not required for subsequent cuts and personnel exposure for the installation and removal of local contaminates was eliminated. A further improvement of this technique was the use of a specially designed saddle valve. The saddle valve was used to drill a hole into a radioactive piping system prior to the first cut. This valve permitted the draining of any residual water left in the piping and provided an adapter so that a vacuum could be applied to the piping internal diameter. Therefore, the first piping cut could be made without containment.

Installation of the Reactor Pressure Vessel Head

When the DOC accepted responsibility for the Shippingport Atomic Power Station from Duquesne Light Company, the radioactive contaminated reactor pressure vessel head was in its storage pit. The reactor pressure vessel was left in a defueled condition, with the reactor internals in place and the pressure vessel full of water. In order to prepare the pressure vessel for removal from its containment chamber, it was necessary to re-install the pressure vessel head and bolt it to the pressure vessel. This operation was estimated to take 12 man-rem of exposure based on performance of this operation in the past.

The detailed procedure and man-rem estimate initially developed for installation of the reactor pressure vessel head was reviewed by the DOC Management Safety Committee. A challenge ALARA goal of 2 man-rem was established for the operation.

In the past, the pressure vessel head which was contaminated with loose surface radioactivity, was installed in total containment. The installation procedure was modified to "fix" the loose surface contamination with paint using a remotely operated, commercially available paint spray gun. The procedure, equipment, and personnel used to perform the painting and installation were trained on a full-size wood mock-up of the pressure vessel head. Once all training was completed, the pressure vessel head was installed successfully. The resulting exposure for the operation was less than the 2 man-rem goal.

Management Involvement

A key element for a successful ALARA program is the leadership and direction provided by management. Management and supervisory personnel's continued involvement in the program is essential to obtain the full endorsement of the workers.

Prior to the start of physical decommissioning, the DOC Project Manager established an ALARA challenge goal of 503 man-rem for the total decommissioning effort. This was one-half the 1,007 man-rem estimate established in the Decommissioning Plan. The new challenge goal was plotted versus time based on the sequence of work activities defined in the Decommissioning Plan. Through March 1987, total actual personnel exposure was maintained below the new challenge goal. In April 1987, a new challenge goal of 225 man-rem was established. The Shippingport Station Decommissioning Project was completed with a total man-rem exposure of 155. The curves of the original 1,007 man-rem estimate, the two challenge goals of 503 and 225 man-rem, and the actual man-rem exposures are shown in Figure 3.

Personnel radiation exposure was controlled at levels far below the SSDP Decommissioning Plan's original estimates. This is attributed to several elements: positive involvement of management, endorsement of the ALARA program by the workers, and the DOC's initial generation of strict radiation control standards and procedures. These elements when combined with innovative work practices will help keep personnel radiation exposures As Low As Reasonably Achievable.

KEY LESSONS LEARNED

Key lessons learned in the application of ALARA at the Shippingport Station Decommissioning Project were:

- Establish "stretch" ALARA goals
- Obtain the full endorsement and support of the ALARA Program by management, supervision and the workers.
- Implement an effective personnel training program
- Monitor compliance of work procedures and work practices
- Challenge the "status quo" of procedures and estimates for both new and repetitive operations.

CONCLUSION

The Shippingport Station Decommissioning Project was completed ahead of schedule and under budget. The Decommissioning Plan estimated 1,007 man-rem exposure. The actual man-rem exposure was 155. The decommissioning was completed without any significant radiological impact on the workers, the public, or the environment. This successful record was the result of an aggressive ALARA Program.

Author Biography

Frank P. Crimi has over 36 years of experience in the nuclear industry. He received his B.S. degree in Mechanical Engineering from Ohio University and joined the General Electric Company (GE) in 1955. He spent the first twenty-five years of his career at the Knolls Atomic Power Laboratory (KAPL) where he held management assignments in nuclear equipment and plant design, plant operations and maintenance, and facility decontamination and decommissioning. Mr. Crimi was responsible for developing and implementing the master plan for decommissioning KAPL's surplus nuclear facilities. Decommissioning projects included low level and high level radioactive waste storage facilities, reactor test facilities, and fuel fabrication shops. In 1981, he transferred to the General Electric Nuclear Energy Division. He was GE's Project Manager for the decommissioning of the Shippingport Atomic Power Station, the nation's first large-scale commercial nuclear power station. In February 1992, Mr. Crimi joined the Lockheed Corporation and he is currently President of the Lockheed Environmental Company. Mr. Crimi is a member of ASME and ANS. He is Chairman of the Long Island Power Authority's Shoreham Decommissioning Independent Review Panel, and a member of the Public Service of Colorado's Fort Saint Vrain Decommissioning Oversight Committee. He is the author of numerous papers on nuclear facility decommissioning.

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PAPER 9-1
DISCUSSION

Granados: I have three questions. One, were the chambers removed from the enclosures, or were they left in place?

Crimi: The chambers were all removed. We salvaged about 1,200 tons of steel. In order to that, we had to vacuum blast the paint, about the lower one-third, which was contaminated over the twenty-five years of service. Everything was removed. Nothing was left in the way of steel. The chambers and turn were inside massive concrete enclosures. Where there was contamination on the enclosures, those were also scabbled.

Granados: The second question is, over in the radioactive waste processing area, there were a lot of underground tanks. Did you find any leakage from them when you pulled those out?

Crimi: No. The plant was very conservatively designed. The tanks were also in concrete vaults which had sumps in them. There were small amounts of radioactivity, but no evidence of tank leakage. There were about 140 tanks that were removed from that project.

Granados: The last question is, over the years a lot of pipes were cut off and abandoned in place and backfilled with dirt. How did you go about finding all of those?

Crimi: The pipes that you are referring to were basically in pipe trenches, which were underground. We had to find where all the piping was located and remove it. As we excavated the ground we would find some electrical cables left over from construction, some pipes, and what not. Everything was done with radiation controls in place until we determined that the material was clean.

Mayfield: I'm not familiar with the term "hydrolasing." Could you describe this process for me?

Crimi: Basically it is a high-pressure water jet, like high-pressure cutting of steel, except done at a much lower pressure.

DECOMMISSIONING ALARA PROGRAMS CINTICHEM DECOMMISSIONING EXPERIENCE

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ABSTRACT

The Cintichem facility, originally the Union Carbide Nuclear Company (UCNC) Research Center, consisted primarily of a 5MW pool type reactor linked via a four-foot-wide by twelve-foot-deep water-filled canal to a bank of five adjacent hot cells.

Shortly after going into operations in the early 1960s, the facility's operations expanded to provide various reactor-based products and services to a multitude of research, production, medical, and education groups.

From 1968 through 1972, the facility developed a process of separating isotopes from mixed fission products generated by irradiating enriched Uranium target capsules. By the late 1970s, 20 to 30 capsules were being processed weekly, with about 200,000 curies being produced per week. Several isotopes such as Mo^{99}, I^{131}, and XE^{133} were being extracted for medical use.

As an expected consequence of this production, the hot cells became contaminated with mixed fission products. In addition to activation products formed in the reactor core structure and biological shield, mixed fission products were also generated in the reactor primary cooling system due to cross contamination from the hot cells and tramp Uranium in the coolant system.

Early in 1990, a decision was made to decommission the facility for unrestricted release after various underground radioactive leaks were discovered in the reactor hold-up tank, transfer canal, and hot cell exhaust system.

Cintichem has been actively decontaminating and dismantling the facility since early 1992, after 1½ years of preparatory work. As of November 1993, about 75% of the physical decommissioning work had been completed, with over 99.9% of the radioactivity (about 4,200 curies) removed. This has been accomplished with about 190,000 person-hours of hands-on management labor, and 138 person-rem of radiation exposure. At completion of the decommissioning project it is expected that 140,000 cubic feet of radioactive waste will have been generated.

Approximately 50% of the incurred exposure can be attributed to five of the thirty-eight major decommissioning tasks, (i.e., preliminary decontamination of the hot cells and underground exhaust removal) (72 person-rem), and reactor core structure dismantling and bioshield decontamination activated concrete removal (7 person-rem). At the onset of this work, radiation dose rates in excess of 1,500 rem/hr (gamma) and 10,000 rad/hr (beta) were encountered in the hot cells along with surface contamination levels in excess of 2E9 DPM/100 cm^2 of aged mixed fission products. During the reactor core structure/activated bioshield work, exposure rates up to 1000R/hr were encountered.

Significant dose reductions were accomplished with a new ALARA management program that was set-up for the decommissioning project. ALARA techniques included remote robotic demolition of activated concrete, remote high-pressure washing of the hot cells, and extensive use of shielding in the work areas.

INTRODUCTION

Cintichem received approval to commence full-scale decommissioning of its reactor and hot laboratory facilities in February 1992. Since that time, the nuclear fuel, reactor core structure, activated bioshield concrete, and a majority of piping and components have been dismantled and removed from the reactor facility. In the hot laboratory facilities, the interior of the 5 hot cells have been decontaminated to less than 1 mrem/hour and demolished, the hot cell underground exhaust duct filtration system removed, and most of the transfer canal structure has been removed as well as most of the surrounding contaminated soil. Currently, over 93% of the radioactivity has been removed from the facility, with about 80% of the overall physical decommissioning work completed. Due to the aggressive nature of the Radiation Protection and ALARA programs, the occupational radiation exposure to date has only been 38% of the original estimate in the Decommissioning Plan, while the physical decommissioning workscope has expanded an additional 60%.

PROJECT BACKGROUND

Description of Cintichem Reactor and Hot Laboratory Facility

The Cintichem, Inc., nuclear reactor facility is located within the Town of Tuxedo, in Orange County, New York. The plant site consists of 100 acres of land, owned by Cintichem. It is in an industrial park area known as Sterling Forest, and is about 3-1/4 miles northwest of the village of Tuxedo Park.

There are six principal buildings at the plant site, with only the Reactor Building, the Hot Laboratory Building and the Class A Low Level Radioactive Waste Storage Building included in this decommissioning.

Reactor Building

The reactor building is a 70 x 92 x 57-ft-high reinforced-concrete structure set into an excavation in the side of the adjacent rock mountain. The exposed portions of the walls and roof are reinforced concrete with a minimum thickness of 12 in. and 8 in., respectively. The volume of the reactor building is about 285,000 ft^3.

The nuclear reactor is a pool-type research reactor, and was licensed to operate at thermal power levels up to 5 MW. The reactor is a light-water-moderated, cooled, shielded, and reflected, solid-fuel reactor.

The reactor had a number of experimental facilities including six beam tubes, a thermal column and pneumatic rabbit tubes. The reactor core was suspended by an aluminum tower from a movable bridge, and was immersed in a 49 x 23 x 32-ft-high pool. The pool was divided into two sections, the narrower stall section contained the fixed experimental facilities and the open end of the pool provided storage space for irradiated fuel and experiments. A 4' wide x 12' deep x 108' long canal connected the open pool with the hot cells to permit the transfer of irradiated material between the two facilities.

A hold-up tank (HUT) (30 x 15 x 10 ft) was designed to provide a delay of the pool water in the primary system during normal operation to allow time for decay of short-lived radioisotopes in the coolant before the water entered the pump room. The HUT was an underground concrete enclosure adjacent to the pump room and buried under 30 feet of soil

Hot Laboratory Building

The Hot Laboratory is a concrete structure 225 feet long by 57 feet wide by 37 feet high. It contained five hot cells, each having 4-foot-thick walls of high-density concrete. The internal dimensions of cell 1 were 16 feet wide by 10 feet long by 15 feet in height. This cell was equipped with a Remote Handling Arm, one pair of heavy duty manipulators, and one pair of standard duty manipulators. Internally, cells 2, 3, and 4 were 6 feet wide by 10 feet long by 12.5 feet in height. Cell 5 is 6 feet wide by 10 feet long by 25 feet in height. Cells 2, 3, 4 and 5 were each equipped with a 4-foot thick lead glass shielding window and all cells are equipped with one pair of Master Slave Manipulators. Major access to all the cells was possible through rear doors (7 feet wide by 6 feet high by 4 feet thick). Access to all cells was also possible via roof openings containing removable plugs.

A canal containing water connected Cell 1 with the reactor pool. Radioactive samples, specimens, and isotopes, etc., were transferred through this canal and brought into Cell 1 via a motorized elevator. This canal also contained a wider area known as the gamma pit, with a large CO-60 source for gamma irradiation experiments.

The waste pit area is located at the north end of the hot laboratory building. It consists of 100 shielded individual radioactive waste storage cells. Each cell is 7-1/4 feet deep by 2-1/2 feet in diameter. The cells are arranged in a honeycomb fashion with additional shielding around the outer perimeter. Each cell has a removable shield plug and is internally vented to the hot cell exhaust ventilation filter system.

Radioactive Waste Storage Building

The Radioactive Waste Storage Building is a single-story concrete block building located 500 feet north of Building 2. It is approximately 24 feet x 60 feet and is utilized for storage of Class A waste, mostly 55-gallon drums.

Site History

The Cintichem facility, originally called the Union Carbide Nuclear Company (UCNC) Research Center, was originally designed and constructed to meet the joint needs of the Union Carbide Nuclear and Ore Companies. Planning and Construction of the Cintichem Sterling Forest facility began in 1957. The 5 MW reactor had its initial criticality in 1961.

Total megawatt hours of usage from initial startup to final shutdown on February 9, 1990 is approximately 906,000 MW hours.

During the operating history of the Cintichem reactor, the following operational occurrences took place which would have an impact on decommissioning safety from a radiological standpoint:

> From 1968 through 1972, the facility developed a process of separating isotopes from mixed fission products generated by irradiating 93% enriched Uranium target capsules. By the late 1970s, 20 to 30 capsules were being processed weekly and several isotopes such as Mo-99, I-131, and Xe-133 were being extracted for medical use. Approximately 200,000 curies of mixed fission products were being produced weekly.
>
> In the mid 1970s, two events occurred which created a significant level of Ag-108/110m contamination throughout the reactor primary water system. During this period, the B_4C reactor control rods were replaced with AgInCd control rods to eliminate the potential for a stuck rod accident. Activated silver from these new rods began leaching into the pool system.
>
> Late in 1989, an underground leak was discovered in the main hot cell bank underground ventilation system. Since this air contained contaminants from the hot cells, mixed fission products were

discovered in the soil and ground water. Bedrock has also been affected in the vicinity of this underground duct work.

Early in 1990, it was discovered that primary water leaks existed in the pool systems' hold up tank, canal, and gamma pit facility. Leaks had been discovered and repaired a few years earlier in the hold up tank. As a result of these leaks, radioactive contamination was found to exist in the vicinity of the drainage system below the reactor building, areas immediately outside of the hold up tank, soil outside the hod up tank and pump room, soil outside of the canal and gamma pit structure, the hot lab building footings, and the north wall area of the reactor building.

Following the discovery of these leaks, operation of the facility was shut down and in May 1990 the decision to decommission the facility for unrestricted release was made. In June 1990, TLG Services, Inc. (TLG), a decommissioning services company, was contracted by Cintichem to co-manage the decommissioning project. At that time, Cintichem and TLG began a site-wide radiological characterization program which lasted approximately three months. Concurrently, conceptual planning, engineering, and cost estimates were prepared. In October 1990, the decommissioning plan was submitted to the USNRC and New York State Department of Labor (NYSDOL) requesting approval to start decommissioning. In November 1991 and January 1992, Cintichem received permission to start active decommissioning work from USNRC's NR and NMSS divisions, respectively.

DECOMMISSIONING PROJECT SYNOPSIS

The decommissioning option selected will require the removal of radioactive material to levels specified by the NRC and NYSDOL for free release. After removal of radioactivity, the buildings will be razed and the site backfilled. The currently projected cost for completion of the decommissioning is $72 million. As of February 1994, about 75,000 cubic feet of radioactive waste has been generated, with an estimated additional 52,000 cubic feet yet to be generated for project completion. The schedule for completion of the physical decommissioning work (i.e., to start final surveys) is January 1995.

Cintichem has been removing radioactivity within the reactor, hot lab building, Class A waste building associated contaminated systems exterior to these buildings, and areas immediately adjacent to them. The major decommissioning workscope to date has included the following:

Removal of activated core structure, associated components and activated portions of the biological shield;

Removal of contaminated equipment, components and fixtures;

Decontamination of building structural surfaces;

Removal of contaminated soil under and adjacent to the buildings; and

Removal of contaminated structures and equipment adjacent to and associated with the reactor and hot laboratory buildings.

RADIOLOGICAL CONDITIONS

A detailed site radiological characterization program, including activation analysis, was performed for the building and the immediately surrounding environs. The purpose of this characterization program centered on the need to obtain specific radiological data concerning areas of the plant that may have become internally or externally contaminated or activated during the reactor and hot lab operating history. This data was necessary for detailed decommissioning planning purposes, and determination of effective and appropriate decontamination and dismantling techniques. This data was also needed for planning radioactive material disposal, assessing potential hazards during decommissioning and determining ALARA controls. This characterization effort was started in June of 1990 and was completed in the fall of 1990.

Two principal sources of radioactive materials were present at the facility, activation products in the reactor building and fission products in the hot laboratory building, with a blend of these two sources found in the transfer canal area. At the time of radiological characterization in 1990, contamination was found to exist in one of two mixtures. In the hot lab the principle long-lived radionuclides consisted of Cs-137, Sr-90 and Ce-144. In the reactor facility the principle long-lived radionuclides consisted of Ag-110^{m}, Co-60, Eu-152/154/155, Cs-134 and Cs-137.

Based upon the initial radiological characterization data, and radiological surveys conducted during D&D work, items listed in Table 1 were found to be the major sources of radioactive material that would cause the majority of radiation exposure during the decommissioning process. This tables lists the principle contaminated structures - equipment, activated components, and subsurface contamination areas (soil, concrete fill and bedrock), and summarizes the radiological characteristics found.

Approximately 4,200 curies of radioactivity was determined to be present. The bulk of this radioactivity, about 3,640 curies, was found in the activated reactor core structure with the remaining 560 curies distributed within the hot cells, equipment, structural surfaces and soil.

EXPOSURE HISTORY EXPERIENCE

Initial Exposure Estimate

The initial radiation exposure estimate for performing the decommissioning was 366 person-rem. This exposure was estimated to be incurred with 75,545 person-hours of exposure time. The bulk of the exposure (286 person-rem, 78%) was expected to be due to 7 of the 38 D&D tasks (39% of the potential exposure hours). These tasks are more directly dealing with the hot cells and reactor core structures. Table 2 provides a breakdown of estimated exposure and exposure hours by major D&D task.

Actual Exposure

As of January 1, 1994, 138 person-rem have been incurred for the decommissioning project with 99.6% of the radioactivity being removed. This exposure was incurred with an expenditure of about 190,000 person-hours where radiation exposure was received. Future exposure needed to complete the project will be negligible (6 to 12 person-rem for 1994) with 99.6% of the radioactivity already removed, the project will be completed with only 38% of exposure estimated to be received, but with 250% of the estimated exposure manpower. If one were to normalize the estimated and incurred radiation exposure by manpower actually incurred, then only 17% of the potential exposure was incurred. While this figure may or may not be entirely valid (because much of the overscope work tends to be performed at the tail end of a task where exposure rates have been reduced early on in the task), it does show that dramatic exposure reductions were realized on this project. This is especially significant in light of the fact that radiological conditions found when areas were "opened up" tended to be radiologically worse and much more extensive.

Table 3 presents a summary of the total decommissioning project as of January 1, 1994. As can be seen, the total project exposure is 137.8 person-rem, with the "average worker" receiving 0.27 rem per year. Worthy of

TABLE 1 - RADIOLOGICAL STATUS

STRUCTURES

Reactor Pool Surfaces - up to 620,000 dpm/100cm^2; up to 1 Rem/hr hot spots.

Transfer Canal Surfaces - up to 1,600,000 dpm/100cm^2 (hot spots); 2 mRem/yr.

Primary Water Hold Up Tank (HUT) - up to 500,000 dpm/100cm^2; up to 100 mRem/hr.

Interior Hot Cells - up to 2,000,000,000 dpm/100cm^2; up to 1,500 Rem/hr Gamma; up to 10,000 Rad/hr Beta.

Interior Hot Cell Filter Bank - up to 1,000,000 dpm/100 cm^2; up to 2 Rem/hr Gamma; up to 50 Rad/hr Beta.

Waste Water Tank Vault (T-1 Room) - up to 1,000,000 dpm/100cm^2; up to 1 Rem/hr.

SYSTEMS

Primary Reactor Cooling System - up to 810,000 dpm/100cm^2; up to 400 mRem/hr.

Waste Water Collection Tank Evaporator - up to 2 Rem/hr; 15 curies Cs-137/Sr-90 sludge.

Buried Hot cell Exhaust System (18"-36' dia. ceramic duct) - up to 10 Rem/hr Gamma; up to 100 Rem/hr Beta.

ACTIVATED COMPONENTS

Rector Core Grid Plate - 800 Rem/hr

Reactor Core Support Tower - 30 Rem/hr

Cooling Water Plenum - 34 Rem/hr

Core Outlet Assembly - 4 Rem/hr

Thermal Column Gamma Shield - 0.8 Rem/hr

Thermal Column Graphite - BKG to 2 Rem/hr

Beam Tubes - 7 Rem/hr (internal)

Activated Concrete Bioshield - up to 3 Rem/hr

TABLE 1 - RADIOLOGICAL STATUS
continued

SOIL/BEDROCK

Exterior soil to Reactor Holdup Tank and Pump Room - up to 10,000 pCi/gm; Co-60, Ag 110m, Eu 152/154/155.

Soil Surrounding Buried Hot Cell Exhaust Duct.

Filter Bank Room and T-1 Room - up to 500,000 pCi/gm; Cs-137, Sr-90, Ce-144.

Soil/Bedrock Under Hot Cells - up to 10,000 pCi/gm; Cs-137, Sr-90, Ce-144.

Soil/Bedrock Exterior to Canal/Gamma Pit - up to 1,000 pCi/gm; Cs-137, Co-60, Ag 110m.

TABLE 2

CINTICHEM DECOMMISSIONING PROJECT EXPOSURE ESTIMATE

TASK	PERSON-REM	EXPOSURE PERSON-HOURS
Remove Reactor Core Structure	32.0	6,400
Remove Activated Thermal Columns	111.0	802
Remove Rx Bldg. Piping and Systems	8.4	2,800
Remove Activated Concrete Bioshield	49.9	998
Remove Rx Beam Tubes	8.6	863
Remove Reactor Coolant Piping	8.5	2,833
Remove Activated Therm Col Liner	6.3	126
Remove Embedded Pipe Bioshield	3.2	650
Dismantle Rx Pump Room	3.6	1,790
Decontaminate Rx Pool	4.7	392
Remove Canal Structure	15.7	3,140
Decontaminate Hot Cells	22.3	11,150
Remove Underground H.C. Exhaust	34.1	3,410
Remove U.G. H.C. Exhaust Filter Rm	20.8	2,080
Other 24 Tasks	36.9	38,111
TOTALS	**366**	**75,545**

TABLE 3

PROJECT PERSON-REM SUMMARY

YEAR	PERSON REM	NUMBER OF EMPLOYEES	AVG REM/ PERSON-YEAR
1991	15.8	98	0.161
1992	72.2	178	0.406
1193	49.8	233	0.214
TOTAL	**137.8**	**509**	**0.270**

TABLE 4

PROJECT PERSON-REM BY WORK GROUP 1991-1993

WORK GROUP	TOTAL PERSON REM	AVG PERSONNEL	AVG REM/PERSON
D&D LABOR	88.60	67	1.320
WASTE MANAGEMENT	21.60	17	1.270
HEALTH PHYSICS/SAFETY	26.10	53	0.491
ENGINEERING/MANAGEMENT	0.95	8	0.118
MAINTENANCE	0.57	25	0..023
TOTAL	**137.80**	**170**	**0.810**

note is the exposure for 1992, where 52% of the project exposures occurred in just one of the three projects first years, This is due to the fact that the NRC part 50 and 70 decommissioning orders were issued in late 1991 and early 1992, respectively. Prior to 1992, only preliminary work involving preparations for actual decommissioning was performed. This work was limited to setup of new site services, equipment, etc., to support the upcoming D&D effort and non-destructive cleanup of the site. Therefore, exposures during 1991 were limited in nature. In 1992, actual decommissioning work started with the two highest level sources of radioactivity, removal of the reactor core structure and preliminary decontamination of the five hot cells. Both of these tasks involved handling radioactive materials with dose rates in excess of 1,000 rem per hour, hence the higher average exposure decreased by about 31% from that of 1992 due to the decreasing inventory of radioactive material on-site. Average exposures in 1993 decreased by about 47% for the same reasons, and also due to a 31% increase in the number of employees.

Table 4 presents the distribution of exposures by work group for 1991 through the end of 1993. The D&D labor group had the highest total exposure and the highest average individual exposures with 88.6 person-rem and 1.32 rem per person over the three year period. This is understandable since this is the work group that actually is decontaminating and dismantling the facility. The next highest exposed work group, on an individual basis, is Waste Management at 1.27 rem per person. This is the work group whose primary function is to treat, package and ship radioactive waste that is generated by the D&D labor group. The Health, Safety, Environmental Affairs (HSEA) department, of which it is primarily the health physics personnel that receive radiation exposure, had a total of 26.1 person-rem, with an average exposure of 0.491 rem per person. However, the HSEA group personnel are diverse in function, ranging from work area crew coverage to environmental monitoring off-site. Therefore, the overall average for this group is misleading. Looking at the D&D HP sub-group, those technicians that cover D&D labor and waste management tasks, the average exposure rises to just slightly less than 1 rem per person. Engineering, Management and Maintenance personnel had only incidental radiation exposures with a total of 1.52 person-rem.

ALARA EXPERIENCE

Radiation exposures by D&D work task for the D&D labor group are presented in Table 5. As can be seen, many of the tasks were completed with much less exposure than anticipated, but with more exposure time on the task. A comparison of estimated exposures and exposure time is given in Table 6. The following presents some of notable experiences from the ALARA standpoint for these tasks.

Removal of Reactor Core Structure

Radiation exposures on this task were reduced to 13% of the initially estimated exposure. This was accomplished through increased use of remote underwater cutting techniques. A remote-operated hydraulically driven underwater circular saw was used to segment the activated core components underwater. This allowed workers to segment components that had exposure rates approaching 1,000 R/hr while incurring exposure rates of only a few millirem per hour due to the use of 15 feet of water shielding and distance. The crew assigned to this work practiced using the remote-operated saw and other remote tools (drills, taps, tongs, eyebolt holders) for about a month using a mocked-up "reactor core."

Removal of Activated Thermal Column Liner

The reactor end of the thermal column steel liner, and the one-inch-thick 5 ft by 5 ft "window" that sealed-off the dry interior of the column from the water filled pool presented a significant ALARA challenge. This work had to be performed with the pool water level below the bottom of the window, thereby removing the water shielding from the 80 R/hr window and nearby activated concrete bioshield. Many surprises were found in the design and construction of this structure that was not shown nor documented from the construction

TABLE 5

ACTUAL D&D LABOR PROJECT EXPOSURES BY D&D TASK[a]

TASK	PERSON-REM	EXPOSURE-HOURS
REMOVE REACTOR CORE STRUCTURE	4.1	8,281
REMOVE ACTIVATED THERMAL COLUMN	1.2	590
REMOVE RX BLDG PIPING & SYSTEMS	2.3	1,219
REMOVE ACTIVATED CONCRETE BIOSHIELD	2.9	3,383
REMOVE RX BEAM TUBES	0.84	1,464
REMOVE REACTOR COOLANT PIPING[b]	0.14	466
REMOVE ACTIVATED THERM COL. LINER	0.83	1,768
REMOVE EMBEDDED PIPE BIOSHIELD	0.33	2,181
DISMANTLE RX PUMP ROOM	6.1	11,271
DECONTAMINATE RX POOL[b]	0.92	6,986
REMOVE CANAL STRUCTURE	7.7	12,526
DECONTAMINATE HOT CELLS	63.8	61,736
REMOVE UNDERGROUND H.C. EXHAUST	8.0	12,372
REMOVE V.G. H.C. EXHAUST FILTER RM[b]	2.6	5,428
OTHER D&D TASKS[b]	5.8	34,814
TOTAL	**107.6**	164,485

[a]excludes routine or non-task related activities
[b]still in progress as of 1/94

TABLE 6

COMPARISON OF D&D LABOR ESTIMATED AND ACTUAL EXPOSURES

TASK	-------------% OF ESTIMATE----------- PERSON REM	EXPOSURE HOURS
REMOVE REACTOR CORE STRUCTURE	13	130
REMOVE ACTIVATED THERMAL COLUMN	1	73
REMOVE RX BLDG PIPING & SYSTEMS	27	44
REMOVE ACTIVATED CONCRETE BIOSHIELD	6	340
REMOVE RX BEAM TUBES	10	169
REMOVE REACTOR COOLANT PIPING[b]	2	16
REMOVE ACTIVATED THERM COL. LINER	13	1,400
REMOVE EMBEDDED PIPE BIOSHIELD	10	335
DISMANTLE RX PUMP ROOM	169	630
DECONTAMINATE RX POOL[a]	20	18
REMOVE CANAL STRUCTURE	49	400
DECONTAMINATE HOT CELLS	286	554
REMOVE UNDERGROUND H.C. EXHAUST	23	363
REMOVE V.G. H.C. EXHAUST FILTER RM[a]	13	261
OTHER D&D TASKS[a]	16	91
OVERALL PROJECT	**29**	**218**

[*]still in progress as of 1/94

drawings. Additionally, this structure was modified without documentation during early facility start-up due to water leaks into the column. The aluminum window was found to have double the number of steel bolts (64 instead of 32) holding it onto the thermal column liner flange. The flange had also been modified to provide a better sealing surface and increased rigidity for the windows. This modification added a stainless steel brace directly into the center of the neutron flux, causing radiation exposure rates to be much higher in the area than anticipated from the window and liner alone. As can be seen from Table 6, this caused the labor effort to increase by a factor of 14. However, exposures were still kept to just 13% of the initial exposure estimate.

To remove the window, a one-inch-thick steel plate slightly smaller than the window was suspended in front of the window such that only the bolts on one side of the window would be unshielded at any one time. After the shield was installed, the pool water level was lowered incrementally to limit the number of exposed bolts as they were removed. The bolts were removed using an electric impact gun with a socket on a 20-foot extension rod. This allowed workers to remove the bolts in a radiation field of 100 rem/hr or less.

Once the activated aluminum window and steel bolts were removed, it was expected that the area exposure rate would decrease significantly. However, it did not, due to the presence of the unknown activated stainless steel brace. An exposure rate of up to 250 rem/hr was encountered. A one-foot-thick, 5-foot-square, high-density concrete shield block was placed in front of the thermal column opening to shield the pool area from the steel brace. Half of the graphite blocks in the thermal column had previously been removed and the remaining half (4 foot thick) shielded the brace from the other side. The graphite blocks were then strategically removed and replaced with high density concrete blocks to allow torch cutting of the carbon steel thermal column liner to which the stainless steel brace was attached. After the liner was cut, the concrete shield block was removed from the pool opening and the cut liner-brace assembly was knocked into the pool with a battering-ram. The pool still had about three feet of water in it which provided some shielding. The brace and liner were then segmented using the same saw that was used to remotely segment the core components.

Remove Activated Concrete Bioshield

The "sphere" of activated concrete surrounding the reactor core area was removed incurring just 6% of the estimated exposure. This was done using a remotely operated demolition device known as a "Brokk," which is an electro-hydraulically operated machine similar to a small backhoe. It has a 4-wheeled base upon which is mounted a turrent with an articulated arm capable of operating a clamshell bucket, a digging bucket, a shear or a demolition hammer. This device was operated remotely at a distance up to 35 feet from the higher exposure rate area. This machine demolished the activated concrete, sheared the massive rebar and steel liner and then loaded the waste material into rad-waste containers. Being a complex device, a significant manpower penalty was incurred due to equipment breakdown time, as can be seen from the 340% manpower overrun.

Decontaminate Hot Cells

This work started in early 1991 and was completed in March of 1994. Decontamination of the hot cells proved to be a very difficult task, requiring over five times the initially estimated effort and almost three times the initially estimated exposure.

The initial decontamination steps were aimed at reducing radiation exposure rates inside the cells to allow removal of equipment and manned entry into the cells. This initial work consisted of using the existing manipulators to remove small objects from the cells and to then wipe down accessible surfaces. A high-pressure water spray was then used remotely with the manipulators to wash down the interior of the cells. This initial washing lowered the exposure rates at the cells' roof access plugs to tens of rem/hr instead of the initial thousands of rem/hr. This allowed workers to use long-handled tools to pick up and rig equipment to the overhead crane. Equipment was then lifted to just below the top of the cells where they could be decontaminated with high pressure water to allow disposal as Class A waste.

Once equipment was removed from the cells, the exposure rates at the top access plug areas was 1-2 rem/hr, with exposure rates still in the range of hundreds of rem/hr down inside the cells. The next step was to remotely remove as much contamination as possible from the interior cell surfaces. This again was done using 10,000 psi high pressure water with 20-foot-long water lances. This technique generally removed all the paint from the cells' interior surfaces. The high-pressure water decontamination continued until no longer effective, with exposure rate being lowered to 1 to 10 rem per hour inside the cells.

At this point, manned entry into the cells was required for further decontamination work. The rear shield plugs were opened to allow access. Shielding consisting of 3/4-inch plywood sheets (for beta) and steel and lead sheets (for gamma) were installed over all surfaces. Only small work areas were left unshielded at any one time to keep exposure rates manageable. Workers then cut out installed fixtures and scarified concrete surfaces. Up to 4-5 inches of concrete surface was removed. This lowered exposure rates to below 10 mrem/hr.

At this point it became radiologically and environmentally feasible to core sample below the cells' floors to look for soil and/or bedrock contamination that could have originated from water leakage from the canal and gamma pit. After sampling, it became apparent that the cell structure could not be decontaminated to release levels due to cracks and fissures in the cells' foundation that became contaminated.

The hot cell structure was then demolished using the same remote-operated Brokks machine which was used to demolish the activated bioshield structure. These same "Brokks" are now being used to excavate contaminated soil and bedrock from beneath the hot lab building.

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RADIATION DOSE OPTIMISATION IN THE DECOMMISSIONING PLAN FOR LOVIISA NPP

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ABSTRACT

Finnish rules for nuclear power require a detailed decommissioning plan to be made and kept up to date already during plant operation. The main reasons for this "premature" plan is, firstly, the need to demonstrate the feasibility of decommissioning, and, secondly, to make realistic cost estimates in order to fund money for this future operation. The decommissioning plan for Loviisa Nuclear Power Plant (NPP) (2x445 MW, PWR) was issued in 1987. It must be updated about every five years.

One important aspect of the plan is an estimate of radiation doses to the decommissioning workers. The doses were recently re-estimated because of a need to decrease the total collective dose estimate in the original plan, 23 manSv. In the update, the dose was reduced by one-third. Part of the reduction was due to changes in the protection and procedures, in which ALARA considerations were taken into account, and partly because of re-estimation of the doses.

In the re-estimation emphasis was put on those works that seem to cause the highest dose contributions, because these will in some cases also yield the biggest reductions. The main means for reducing the doses were addition of shielding and changes in work procedures. For instance, work with activated components cause 4.4 manSv (reduced by 3.4 manSv), and with contaminated components 4.0 manSv (reduced by 1.4 manSv). The most significant dose reducing actions were introduction of additional shielding into the RPV during disassembly of the surrounding structures, and adding shielding to the dummy fuel assembly containers and operator positions in addition allowing for increased decay time.

There is still potential for further ALARA-based dose reduction. The problems with this are, however, greater, because of the uncertainty in the actual radiation conditions at the plant at the time of decommissioning, and also because of incomplete knowledge of activity levels. An outline of further optimisation work is discussed, and guidelines for a methodology are given.

INTRODUCTION

There are four NPP units in operation in Finland. The Loviisa NPP consists of two PWR units of net capacity 445 MW. The reactor plant is of Russian VVER-440 design, but a considerable part of the plant differs radically from VVER-440 plants in Eastern Europe. One main basic difference is safety, for which normal Western standards were adopted from the beginning. This is reflected in many features of the plant. For example, the plant has an ice condenser containment based on Westinghouse design. On the other hand, the basic Russian design can be seen e.g. in the primary circuit layout, with six loops and horizontal steam generators, and in a certain complexity of e.g. process systems, with lots of spare capacity, a large number of components, etc. This has certain implications for decommissioning.

The units were commissioned in 1977 and 1980. Operation has been successful with typical load factors around 85-90%. It is expected that problems encountered can be handled in a positive way, and that operation in the future will not differ significantly from what it has been until now. It is also probable that no major changes will take place in the radioactivity and radiation conditions, although the behavior of the two units is quite different in this respect.

The spent fuel has so far been exported to Russia after being stored at the plant. Intermediate and low-level waste is stored at the plant, and will eventually be placed in an underground final repository at the site. The repository is being built at present. The decommissioning waste will be disposed of in the same repository, which will be enlarged at the time of decommissioning.

The general waste handling philosophy in Finland is that waste handling costs are taken into account during the operation of the plants and included in the price of the energy produced. The basic arrangement is defined in Finnish legislation, especially the Nuclear Energy Law, and clarified elsewhere. The decommissioning cost estimates may be more or less conservative, and it is therefore in the interest of the utility to have a detailed, plausible decommissioning plan, in order to be able to estimate and keep up-to-date the costs involved with a small error margin.

The decommissioning is planned to take place at around 2010, after about 30 years of operation. The decision and the final decommissioning plan will be made only when the technical and economical life of the plant is coming to an end. It is of course possible that the lifetime of the plant will be extended, if it proves to be economically feasible.

DECOMMISSIONING PLAN

The first decommission plan for Loviisa /1/ was completed in 1987. It describes in great detail a feasible way of dismantling the plant and of depositing and isolating the radioactive waste into the final repository on site. It is of course realized using technology of today, keeping in mind some probable developments in decommissioning techniques when planning the procedures. It is based on the concept of almost immediate dismantling without utilizing decay, dismantling of main components without cutting into smaller pieces, and leaving the non-active parts of the plant mainly intact on site. The reactor pressure vessel will be removed in one piece, placed on a transport vehicle and transported into the final repository, where it will serve as a container for the reactor internals. These will be transported in a shielding cylinder. Other large components, such as the steam generators, are also removed in one piece. Other activated or contaminated material will be cut into manageable pieces and placed into special shielded containers, depending on their activity.

The total volume of the waste produced is about 13000 m^3.

ORIGINAL DOSE ESTIMATES

The total dose estimate obtained in the original decommissioning plan was 23 manSv. This number is based on addition of a large number of dose components, some of which could be evaluated rather reliably. Other components were not very accurate. Some were even rough estimates or educated guesses. In the first attempt to arrive at a plausible number it was essential to study the distribution of doses, and in this respect the original plan serves its purpose rather well. The absolute value of the total dose was of less interest at this stage, because it was well recognized that the estimate contains large uncertainties both due to approximate definition of the work procedures (compared to what they actually will be), incomplete knowledge of the radioactivity and radiation conditions of some parts of the plant, uncertainties in how the activity levels will develop in the future, rough activation estimates of the main components without much verification based both on limited information on exact material composition and on neutron field distributions, uncertain dose estimates for complicated geometries and so on. In addition, enough attention was not given to individual doses, dose distributions and dose limits, but mainly on the collective doses irrespective of the details of how they are generated.

The Finnish Centre for Radiation and Nuclear Safety (STUK) required in their statement additional studies to be made on the possibilities to decrease the radiation doses. Such a study was initiated in 1991.

LOWERING AND RE-ESTIMATION OF DOSES

The re-evaluation of the doses concentrated both on reconsidering some of the original dose projections, partly because some new information and more sophisticated analytical methods were available, and on studying the possibilities to improve work procedures, protection and other factors, which have an influence on the doses. In a broad sense an optimisation approach was taken to the second part of the problem.

In the following an overview is given of radiation doses and dose reduction efforts for some important stages of decommissioning and types of activities involved.

Preparing for Decommissioning

The main dose-causing activities during the preparation stage are

- unsealing of the reactor, defuelling,

- flushing of process systems associated with primary circuit,

- removal of filters (main coolant pump sealing system, ventilation, off-gas treatment),

- radiation surveys,

- fuel handling,

- decontamination of primary circuit.

Most of these operations are well known, because they are directly based on normal outage operations. No special dose reducing actions have been taken. The main contributions are from radiation surveys (0.21 manSv) and reactor operations (below 0.1 manSv). The survey value is the most difficult one to estimate, and it is possible to apply optimisation to it in planning once the survey program is specified, and in addition when it is carried out.

The preparation stage causes 0.3 manSv, which is a reduction of 1.1 manSv compared to the original estimate. The reduction is due to more careful evaluation.

Activated Components

In the re-evaluation attention is mainly paid to the most important activated components and structures based on the estimate of the original plan that these cause 82% of the activated component doses. These are

- dismantling, temporary storage, transportation and final storage of the reactor pressure vessel (RPV),

- packing, transportation and final storage of the "dummy" fuel assemblies (these are inactive fuel element-like components replacing the outermost layer of fuel assemblies in the core in order to decrease neutron embrittelment of the RPV),

- disassembly, packing, transportation and final storage of the reactor biological shielding.

For these a number of dose reducing actions have been proposed, mainly based on ALARA considerations.

Dismantling of the RPV

During dismantling of the RPV most work is done in the vicinity of the loop nozzles. The main radiation sources are the loops, the activated and contaminated inner surface of the RPV above the water level and the activated RPV material at core level. The following changes in work procedures were proposed in order to save dose:

- the loops are cut at the upper nozzles and shield disks are welded to the nozzles before dismantling of the shielding around the reactor,
- the remaining parts of the loops, e.g. the lower nozzles, could be shielded in an equivalent way depending on the situation and need ,
- temporary shields are used during the welding,
- as much of the work as possible in the nozzle area is done prior to disassembling the shielding around the reactor.

Radiation from the inner surface of the RPV can be decreased significantly by installing a shield consisting of a massive shielding cylinder or possibly separate massive plates into the RPV. A preliminary optimisation analysis for this indicates an optimal thickness of 4 cm, but up to 10 cm could be installed.

Shielding and Transportation of the RPV

The RPV will be partially enclosed into a concrete shield of thickness 300 mm, which decreases the dose rate at the core level to about 1.5 mSv/h. This served as a base case for the optimisation study. The main variant is to embed 6 cm of steel into the concrete at the core elevation to decrease the dose rate locally to 0.3 - 0.4 mSv/h.

During lifting the RPV the only shield is the cylinder. During transportation on a special vehicle, additional shielding can be added to protect the driver. Lifting the RPV into its final storage silo is partly done remotely.

The total dose caused by these operations is about 0.5 manSv. The dose saved by changing the shielding cylinder is 0.12 manSv.

Reactor Internals

Removal and transportation of the reactor internals is in principle a routine operation, because they are normally transported in a steel shield (not submerged under water). The dose expenditure is low, about 0.04 manSv.

Dummy Fuel Assemblies

The dummy assemblies belong to the most severely activated components at the plant. They are transferred from the reactor into a pool. From there they are lifted one at a time into a concrete transport container, using a steel shielding container. The transport container is transferred into the final repository, and the assemblies are again lifted one at a time into the RPV using the steel cask.

To lower the dose optimally a 6 cm increase in the original 25 cm thickness of the steel container wall is necessary. A more important reason is the individual dose limit. In addition, some improvement is achieved by transferring the assemblies in the right order and utilizing the delay between decommissioning of the two units for additional cooling. The Lo1 assemblies can be stored in the Lo2 RPV.

Control Rod Absorbers

Absorber Connection Rods and Other Activated Core Components

Activated parts have been dumped into a temporary shielded storage silo. Dismantling the storage, which consists of separate steel tubes embedded in concrete, can be done in several ways, none of which is easy. The main improvement is based on arranging the contents into the final storage container in such a way that the most heavily activated parts are put in the middle.

Biological Shield

The activation of the biological shield was re-estimated using more sophisticated analytical tools, better knowledge of the composition and a few radiation level values measured inside the shield. The dismantling consists of eight main phases, of which nos. 3, 4 and 6 cause the largest doses. These are

- cutting of the reactor heat shield,
- cutting of the dry serpentinite concrete shield,
- cutting of the structural concrete at around core elevation.

Dismantling of the biological shield is very work intensive, and therefore even quite low dose rates can cause significant doses.

The total re-estimated dose burden is 0.089 manSv from cutting, 0.044 manSv from packing, 0.095 manSv from transportation and 0.095 manSv from the repository. Although the single components changed significantly from the original plan, their sum did not. The total dose is 0.3 manSv. No changes in the original arrangements (performing some operations under water, temporary shielding above the reactor pit, waste containers) were proposed, except pointing out the need to consider the order of packing.

Dose Distribution in Dismantling Activated Material

The dose components caused by dismantling activated components are presented in Table 1.

Table 1. Dose components (manSv) caused by activated material

	Dismantling	Packing	Transport	Repository	Sum
Reactor pressure vessel	0.86	0.10	0.07	0.26	1.29 (-1.12)
Reactor internals	-	0.02	-	0.02	0.04
Dummy fuel assemblies	0.003	0.067	0.02	0.121	0.21 (-0.55)
Control assemblies	-	0.04	0.07	0.05	0.16
Small activated parts	0.01	0.14	0.01	0.02	0.18
Reactor heat shield	included in the biological shield doses				(-0.03)
Biological shield	0.089	0.044	0.095	0.095	0.32 (-0.04)
Sum					2.20 (-1.74)

Contaminated Material

Doses caused by dismantling contaminated equipment are estimated to be of the same order of magnitude as those caused by activated components. Typical for contaminated equipment are the low dose rates in many cases, in combination with a large volume of very slightly contaminated waste. There are also single highly contaminated components, e.g. the loops and the primary coolant purification system heat exchangers. The uncertainties are very large, until a complete survey of the actual situation is done prior to decommissioning. Contamination levels are hard to predict.

The main dose saving improvements to be utilized, are

- use of various remotely controlled equipment,
- flexible use of mobile local shields,
- optimising the order of dismantling according to activity and dose rate levels,
- decontamination of single systems or components (potential method).

Decommissioning Doses

Table 2 shows the projected dose contributions from main types of activities for both units, and the dose savings compared to the original plan.

Table 2. Distribution of total decommissioning doses (manSv)

	Lo1	Lo2	Lo1 & Lo2	
Preparation works	0.3 (-1.1)[a]	0.3 (-1.1)	0.6 (-2.2)	
Decontamination of primary circuit	0.06	0.06	0.12	
Dismantling				
- activated components	2.2 (-1.7)	2.2 (-1.7)	4.4 (-3.4)	
- reactor building contaminated components	2.0 (-0.7)	2.0 (-0.7)	4.0 (-1.4)	
- other contaminated components			2.5 (+0.65)	
IVO staff			2.1 (-0.8)	
Sum				13.7 (-7.2)
		+10%	1.4	
Total				15.1 (-7.9)

[a]Values in parenthesis are changes relative to the original plan

OUTLINE OF FURTHER DOSE OPTIMISATION

The re-evaluation of doses included elements of ALARA-based reconsideration of operations and procedures, and of some specific optimised improvements. Optimisation was not, however, carried out to the extent possible. One reason for this was the lack or incompleteness of some information.

In the updated dose estimate special attention was given to the largest dose components. It was felt that these had the largest potential for decreasing the doses. Although this is not necessarily the whole truth, and in addition actually is not consistent with the ALARA principle, no big effort can be made to optimise small dose components. Instead, general procedures which are efficient in further lowering many already low dose components should be looked for. For instance, the gain from covering the most significant sources during dismantling of large, slightly contaminated process systems should be estimated in a systematic way, although there does not seem to be any apparent reason for this, based on the low dose rates.

Optimisation must be based on realistic dose estimates, which are derived from realistic values for all relevant factors affecting the doses. Therefore, such optimisation should be done at an appropriate stage. There are only limited possibilities for this as long as the necessary information does not exist.

It is still necessary to improve various single dose estimates as part of the optimisation. At the present stage they are based on a variety of information, such as

- gross collective and individual dose experience,
- conclusions drawn by analogy from single dose experiences,
- crude conceptual models for collective dose components.

It is useful to include some explicit measure of uncertainty in the various dose component estimates, in order to see more clearly where there is need for improving the quality of the information.

For optimisation purposes it may be advantageous at some later stage to systematically generate and consider several feasible options for performing single operations. Such options have to some extent been considered in the present re-evaluation, but not very explicitly.

In general more formal ALARA studies should be made, in order to restrain from going too far in dose reduction. Additional dose reductions can be required, but good reasons based on other considerations than pure radiation safety should be given explicitly.

A special data base should be established at an early enough stage, in order to collect information useful in the decommissioning planning, which is obtained as by-products in other activities. It is probably more economical to gather such data during a long period of operation than by brute force in a short time. This would gradually improve the quality of the dose estimates and dose planning.

It has still not been possible to consider in enough detail the role of individual exposures. Inclusion of such considerations will introduce additional motives for improving protection and lowering doses because of the mere expenses caused by a possible need to expose a larger number of persons. Rotating workers in such a way has multiplicative effects on costs, because of the work arrangements, administration, education and training and even employing more people. These considerations may well be treated on a cost-benefit basis.

CONCLUSIONS

It must be noted that the re-estimation of doses is still based on the original assessment. The estimated dose savings and the increase in some components should be seen in relation to this. There are still many open questions regarding the absolute level of the doses, although there is no reason to believe that the actual doses would differ much from the estimated ones, if decommissioning is performed according to the plan.

A quite realistic estimate of the dose savings relative to the original plan is 7.9 manSv, leading to a total collective dose of 15.1 manSv for both units.

The role of operative radiation protection during the actual decommissioning has not been fully utilized in the planning. Depending on the actual conditions and needs, operative radiation protection has available an extensive repertoire of means to decrease doses in single works. By acting reasonably and applying normal procedures of radiation protection and health physics based on ALARA considerations, essential dose savings can probably be achieved.

There is still potential for ALARA-based dose saving efforts. The main obstacle to this at present is that all necessary information is not available. In later re-estimations additional improvements in dose reduction will be possible.

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ALARA AND DECOMMISSIONING - THE FORT ST. VRAIN EXPERIENCE

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ABSTRACT

The Fort St. Vrain Nuclear Generating Station, the first and only commercial High Temperature Gas Cooled Reactor to operate in the United States, completed initial fuel loading in late 1973 and initial startup in early 1974. Due to a series of non-nuclear technical problems, Fort St. Vrain never operated consistently, attaining a lifetime capacity factor of slightly less than 15%. In August of 1989, the decision was made to permanently shut down the plant due to control rod drive and steam generator ring header failures. Public Service Company of Colorado elected to proceed with early dismantlement (DECON) as opposed to SAFSTOR on the bases of perceived societal benefits, rad waste, and exposure considerations, regulatory uncertainties associated with SAFSTOR, and cost. The decommissioning of Fort St. Vrain began in August of 1992, and is scheduled to be completed in early 1996. Decommissioning is being conducted by a team consisting of Westinghouse (engineering and contract support), MK-Ferguson (craft) and Scientific Ecology Group (radiation protection). Public Service Company of Colorado as the licensee provides contract management and oversight of contractor functions.

An aggressive program to maintain project radiation exposures As Low As Reasonably Achievable (ALARA) has been established, with the following program elements: temporary and permanent shielding; contamination control; mockup training; engineering controls; worker awareness; integrated work package reviews; communication; special instrumentation; video camera usage; robotics application; and project committees. To date, worker exposures have been less than project estimates. From the start of the project through February of 1994, total exposure has been 98.666 person-rem, compared to the project estimate of 433 person-rem and goal of 347 person-rem. The presentation will discuss the site characterization efforts, the radiological performance indicator program, and the final site release survey plans.

INTRODUCTION

Background

Decommissioning of commercial nuclear generating stations has been initiated in the United States with the dismantling of the Fort St. Vrain Nuclear Generating Station. Fort St. Vrain is owned and was operated by Public Service Company of Colorado (PSC) and is located on a 2,798 acre site approximately 35 miles Northeast of Denver, Colorado.

Fort St. Vrain is unique in the United States as it is a High Temperature Gas Cooled Reactor. The facility was rated at 842 MWth and 330 MWe. Significant Milestones for the facility are as follows:

Construction initiated in 1968

Construction completed in 1973

Initial nuclear criticality was achieved in 1974

Unit committed to commercial operation in 1979

Stipulation & Settlement Agreement removes facility from rate base 1986

PSC informs NRC that facility will be shutdown not later than June 30, 1990

Decision by PSC to terminate facility operations August 29, 1989

Possession Only License Issued 1991

Decommissioning/dismantlement activities begin August 1992

In response to the historical reduced levels of generation at Fort St. Vrain (FSV), the Colorado Public Utilities Commission instituted penalties against PSC in 1986 which reduced the revenues that could be recovered from its customers. As a result of unfavorable plant operating performance, Fort St. Vrain did not produce adequate revenues to offset expenses during 1987-1989. In August 1989, following plant shutdown due to control rod drive problems, significant cracking was discovered in the steam generator main steam outlet piping assemblies. Due to these problems along with other "mechanical" and "financial" concerns PSC decided not to restart the plant.

DECON Versus SAFSTOR Decision

PSC initially filed a Preliminary Decommissioning Plan based on the SAFSTOR decommissioning option. Following a conceptual dismantlement study that verified that the technology existed to dismantle the reactor vessel, PSC elected to pursue the DECON (early dismantlement) option. Several major considerations were associated with this decision, including:

Future regulatory risks and standards could not be quantified

Low Level Radioactive Waste (LLRW) disposal rates will continue to escalate

Significant personnel knowledge will be lost by waiting the 55 year SAFSTOR time period

No significant reduction in LLRW volume occurs during the 55 year SAFSTOR

Dismantlement provides for a significant reduction in future liability

Projected Radiological Conditions and LLRW Inventories

Activation Analysis and Site Characterization activities yielded project LLRW Classification estimates, project curie estimates, and project dose rate estimates for the Prestressed Concrete Reactor Vessel (PCRV) as presented in Tables 1 through 3.

TABLE 1

DECOMMISSIONING
WASTE CLASSIFICATION PREDICTIONS

CLASS	VOLUME (CUBIC FEET)
A	79,157
B	20,279
C	636
TOTAL	100,072

TABLE 2

ESTIMATED CURIE TOTAL AT FSV
(Three Years After Shutdown)

NOTE: The systems listed below are those systems which are known to be contaminated, or experiencing on-going maintenance, defueling and component removal which may transfer contamination to other systems and/or locations.

System No.	System	Total Curies	
		From Activation	From Loose Contamination (1)
11	PCRV and Internal Components	7.94 E+05	2.54 E+02
12	Controls Rods and Drives	1.84 E+04	N/A
13	Fuel Handling Equipment	N/A	8.95 E-03
14	Fuel Storage Facility	N/A	2.08 E-02
16	Auxiliary Equipment	N/A	9.05 E-03
17	Reactor Removable Reflector	4.82 E+05	N/A
21	Primary Coolant	N/A	6.01 E+01
22	Secondary Coolant	N/A	5.68 E+03
23	Helium Purification	N/A	9.33 E-01
61	Decontamination Systems	N/A	1.06 E-05
62	Radioactive Liquid Waste	N/A	4.06 E-05
63	Radioactive Gas Waste	N/A	8.15 E-05

(1) Includes an estimate of loose surface contamination due to activated corrosion products.

TABLE 3

PCRV DOSE RATES ESTIMATES IN AIR
AT 5 YEARS AFTER SHUTDOWN

RADIAL	GAMMA DOSE RATE R/Hr
All components (from large side reflector to PCRV concrete)	9.7E + 01
Large side reflectors removed (from spacers to PCRV concrete)	2.3E + 02
From core barrel to PCRV concrete	2.1E - 02
PCRV liner and concrete only	8.8E - 03
PCRV concrete only	4.5E - 03
22" PCRV concrete removed	6.3E - 06
24" PCRV concrete removed	3.4E - 06
AXIAL UP	
All components (from Kaowool insulation to PCRV concrete)	1.7E - 01
PCRV liner and concrete only	4.4E - 01
PCRV concrete only	1.7E - 01
32" PCRV concrete removed	7.6E - 06
34" PCRV concrete removed	4.4E - 06
36" PCRV concrete removed	2.6E - 06
AXIAL DOWN	
All Components (from core support blocks to core support floor)	6.1 - 02
PCRV liner and concrete only	2.5E - 01
PCRV concrete only	1.8E - 02
20" PCRV concrete removed	5.3E - 06
22" PCRV concrete removed	2.7E - 06

INITIAL SITE CHARACTERIZATION

The initial characterization of the Fort St. Vrain site began in April, 1991, approximately 16 months prior to the start of decommissioning. The duration of the site characterization program was 9 months, and approximately 20,000 person-hours were utilized. Key elements of the program included background determination, steam system characterization, auxiliary systems characterization, structural and environmental characterization, and development of isotopic scaling factors. An aggressive radiological monitoring program typically translates into a large number of measurements, and the site characterization program was no different - over 25,000 direct measurements were taken. State of the art instrumentation was utilized which provided for the use of a microprocessor linked with a relational database and a bar coding system. The data from the site characterization fills 15 volumes and served to help determine the project scope and remediation alternatives. Were there any surprises? Yes, contamination was found in several systems which had not been previously suspected of being contaminated.

ALARA PROGRAM ELEMENTS

The key elements of the Fort St. Vrain decommissioning ALARA program are as follows:

AGGRESSIVE SHIELDING PROGRAM

- worker involvement
- radiation protection involvement
- engineering analysis on site
- inspection/approval by ALARA group

CONTAMINATION CONTROL

- liberal use of stripcoat
- hydrolazing operations
- use of glove bags/shrouds
- use of drippans, dams and rinsing
- worker awareness

MOCKUP TRAINING PROGRAM

- classroom training held prior to field mockup
- mockups are videotaped
- small models used in classroom
- field mockups require demonstration of proficiency

RADIOLOGICAL ENGINEERING CONTROLS

- plant ventilation (20,000 acfm)
- work platform ventilation (17,000 acfm)
- local or portable ventilation
- box containments with ventilation attached

ALARA AWARENESS PROGRAM

- employee ALARA suggestion program
- ALARA included in general training program
- ALARA newsletter and posting of exposure goals
- ALARA briefings and worker "toolbox" sessions
- reviews of work in progress and exposure investigations

INTEGRATED WORK PACKAGE REVIEWS

- engineering/ALARA personnel work jointly on packages
- RP hold points/field sign offs incorporated
- ALARA must approve all revisions

COMMUNICATIONS

- hand-held radios/headsets
- paging/intercom systems

SPECIAL INSTRUMENTATION

- automated alarming digi-dose system with alarms/visual signals
- remote monitoring devices with alarms and visual readouts
- underwater sampling/monitoring equipment

VIDEO CAMERA USE

- aids in supervisory inspections
- reduces exposures for visitors and plant tours
- allows monitoring of work in progress from low dose areas

PROJECT COMMITTEES

- ALARA and Decommissioning Safety Review Committee (DSRC)
- review high exposure tasks
- review project performance indicators
- discuss project concerns
- include all plant work groups
- field working level personnel are included

RADIOLOGICAL PERFORMANCE INDICATOR PROGRAM

The following areas are included in the program:

PERSON-REM

An exposure estimate was developed for each work task involved in the decommissioning. In addition the project establishes aggressive goals for each work task. The exposure estimates and goals are compared to the actual exposure for the period of interest (month, year, or project to date).

SOLID RADIOACTIVE WASTE

The volume of waste shipped to waste processors or the disposal site are tracked by type of shipment (cask, van, boxes, etc.) for the period of interest. Total volume sent to the disposal site is compared to our contractual limit at the disposal site of 140,000 cubic feet for the entire decommissioning project.

PERSONNEL CONTAMINATION EVENTS

Goals for clothing and skin contaminations are established, and actual performance for the period of interest is compared to the goals. In addition, a measure of contaminations per unit time in radiological control areas is compared to similar data for light-water reactors.

RADIOLOGICALLY CONTROLLED AREA DATA

Time spent in radiologically controlled areas and number of respirators issued are tracked for the period on interest.

ALARA SUGGESTIONS

The number of suggestions received and implemented for the period of interest are tracked.

POSITIVE BIOASSAY RESULTS

The number of positive whole body counts and tritium analyses are evaluated for the period of interest.

RADIOLOGICAL OCCURRENCE REPORTS

The number of radiological occurrence reports are evaluated for the period of interest.

A typical performance indicator report is presented in Table 4.

TABLE 4

SUMMARY OF FEBRUARY PERFORMANCE INDICATORS

PERSON-REM			
	Estimate	Goal	Actual (to date)
Month (Feb.)	6.5	5.2	5.086 (1)
1994 (Jan-Feb)	12.0	9.6	9.249 (1)
Total Project	433	347	98.666(2)

(1) Based on TLD and DIGI Results
(2) Based on TLD and DIGI Results and adjustments for past TLD read corrections

SOLID RADWASTE VOLUME PROCESSED AND SHIPPED	
Thru (Feb.)	6,974 ft^3
Projected (Mar.)	4,000 ft^3
Project Total to Date	76,104 ft^3 *

* As of June 1, total adjusted to include volume of waste sent to SEG for volume reduction

RADIOACTIVE MATERIAL SHIPPING UPDATE (1992/1993 TOTALS)				
1992 TOTALS				
Shipment Type	Number	Weight (lbs)	Volume (ft^3)	Activity (Ci)
Vans	11	NA	6264	7.10
Casks	38	NA	4245	8671.20
SEG Shipments	1	37340	2080	.34
Totals	50	37340	12589	8678.64
1993 TOTALS				
Vans	5	NA	3,727	6.13
Casks	77	NA	12,503	22,405.03
Concrete Boxes	36	NA	22,167	68.34
SEG Shipments	11	404,460	18,144	4.26
Totals	129	404,460	56,541	22,483.76

TABLE 4

SUMMARY OF FEBRUARY PERFORMANCE INDICATORS - Continued

1994 RADWASTE SHIPPING TOTALS				
Vans	0	NA	0	NA
Casks	26	NA	4894	3131.27
Concrete	0	NA	0	0
SEG Shipments	2	35,850	4894	.29
Total Thru 2-28	28	35,850	6974	3131.56

PERSONNEL CONTAMINATION EVENTS		
Period	Clothing	Skin
Month (Feb.)	2	0
1994	2	0
Total Project	36	12

RADIOLOGICALLY CONTROLLED AREA DATA		
Period	RCA Person Hours	Respirators Issued
Month (Feb.)	9,861	85
1994	21,017	96
Total Project	202,796	1,725

ALARA SUGGESTIONS		
Period	Received	Implemented
Month (Feb.)	6	4
1994	9	7
Total Project	157	114

TABLE 4

SUMMARY OF FEBRUARY PERFORMANCE INDICATORS - Continued

POSITIVE BIOASSAY RESULTS		
Period	**Whole Body Counts**	**Tritium Analysis**
Month (Feb.)	0	0
1994	0	0
Total Project	0	0

RADIOLOGICAL OCCURRENCE REPORTS	
Month (Feb.)	8
1994	10
Total Project	60

FINAL SITE RELEASE SURVEY

The final site release survey plan was submitted to the Nuclear Regulatory Commission (NRC) in February of 1994. The NRC has given a preliminary indication that they will be able to perform their initial review in about six weeks time. We hope to begin performing limited final release surveys in the third or fourth quarter of 1994.

CONCLUSION

Fort St. Vrain is one of the very first large scale reactor decommissioning projects to be undertaken in this country. An aggressive multi-faceted ALARA program has contributed to very impressive radiological performance indicators to date. Public Service Company of Colorado and our decommissioning contractor team are looking forward to a successful conclusion to the project in the early 1996 time frame.

Author Biography

Ted Borst, CHP, is the Facility Support Manager for Public Service Company of Colorado at the Fort St. Vrain Nuclear Station. Mr. Borst also serves as the Radiation Protection Manager at Fort St. Vrain. Apart from his radiation protection responsibilities, Mr. Borst is also responsible for the training, emergency preparedness, security, and document control programs at Fort St. Vrain. Before joining Public Service Company of Colorado, Mr. Borst worked for Battelle Pacific Northwest Laboratories as a research scientist, where he worked on an assortment of Department Of Energy and Nuclear Regulatory Commission sponsored projects. Previously Mr. Borst served as a health physicist for the Nuclear Regulatory Commission, performing compliance inspections out of the Region III office. He has B.S. in Physical Sciences and an M.S. in Radiation Biology, both received from Colorado State University. Mr. Borst is Certified by the American Board of Health Physics.

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AN OVERVIEW OF ALARA CONSIDERATIONS DURING YANKEE ATOMIC'S COMPONENT REMOVAL PROJECT

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ABSTRACT

In February 1992, Yankee Atomic Electric Company (YAEC) permanently shutdown Yankee Nuclear Power Station in Rowe, Massachusetts, after thirty-two years of efficient operation. Yankee's plan for decommissioning is to defer dismantlement until a low level radioactive waste (LLRW) disposal facility is available. The plant will be maintained in a safe storage condition until a firm contract for the disposal of LLRW generated during decommissioning can be secured. Limited access to a LLRW disposal facility may occur during the safe storage period. Yankee intends to use these opportunities to remove components and structures. A Component Removal Project (CRP) was initiated in 1993 to take advantage of one of these opportunities. The CRP includes removal of four steam generators, the pressurizer, and segmentation of reactor vessel internals and preparation of LLRW for shipment and disposal at Chem-Nuclear's Barnwell, SC facility. The CRP is projected to be completed by June 1994 at an estimated total worker exposure of less than 160 person-rem.

INTRODUCTION

Yankee Nuclear Power Station (YNPS) ceased power operations on February 26, 1992 after 32 years of safe operation at an average capacity factor of 74%. This paper provides a brief overview of ALARA lessons associated with component removal activities of YNPS decommissioning.

DECOMMISSIONING PLAN

Yankee's Decommissioning Plan was submitted to the NRC in December 1993. This plan presents the programs, processes, and procedures that will be used to fully dismantle the plant before the end of 2002, depending on the availability of a low level radioactive waste site. Given the uncertainties with the DOE high level waste programs, Yankee may build an on-site dry cask storage facility. The schedule for decommissioning assumes that a low level radioactive waste facility will be available to YNPS in 2000 and that greater than Class C waste (GTCC) and spent fuel will be transferred to a dry cask storage facility. Based on these assumptions about low level and high level waste disposal, several milestones are presented in the Decommissioning Plan:[1]

- The facility will remain in a safe storage condition until the year 2000.
- NRC approval of the Decommissioning Plan is expected before January 1, 1995.
- Detailed site radiological characterization of the plant systems, structures, components, soil and groundwater necessary to support dismantlement activities will be initiated in 1994.
- A dry cask spent fuel storage facility will be constructed and loaded with spent fuel sometime after 1996.

- Detailed engineering and planning for plant decontamination and dismantlement activities are scheduled to begin in 1999. Dismantlement begins in 2000 and continues through 2002.

- GTCC waste and spent fuel will be shipped to the DOE beginning in 1998 and ending in 2018.

The total cumulative occupational radiation exposure for the entire decommissioning effort is estimated to be less than 702 person-rem. This value is conservatively estimated based on 1994 dose rates. Dose reduction programs and implementation of decommissioning activities over the next 5 to 10 years will significantly reduce the actual dose received from decommissioning activities. Radioactive waste burial volume for decommissioning and the CRP is estimated to be less than 105,000 cubic feet.

COMPONENT REMOVAL PROJECT DESCRIPTION[2]

The Component Removal Project is the first phase of plant decommissioning. Using the January, 1993 NRC Staff Requirements Memorandum[3] on implementing decommissioning activities prior to decommissioning plan approval, Yankee initiated the CRP. CRP activities were paid for using monies from the decommissioning trust fund. Project staffing and on-site planning began in April 1993, with June 1994 as the target completion date.

Major milestones for the CRP are:

- component asbestos removal,
- shield tank cavity modifications,
- steam generator (S/G) removal,
- S/G preparation and shipment,
- pressurizer removal and shipment, and
- segmentation, packaging and shipment of reactor vessel internals.

The estimated collective dose for CRP is 160 person-rem. In addition, approximately 16,000 cubic feet of low level radioactive waste will be disposed.

Component removal activities are performed under the 10CFR50.59 review process with additional decommissioning-related considerations stipulated by the NRC.[3] The 50.59 process has been used for preparation and documentation of analyses, reports and procedures to implement plant modifications throughout Yankee's operations. Engineering Design Change Requests were developed and approved by the Plant Operations Review Committee for the steam generator and pressurizer removal process and the reactor vessel internals segmentation.

ASBESTOS ABATEMENT

The removal of asbestos insulation at Yankee Rowe to support CRP presented two challenges from the standpoint of exposure to asbestos fibers. The first was containment of the fibers and the second control of radiologically contaminated insulation. An initial concept was to establish an asbestos controlled area for the entire containment and remove all asbestos. However, because of the high potential radiation exposure required to accomplish this task, a decision was made to remove only asbestos necessary to support CRP.

Asbestos abatement activities proved to be the greatest contributor to personnel radiation exposure (53 percent). An estimate of 150 person-rem was developed to erect scaffolding, build enclosures, and abate asbestos. Regulations for asbestos removal and handling required the construction of elaborate enclosures to confine asbestos contamination and prevent personnel asbestos exposure. Based on ALARA considerations, variances were obtained from state regulators for certain asbestos handling and personnel decontamination requirements. Dress-out and bag-out areas were established in low dose areas. Despite these and other changes in work practices, asbestos removal, bag-out and decontamination proved to be time and exposure intensive.

Asbestos abatement was completed with a total exposure of about 73 person-rem. A savings of over half the estimate can be attributed to the learning curve by the radiation workers as the installation of scaffolding and enclosures progressed into successive plant areas. This conclusion is supported by a corresponding decrease in labor hours. For example, labor hours for building the third steam generator scaffolding was 75% of the time required to erect the first.

Another dose saving feature was the relocation of the asbestos decontamination chamber, asbestos packaging area, and bag-out area to the charging floor to reduce time spent in higher dose rates. Table 1 lists the estimated and actual dose received for the asbestos abatement. Approximately 1500 cubic feet of asbestos insulation was removed and disposed as LLRW.

Steam Generators and Pressurizer

Four steam generators and pressurizer have been removed and shipped to the Chem Nuclear Systems Inc. (CNSI) Barnwell facility. These activities were completed by mid-December 1993. These tasks were accomplished almost one month ahead of schedule and under budget. Each of the components required asbestos removal, mechanical closure, and contamination fixation prior to removal from the Containment. Figure 1 shows the average steam generator dose rates with the secondary side filled with water, the secondary side drained and the steam generator shielded for shipment. The estimated dose for removal of the steam generators was about 72 person-rem and the actual dose is about 50 person-rem.

ALARA lessons learned during this phase of CRP are presented below:

<u>Component Removal</u>

1. Maintain water level in components as long a practical.

2. If possible, inject secondary side with grout after draining while S/Gs are still in containment.

3. Fabricated, based on conservative calculations, large area shield plates and stage lower plates on the transport cradle.

4. Use automatic mechanical cutting of reactor coolant piping to reduce exposure by increasing the distance from the source and reducing time in high radiation areas. Mechanical cutting was also effective in confining contamination and minimizing airborne radioactivity.

5. Use an asbestos encapsulant (or radioactive contamination fixative) designed to adhere to carbon steel that is consumable in welding operations to minimize the need for shield weld preparation.

6. Decontaminate and fix smearable radioactive contamination to reduce personnel exposure by eliminating contamination controls and protective clothing during the majority of work after removal from the containment.

TABLE 1

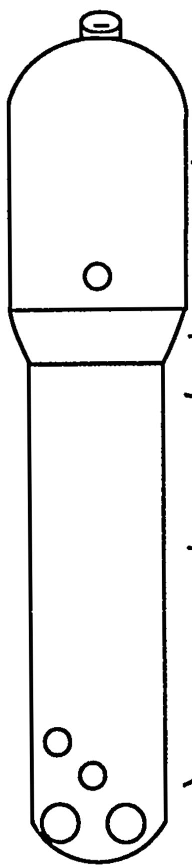
YANKEE NUCLEAR POWER STATION
CRP RADIATION EXPOSURE SUMMARY

ALARA REVIEW	TASK DESCRIPTION	ESTIMATED PERSON-REM	ACTUAL PERSON-REM
93-002	CRP ASBESTOS ABATEMENT SCAFFOLD PROJECT	27.6	18.9
93-003	CRP ASBESTOS ABATEMENT PROJECT	121.8	53.7
93-004	SHIELDING TO SUPPORT CRP	2.5	1.1
93-006	VC COMPONENT REMOVAL	72.3	44.0
93-006	SHIELD TANK CAVITY PREPS AND MODIFICATION	5.4	2.6
93-007*	RX VESSEL INTERNALS SEGMENTATION	32.7	6.2
93-008	COMPONENT PREPARATION	22.9	12.5
	TOTALS	**285.2**	**138.0**

*Exposure through 12/21/93. (Work is continuing under this review in 1994.)

FIGURE 1

AVERAGE STEAM GENERATOR DOSE RATES IN MREM / HOUR



LOCATION	SECONDARY SIDE FULL OF WATER CONTACT	SECONDARY SIDE FULL OF WATER @ 1′	SECONDARY SIDE DRAINED CONTACT	SECONDARY SIDE DRAINED @1′
STEAM DRUM	1.3	0.7	24	17.8
STEAM DRUM TRANSITION	58	39	360	226
TOP OF TUBE BUNDLE	138	79	475	325
TUBE BUNDLE CENTERLINE	94	50	350	250
CHANNEL HEAD AREA	175	88	563	319

Internals Segmentation

1. Evaluate mechanical vs plasma cutting technology in terms of costs for cavity water clean-up systems and costs for filter handling and disposal.

2. Design the water clean-up system to remove the soluble Co-60 generated during plasma cutting.

3. Design the filtration system with high flow rates to reduce delays from water clarity.

4. Consider an underwater curtain arrangement to prevent the migration of cutting fines into the cavity areas outside the cutting table.

5. Design tooling used for underwater cutting to eliminate crevices that can trap cutting debris.

When the plant shutdown occurred, many of the areas in the containment were shielded for maintenance activities. Additional shielding was applied directly on the loop piping adjacent to the steam generators and remained in place throughout the entire project. Pipe cuts on the hot and cold legs of the steam generators were performed with automatic mechanical cutters. Workers practiced installing, cutting and disassembly of equipment on a S/G mock-up with field operations providing recommendations based on time and motion studies. After securing and bracing of the S/Gs, two cuts were made on each leg and a 1 inch ring was removed. Shielding, trained workers, and minimal mechanical failures resulted in a dose saving of about 4 person-rem.

Each steam generator and the pressurizer was removed from the containment using the same crane that installed them during construction. Radiation surveys were performed outside containment in a low background area to determine final shielding requirements and waste classification. A 56 wheel transporter was used to move the S/Gs from the yard area to a S/G preparation facility. The steam generator preparation facility was designed for two S/Gs, using shield walls and tents for exposure and contamination control. Portable ventilation units were used to filter and direct fumes from welding and potential airborne contamination into the plant ventilation system where all releases could be monitored. Following preparation, the S/Gs were transported by road approximately six miles to a rail line, loaded onto rail cars and shipped to Barnwell. The pressurizer was lowered directly onto a standard flat bed trailer, for over-the-road shipment to Barnwell.

Reactor Internals Segmentation

The collective dose required to remove and dispose of the reactor vessel internals was estimated to be about 33 person-rem. As of March 30, 1994 approximately 9 person-rem has been used. Table 2 summarizes the different task estimates and expenditures as of March 30, 1994. The reactor vessel internals and the thermal shield are being removed from the reactor vessel (underwater) and segmented using a plasma torch and metal disintegration machining.

The internals segmentation and packaging plan was developed by Yankee in conjunction with Power Cutting, Inc., Chem-Nuclear System, Inc. (CNSI) and WMG, Inc. The plan identified packaging requirements for the various reactor internal components based on activation analyses, transport cask design limits, regulatory classification, transportation criteria and segmentation equipment capabilities.[2]

Several liners were staged in the cavity for blending various reactor internals components. Underwater radiation surveys were performed to determine dose rates, and then activity calculations performed to determine the final loading and characterization.

TABLE 2

ALARA REVIEW SUMMARY

Rx VESSEL INTERNALS SEGMENTATION

TASK #	TASK DESCRIPTION	ESTIMATED HOURS	ACTUAL HOURS	PERCENT OF ESTIMATED HOURS	ESTIMATED DOSE (REM)	ACTUAL DOSE (REM)	PERCENT OF ESTIMATED DOSE	EFFECTIVE DOSE RATE (REM/HOUR)
1	SEGMENT AND PACKAGE Rx INTERNALS	13000	7522	57.9%	19.500	4.722	24.2%	0.006
2	DOSE PROFILE Rx INTERNALS COMPONENTS	2000	1035	51.7%	3.000	0.545	18.2%	0.0005
3	LINER LOADING AND CASK HANDLING	4800	793	16.5%	7.200	0.815	11.3%	0.0010
4	LOWER INTERNALS HANDLING	250	101	40.2%	0.750	0.110	14.7%	0.0011
5	SPENT FILTER REMOVAL AND SHIPMENT PREPARATION	120	216	179.8%	0.600	1.255	209.2%	0.0058
6	SECURITY COVERAGE	1500	0	0.0%	1.200	0.000	0.0%	0.0000
7	UPENDER OPERATION AND SURVEILLANCE	600	87	14.5%	0.480	0.140	29.2%	0.0016
REVIEW TOTALS		22270	9752	43.8%	32.730	7.587	23.2%	0.0008

ORGANIZATION AND STAFFING

A CRP management team was created from a selection of plant and corporate personnel to begin the process of selecting contractors, developing engineering packages, planning project activities and developing project schedules. Key individuals were selected for project managers to oversee the asbestos removal, S/G removal, pressurizer removal, heavy hauling/lifting, engineering, crane support, and internals segmentation.

Throughout the CRP, emphasis was placed on maintaining radiation exposures ALARA. The ALARA program used during plant operation was sufficient to meet the requirements of the Component Removal Project. Yankee Nuclear Services Division, which was responsible for project management and engineering, incorporated RP Engineering recommendations into each Engineering Design Change Request package. In addition, RP Engineering established the ALARA controls used during the CRP. Due to the dynamic nature of the project, RP staff responsibilities shifted from ALARA and job planning, to RP coverage and LLRW management as the work in progress changed.

The construction organization developed many time saving methods to increase efficiency, minimize rework and reduce exposure. Debriefings were held after each steam generator removal and full advantage was taken of lessons learned. Three welding machines were purchased for steam generator shield welding based on increased productivity and exposure savings. Total dose saving due to use of welding machines was determined to be about 8 Person-rem.

The YNPS RP organization was reduced to 7 people at the end of 1992 to support license conditions and infrastructure needs of a permanently shutdown facility. During CRP, at peak loading, about 25 RP technicians, 25 decontamination - rad waste handlers, 2 lead RP technicians and 2 radiological engineers were used to supplement the plant RP staff (Radiation Protection Manager, 2 RP Engineers, 3 technicians, and 1 dosimetry clerk). Total on-site contractor and YNPS staff (excluding security) varied from a peak of about 200 people during asbestos removal to about 125 people for removal and preparation of the S/Gs and the pressurizer and the reactor internals segmentation, packaging and shipping.

ALARA PLANNING

Radiation exposure projections at the start of the CRP were conservatively estimated to be 285 person-rem. Early involvement with engineering and scheduling is expected to result in a dose savings of about 100 person-rem. Key areas where radiation exposures were avoided are presented in the Radiological Engineering, and ALARA Reviews and Exposure Estimates sections.

RADIOLOGICAL ENGINEERING /ALARA

Steam generator removal and reactor internals segmentation were the two areas of primary focus within the Radiological Engineering group. These tasks had the greatest potential for high personnel radiation exposures.

Steam generator removal required a significant effort to remove asbestos insulation, to remove physical interferences and to prepare the vessels for lifting. The water in the secondary side of the steam generators provided significant shielding to personnel during these preparation efforts. Draining the steam generator resulted in a two to three fold increase in dose rates (refer to Figure 1, Steam Generator Average Exposure Rates). Engineering concentrated on methods to perform as much work as possible with this shielding in place without compromising personnel or plant safety. The work plan allowed the vessels to be shimmed under their support lugs, rigged to the crane and the vessel supports cut free with the water still in the secondary side of the vessel. The only work which could not be performed with the secondary side filled was the cutting and capping of the three lower secondary small bore nozzles. These efforts resulted in dose savings of about 20 person-rem.

The S/Gs were fabricated in the late 1950s. Fabrication was conducted in accordance with ASME section VIII for unfired pressure vessels resulting in a shell wall thickness of 2.75 inches adjacent to the tube bundle. This wall thickness combined with the S/G source term required 2 inches to 2.5 inches of steel shielding be installed on the vessel shells to meet the transportation dose rate criteria. Fabrication and installation of this shielding in successive 0.25 inch to 0.5 inch layers would have proved exposure intensive. Mapping the steam generator shells for lug and nozzle interference locations allowed for the prefabrication and rolling of large coverage, 1 inch to 2.5 inches thick shield plates. These plates were placed on the steam generators in the yard area with a mobile crane. The shielding was tack welded in position and final welding was performed after concrete injection in the preparation facility. This sequence saved an estimated 5 person-rem.

The reactor internals segmentation was performed entirely underwater. Engineering efforts focused on controlling cutting debris and maintaining shielding (water) between personnel and segmented components. Underwater tooling and rigging were engineered or marked as appropriate to maintain water shielding over segmented components.

Reactor vessel internals segmentation began October 14, 1993. The reactor vessel internals, comprising 19 separate components, are estimated to contain 1.235 million Curies and weigh about 125,655 pounds. About 80% of the total radioactivity is contained in the core baffle which will be cut and stored on-site as greater than Class C material. Through March 30, 1994 there have been eight 8-120 cask shipments and sixteen 3-55 cask shipments.

Based on radiochemistry data, initial operation of the plasma cutting torch resulted in a small percentage (<1%) of activity (predominately Co-60) in a soluble state. The concentration of soluble Co-60 gradually increased over a two month period of cutting low activity components. Cavity clean-up is currently being supplemented by the plant mixed bed ion exchange system to remove soluble compounds.

Airborne contamination is controlled with a floating hood which is positioned above the cutting table during the cutting process. Two portable HEPA units draw a suction from the hood and discharge to the intake of the YNPS containment purge system. The hood arrangement has been very effective in reducing airborne radioactivity. Air samples taken under the hood show typically 1E-8 uCi/cc and breathing level air samples are typically less than 1E-10 uci/cc (gross beta-gamma).

ALARA REVIEWS AND EXPOSURE ESTIMATES

Seven ALARA reviews were developed for CRP to address various areas of the work scope. Table 2 provides a description of the ALARA reviews, exposure estimates and actual exposure information.

The detailed ALARA reviews continued the process begun by the radiological engineering group during design development. The ALARA reviews established controls for work activities and provided guidance for activity sequencing. In conjunction with the development of the ALARA reviews, ALARA personnel were continuously involved in the planning and scheduling process. Every attempt was made to eliminate, simplify or increase the efficiency of work activities without compromising personnel safety.

WASTE MANAGEMENT

YAEC and CNSI prepared a safety analysis report for the purpose of obtaining a Certificate of Compliance for the S/Gs as a Type A container. The pressurizer was certified as a DOT Type 7A container. Shielding was designed to maintain radiation levels on the package about 75% of the DOT limit. Contamination levels were maintained less 1000 dpm/100cm^2 loose beta-gamma.

The Steam Generators were shipped as Class A stable waste (>A_2 quantity LSA). The S/Gs were filled with a low density (0.33g/cc) concrete to fix contamination and facilitate specific activity calculations for LSA. The activity calculation was performed using MicroShield's[4] model for a cylinder volume source with side shields and assuming the source volume and mass were represented by the tube bundle and the concrete which filled the annular space between the tube bundle and the outer shell and inside the tubes. The average activity of an individual S/Gs was about 325 Curies. The critical nuclides for waste classification were Ni-63, Pu-241 and TRU's with a halflife greater than 5 years.

SUMMARY AND CONCLUSIONS

The CRP has been a successful project. Due to strong commitment to radiation safety, personnel exposures during the CRP have been maintained ALARA. With the costs for waste disposal projected to escalate and limited availability of waste sites, early dismantlement of the steam generators, pressurizer, and reactor vessel internals has been cost effective with no negative impact on the site radiological conditions and no effect on the environment around the site.

REFERENCES

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Author Biography

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COLD WEATHER EFFECTS ON DRESDEN UNIT 1

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INTRODUCTION

Dresden Unit 1 is in the final stages of a decommissioning effort directed at preparing the unit to enter a SAFSTOR status. Following an extended sub-zero cold wave, about 55,000 gallons of water were discovered in the lowest elevation of the spherical reactor enclosure. Cold weather had caused the freezing and breaking of several service water lines that had not been completely isolated. Two days later, at a regularly scheduled decommissioning meeting, the event was communicated to the decommissioning team, who quickly recognized the potential for freezing of a 42" diameter Fuel Transfer Tube that connects the sphere to the Spent Fuel Pool. The team directed that the pool gates between the adjacent Spent Fuel Pool and the Fuel Transfer Pool be installed, and a portable source of heat was installed on the Fuel Transfer Tube. It was later determined that, with the fuel pool gates removed, and with a worst case freeze break at the 502' elevation on the Fuel Transfer Tube (in the Sphere), the fuel in the Spent Fuel Pool could be uncovered to a level 3' below the top of active fuel.

Brief History and Status of Dresden Unit-1

The Dresden Nuclear Power Station, which includes Units 1, 2, and 3 is located on a 953 acre site near the confluence of the Des Plaines and Kankakee Rivers, about 50 miles southwest of Chicago in Grundy County, Illinois. The nearest population center of Morris, Illinois, is located eight miles to the West.

Dresden 1 was a first generation, turnkey, demonstration plant that was the first full-scale privately financed nuclear power plant in the United States. When built, Dresden 1 was the largest single operating nuclear reactor in the world. Initially, it was rated at 180 MWe (net) and was subsequently up-rated to 210 MWe.

Dresden 1, owned and operated by Commonwealth Edison, received its construction permit May 4, 1956 and its operating license, DPR-2, November 16, 1959. Commercial power operation between August 1, 1960 and October 31, 1978 generated approximately 15.8 million MWhrs of electricity.

On October 31, 1978, CECo suspended operations of Dresden 1 to refuel, perform major system modifications, to add a High Pressure Coolant Injection (HPCI) system, and to perform a major primary system chemical cleaning. Following the Three Mile Island-2 accident, the cost of additional modifications grew to more than $300 million to bring the unit into compliance with federal standards. Company officials concluded that the age of the unit, together with its relatively small size, (compared to the available power at the time), made such an investment impractical. On August 31, 1984, it was announced that Unit-1 would be retired.

In July 1986, the Dresden 1 provisional operating license was amended to a possession-only status. Amendment No. 36 to operating license DPR-2 continued CECo's authority to possess the facility and its contents, and permitted maintenance of Dresden 1 in its present status. By maintaining the facility in the proposed manner, the safety of the public would be assured.

The Dresden 1 reactor was de-fueled in 1978. All 464 spent and partially-spent fuel assemblies in the reactor were discharged to the pool. There are currently 660 assemblies stored in the Spent Fuel Pool (SFP) and 23 assemblies in the Fuel Transfer Pool (FTP). These fuel assemblies will remain in their present storage locations until permanent disposal alternatives are available. CECo is presently pursuing alternate fuel storage options, including dry fuel storage.

Dresden is currently undergoing preparations for the dormancy period as part of our SAFSTOR plan for decommissioning. A chemical decontamination of the primary system, completed in September 1984, removed the bulk of the internal contamination.

Areas of Unit 1 no longer required to be vital areas have been devitalized.

Primary systems have been drained, and supporting systems are being reviewed for lay-up options. The Fuel Service Building is being cleaned, and a new ventilation system is being installed. Presently, the existing pool filtration system in the SFP is not operational and will remain inoperative. A portable demineralizer/filter system is being installed for water chemistry control. Fuel pool storage rack metal monitoring coupons are in place and being trended for corrosion rate.

Unit 1 procedures have been reviewed, and revisions are in progress. A Unit 1 structural monitoring program has been developed, and baseline walk-downs are near completion.

A baseline radiological survey is complete, and quarterly radiological surveys are being conducted in accordance with Reg. Guide 1.86.

CHARACTERISTICS OF UNIT 1 FUEL HANDLING SYSTEM

The fuel handling system essentially consists of a water shielded path for the spent fuel to travel from the reactor vessel to the Fuel Service Building (located outside of the containment), equipment for handling the fuel and various reactor parts, a water shielded storage pool, and shipping and receiving facilities.

To move fuel from the reactor vessel, the following operations were performed. The reactor vessel head and turning vanes were stored in the fuel handling canal. Fuel assemblies were withdrawn from the reactor using the fuel bundle grapple and hoist.

Sixteen fuel assemblies were placed in a Fuel Transfer Basket in the refueling canal. These baskets were lifted with the 7-1/2 -ton auxiliary hook of the reactor service crane and lowered 55 feet through the 42-inch diameter Fuel Transfer Tube into the Fuel Basket Carrier located in the Fuel Transfer Tunnel (beneath the Sphere)

To store the used fuel in the pools, the Fuel Basket Carrier was moved through the tunnel to the south end of the transfer pool by means of a cable drive assembly. The fuel assemblies were lifted from the Fuel Transfer Basket using the fuel bundle grapple and hoist, and were moved to the Spent Fuel Pool to be stored in fuel storage racks.

The Dresden 1 fuel pools consist of two to three foot thick concrete walls that were poured into excavated bedrock. The pools do not have a liner, but the original construction included an epoxy-type concrete coating.

The site grade elevation is at the 517' elevation. The SFP bottom is at the 494' elevation, and the FTP is about 20 feet deeper to accommodate the Fuel Transfer Tunnel. The top of active fuel is approximately the 505' elevation in the SFP.

A gate between the SFP and the FTP is located at the north end of the pools. The gate consists of two sections (one above the other) to isolate the SFP from the FTP. This gate is of an older design, must be lifted vertically and completely out of the pool to be removed, and extends completely to the bottom of the SFP. The normal configuration of this gate is out per the equipment manual. Due to the fuel grapple design, the gates were left out to allow movement of the fuel grapple bridge between the two pools. The design of the SFP is unusual in that it is relatively shallow (~ 25 feet) which requires that the fuel be moved horizontally between rows of racks, and then tilted into a storage location.

JANUARY 1994 COLD WEATHER EVENT

Following an extended sub-zero cold wave, on January 24, 1994, water was discovered on the floor of the Unit-1 Offgas Filter Building which was unheated and not in service at the time of the event. Unable to isolate the frozen and thawing pipes and valves associated with the affected Service Water System, the Operating Dept. shut down the Unit-1 Service Water System. The following day, water and ice were discovered in the basement of the Unit-1 containment during a quarterly radiation survey (Reg. Guide 1.86). The majority of this water was traced to frozen and thawing service water lines inside the containment. Shutting down the Service Water System as a result of the Offgas Filter Building leak precluded additional leakage of service water into the Sphere.

Heat had been discontinued to the Sphere in the spring of 1989 following maintenance problems with the steam heating boiler. A subsequent engineering review to determine the acceptability of not continuing heat to the containment structure erroneously assumed that all systems inside the containment had been appropriately isolated and drained. In fact, some systems could not be isolated by valve closure except inside the unheated containment, leaving a portion of the piping subjected to freezing conditions.

Service water leakage in the Offgas Filter Building was contained within the floor curbing. The estimated 55,000 gallons which collected in the Sphere basement was eventually pumped to radwaste for processing. The water in the Sphere was found to be contaminated, probably from the flushing of material from the floor drains in the Sphere basement.

On January 27, 1994, it was realized that the freezing conditions in the Sphere basement could subject the vertical fuel transfer tube to freezing. To preclude any significant consequences from freezing and possible rupture of the fuel transfer tube inside the containment (possibly below the top of active fuel in the Spent Fuel Pool), the pool gates separating the Fuel Transfer Pool and the Spent Fuel Pool were immediately installed. Temperature readings of the Transfer Tube were obtained by use of remote heat sensing equipment. The temperature below the isolation valves in containment was 64 ^{0}F and above the valve 36 ^{0}F. No freezing of this tube was identified and electric heat and a surveillance program for the tube were instituted immediately.

An internal investigation was begun, the NRC issued a Confirmatory Action Letter, and the NRC dispatched an inspection team to the site for a two week period to examine the event and other aspects of the decommissioning plan for Unit 1.

Although a UT inspection of the transfer tube on February 16th identified the possibility of a gas pocket under the 42" isolation valve on the Fuel Transfer Tube, this was later proven not to exist via direct sampling and measurement of tube contents. The UT signals are believed to have been influenced by a heavy accumulation of material on the inside of the transfer tube. This may have resulted from normal corrosion of the carbon steel pipe and from the existence of microbiological growth in the pool during the course of a few years in the late 1980's.

The investigation into the pipe freezing event also identified concerns with inoperable HVAC systems, fuel pool water inventory monitoring, possible siphoning paths from the fuel pool through the original (but no longer used) decay heat removal and filtration system, and a lack of management attention to the decommissioning effort. Fuel pool integrity continues to be examined and water inventory better documented.

CORRECTIVE ACTIONS

1. The Service Water system was shut-down to preclude additional water leakage into the Offgas Filter building, which also stopped water ingress into the Sphere

2. Pool gates were installed to protect the fuel stored in the Spent Fuel Pool section, which was vulnerable to a loss of integrity of the Fuel Transfer Tube (inside the Sphere)

3. Heat was applied to the Fuel Transfer Tube inside the containment to preclude freezing and a surveillance was established to assure that the tube was adequately heated and monitored. This eventually included remote monitoring via CCTV and thermocouple read-out.

4. Water in the sphere was pumped to radwaste for processing.

5. An in-house investigation was conducted to determine the cause of the event and to recommend corrective actions.

6. In-progress system walk-downs were accelerated to identify additional sources of piping inside the Sphere which could possibly be charged with water and be subjected to freezing and rupture. Several pipes associated with Service Water and Contaminated and Clean Demineralized Water were cut and capped, and additional (less vulnerable) pipes have been identified for future cut and caps.

7. A procedure controlling the fuel pool gates is being written to provide more positive control over when they can be removed. Gate installation will provide additional separation between the majority of the fuel and the transfer tube.

8. A new Unit 1 organization has been formed to dedicate resources to the decommissioning effort, in place of utilizing management personnel with shared responsibilities with the operating units on-site.

9. Emphasis on decommissioning activities has been heightened

10. Water inventory monitoring of the fuel pool has been improved

11. In-progress efforts aimed at achieving SAFSTOR status have been accelerated and intensified to assure compliance with the program and the NRC. SER on SAFSTOR.

RESULTS OF ROOT CAUSE ANALYSIS

The removal of heat from the Sphere was the result of the limited scope of the engineering studies which only evaluated the potential impact on four systems and the Sphere structure itself. The assumptions that other systems had been appropriately isolated and drained were incorrect. This resulted from:

1. Deficiencies in personnel knowledge and training concerning the transfer tube

2. Deficiencies in communications between Station and Engineering personnel regarding which systems and components had been drained and where they should be isolated

3. Inadequate application of 10 CFR 50.59

4. The absence of a formal review and approval of the engineering studies by either the station or engineering.

The root cause of other decommissioning issues identified as a result of the investigation into the pipe freezing and water spills is related to organizational and priority deficiencies associated with the decommissioning effort. Unit 1, an early generation nuclear unit, is not closely related to the other Dresden or CECo units in design or level of available documentation on system design or operation. The early retirement of Unit 1 (due to age and design concerns) compared to the Units 2 & 3 resulted in a lack of focused attention on the multi-unit site. Decommissioning was being addressed by existing staff without adequate guidance, oversight, or expectations and generally decommissioning assumed lower priority in staff assignments compared to issues associated with the operating units on-site. The "Order to Authorize Decommissioning of Dresden Nuclear Power Station, Unit 1, and Amendment No. 37 to License No. DPR-2" were received on September 3, 1993.

SUMMARY OF DRESDEN UNIT-1 DECOMMISSIONING CORRESPONDENCE

The decision to retire the unit was announced on August 31, 1984. In July 1986, Dresden received a license amendment bringing the unit to a possession-only status. Decommissioning plans and revisions were submitted in December 1987, April 1988, and February 1992. The "Order to Authorize Decommissioning of Dresden Nuclear Power Station, Unit 1, and Amendment No. 37 to License No. DPR-2" were received on September 3, 1993.

POTENTIAL EFFECT ON THE FUTURE OF DECOMMISSIONING ACTIVITIES

✓ The NRC is likely to take a much closer look at some of the older fuel storage systems in the industry, in particular those of shut-down plants. Emphasis is likely to be on water mass balance and inventory, and trending of water additions to fuel pool systems.

✓ Depending upon the vulnerability of the fuel pool to leakage, for units in SAFSTOR, adequate demineralization may be required to limit the total amount of Cesium activity suspended in the pool water. Leaking fuel pins can result in a significant amount of Cesium inventory in the water if it is not removed through demineralization.

✓ The NRC is likely to scrutinize decommissioning submittals with much more detail. This will probably include follow-on visits to spot check key details of a proposed decommissioning plan.

✓ The decommissioning process itself may change based upon some of the lessons learned by key NRC officials during their visit to Dresden and some other shut-down plants. Identification of areas of risk, and lessons learned could make decommissioning issues clearer, and the information exchange process quicker.

✓ A possible lesson learned is that the industry may see more emphasis on the prompt dismantlement option of decommissioning vice the SAFSTOR option. Prompt dismantlement allows a utility to "gear up" and "put the unit to rest" quickly, thereby providing less long term risk for the utility to manage. Closure of waste disposal sites, however, is a major deterrent to prompt dismantlement.

- ✓ The industry may see an accelerated and greater emphasis on alternate fuel storage systems such as dry fuel storage, especially for utilities with older or obsolete design fuel pools.

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Author Biography

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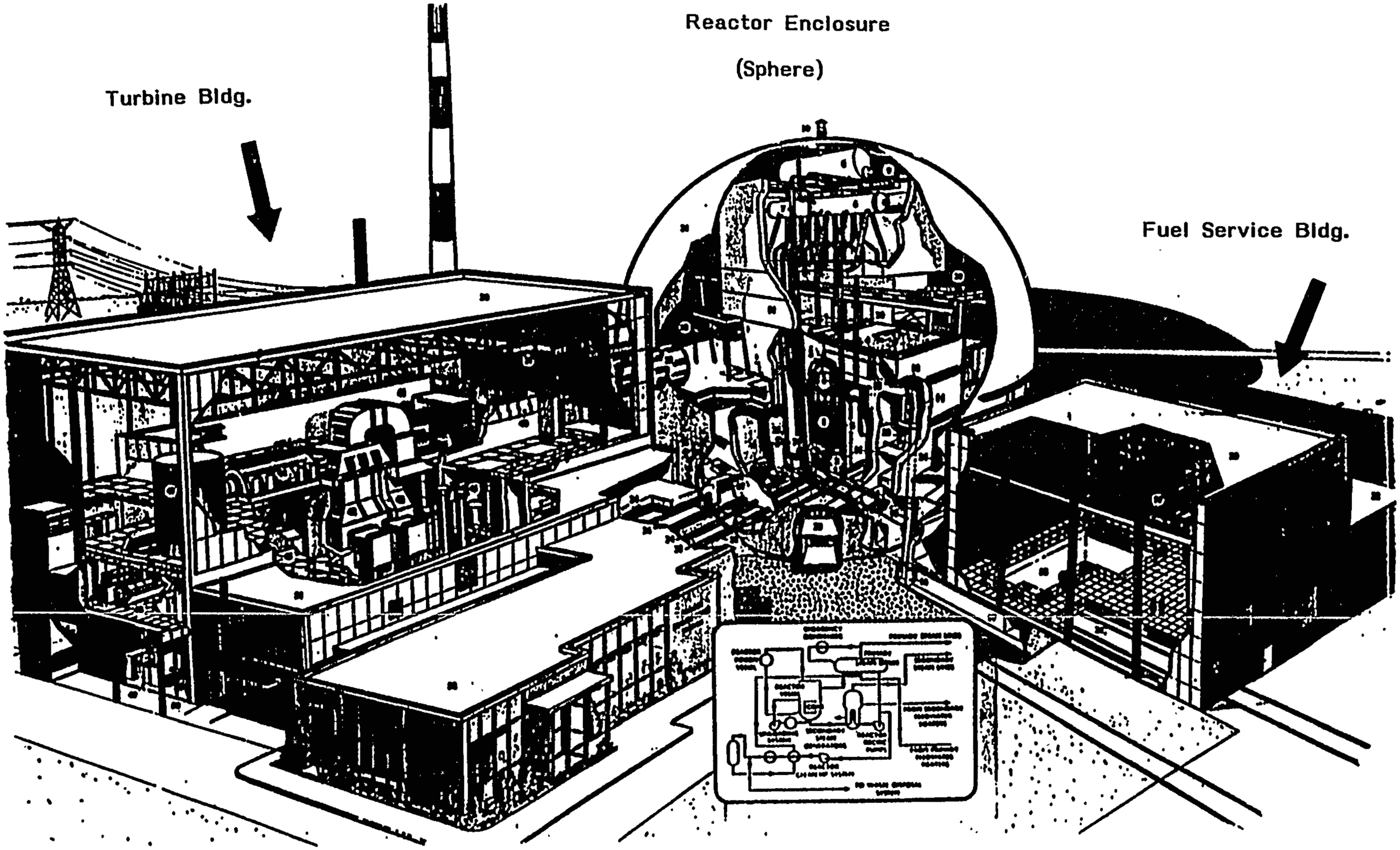
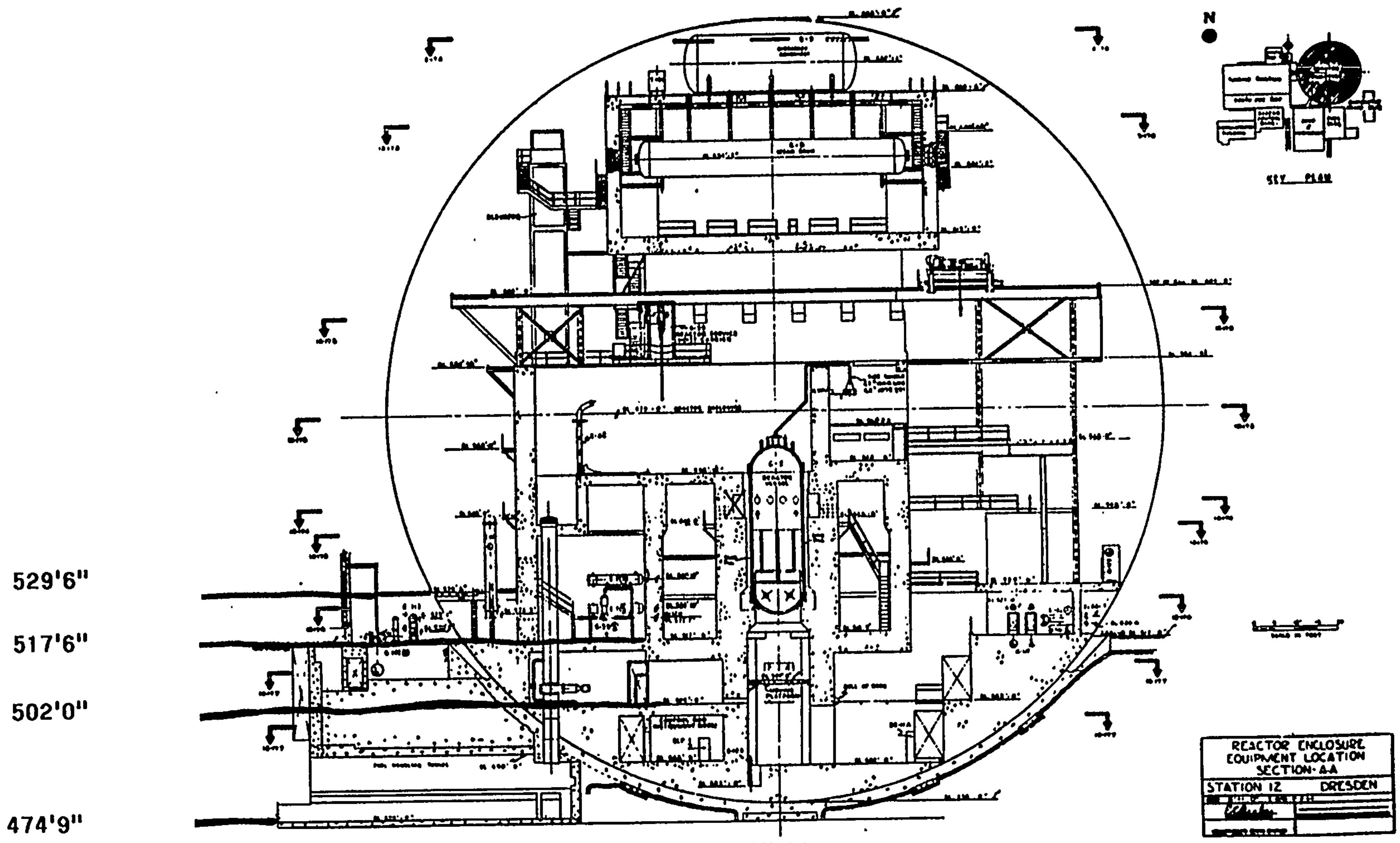
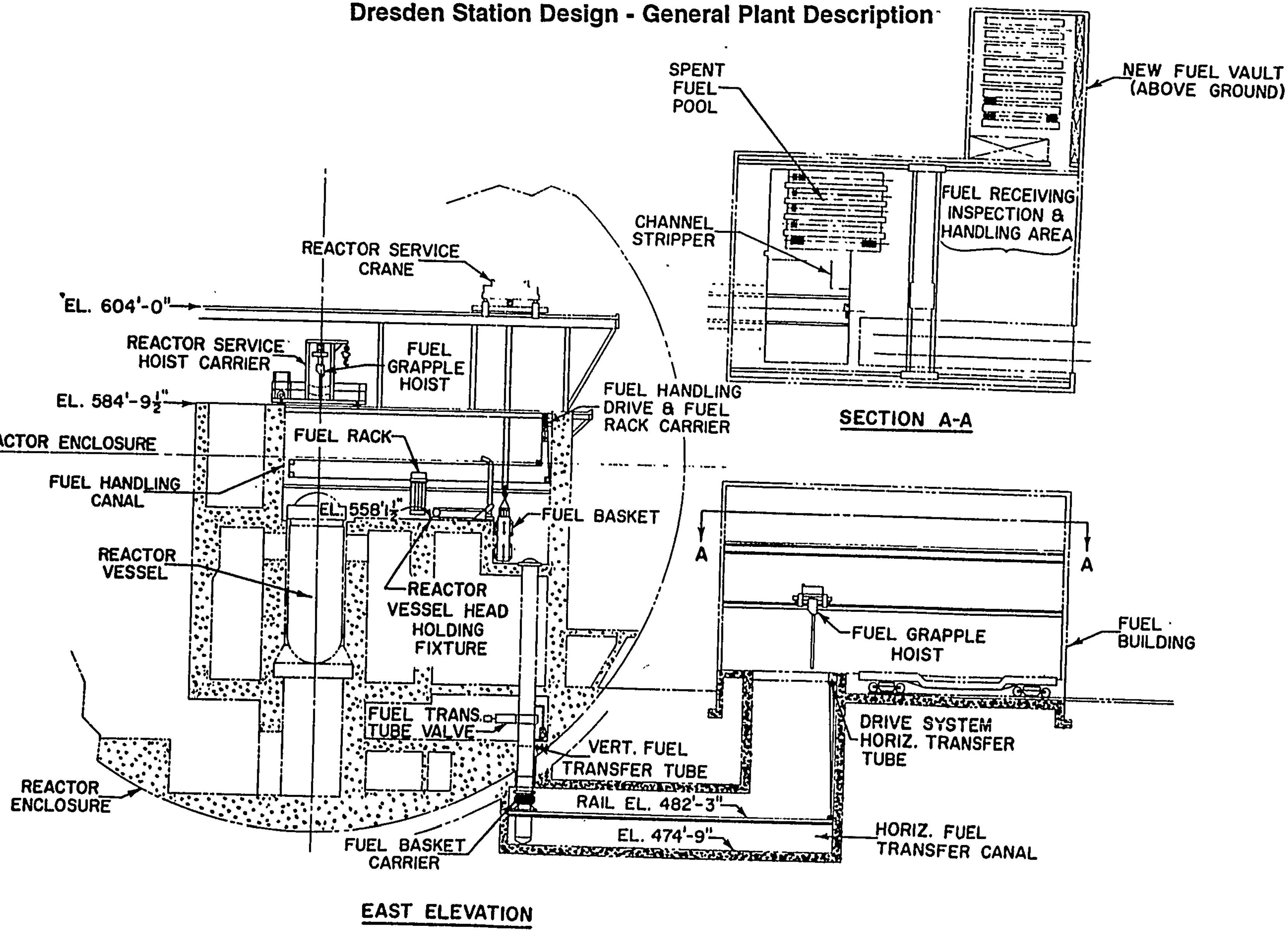
DRESDEN UNIT-1

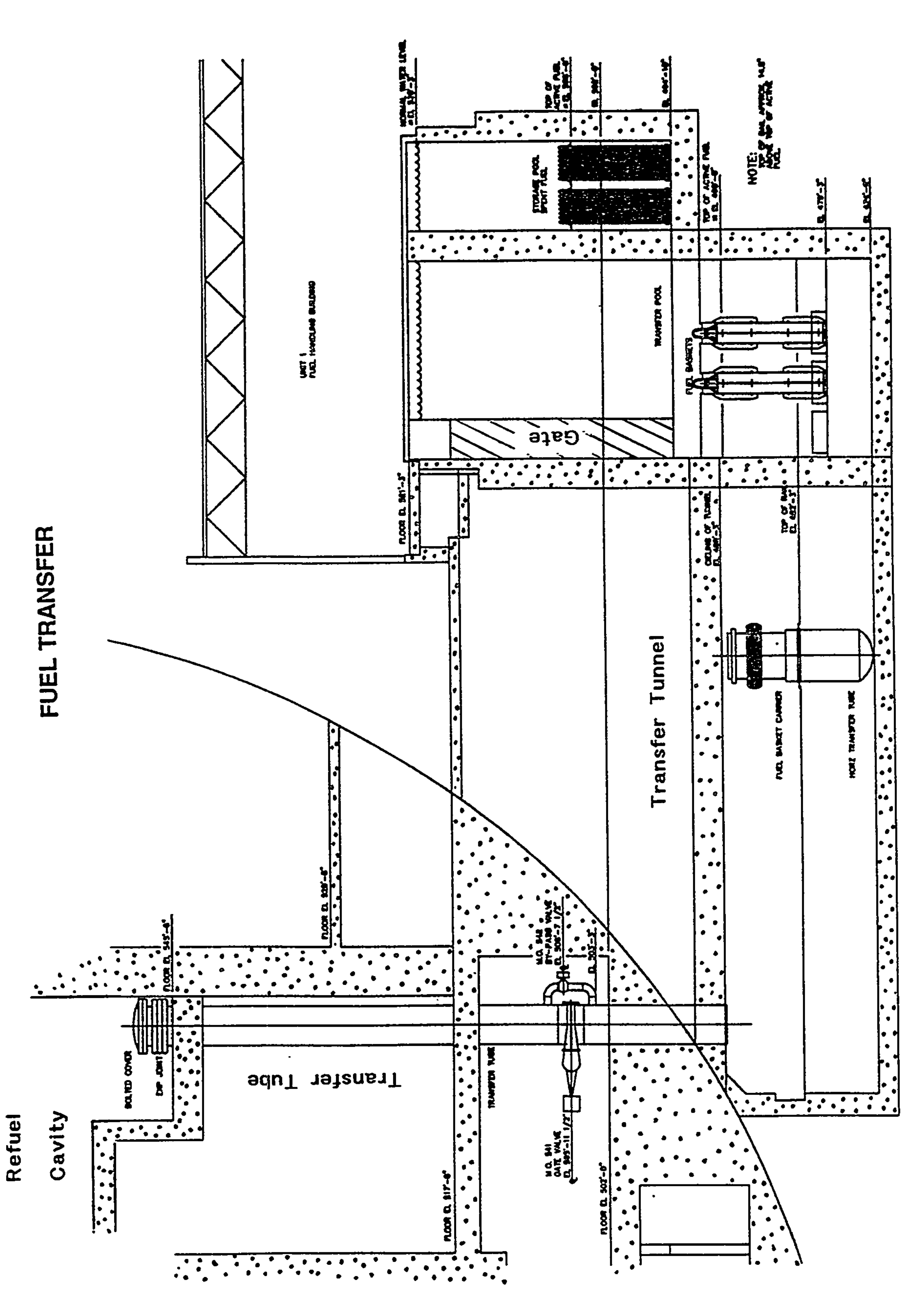


SPHERE - ELEVATION VIEW





FUEL HANDLING SYSTEM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Transfer Tube
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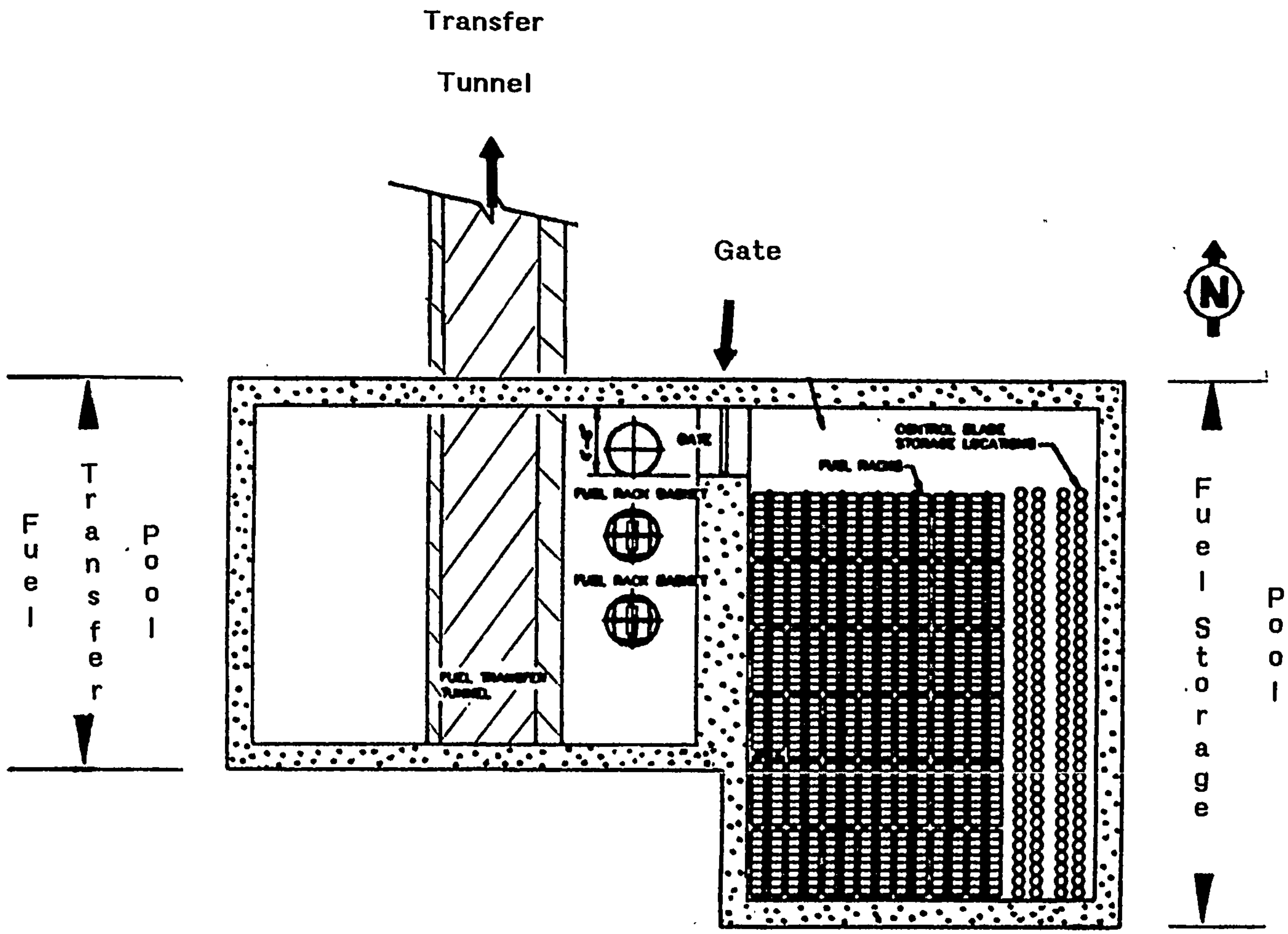
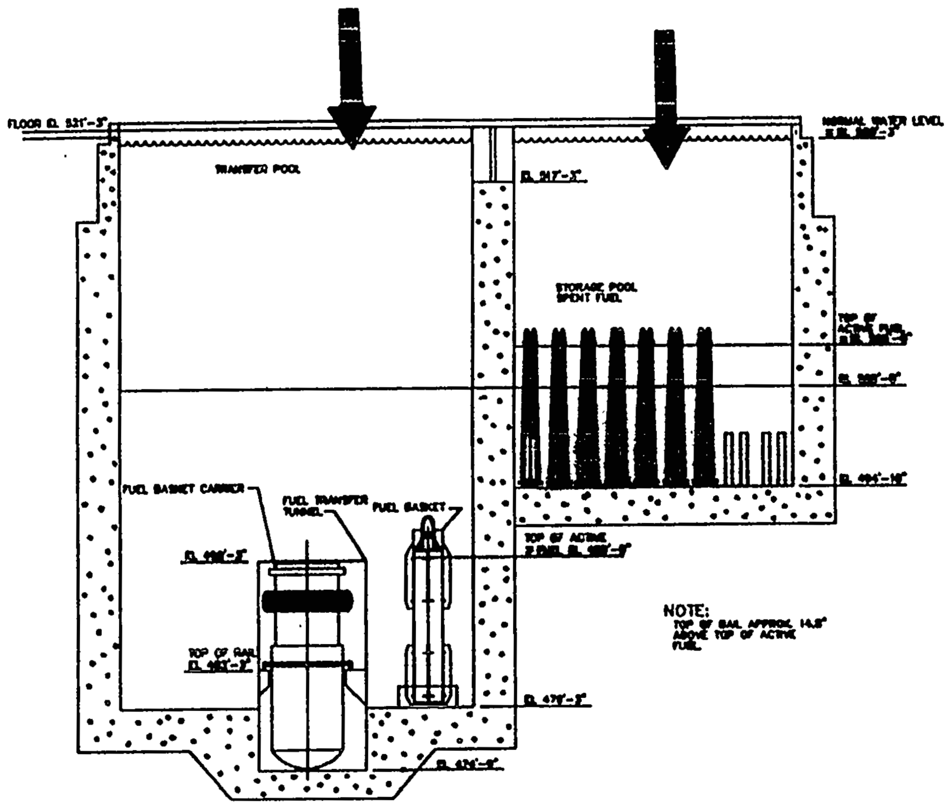
NOTE:

DRESDEN UNIT-1 FUEL POOLS

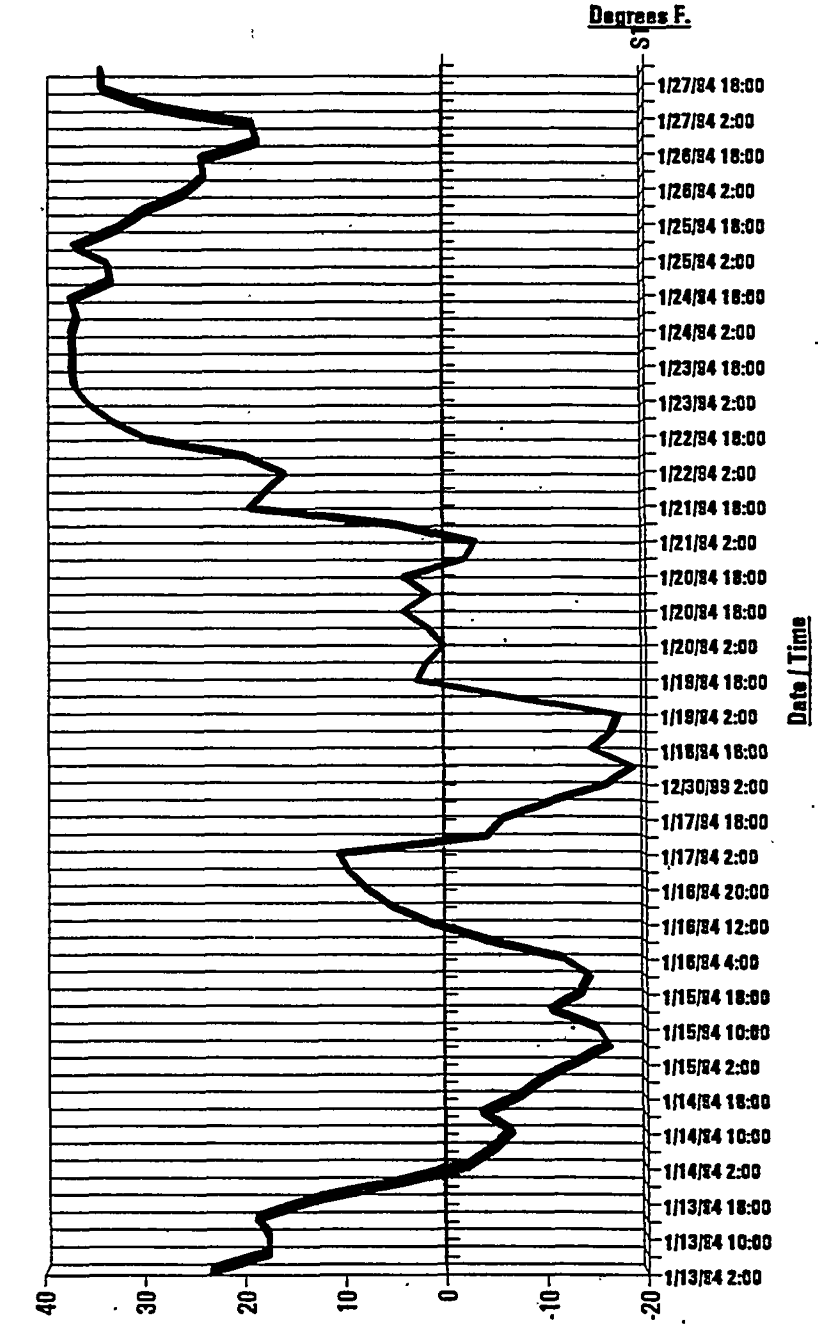


FUEL POOLS

Transfer Pool Storage Pool



External Air Temperature at Dresden Station



SESSION 10

DECONTAMINATION OF NUCLEAR POWER PLANTS

Chair:

Christopher J. Wood

RECENT DEVELOPMENTS IN CHEMICAL DECONTAMINATION TECHNOLOGY

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INTRODUCTION

Chemical decontamination of parts of reactor coolant systems is a mature technology, used routinely in many BWR plants, but less frequently in PWRs. This paper reviews recent developments in the technology - corrosion minimization, waste processing and full system decontamination, including the fuel. Earlier work was described in an extensive review published in 1990 (ref. 1).

Part System Chemical Decontamination Experience

In the 30 months up to March 1994, over forty part-system decontamination applications were carried out by U.S. vendors, primarily at operating nuclear power plants in USA. A similar number of applications were carried out by vendors in other countries. Table 1 summarizes the decontamination applications carried out in the United States between September 1991 and March 1994. Clearly, chemical decontamination is a mature technology that is playing a major role in the worldwide drive to reduce occupational exposures. For example, applications of the LOMI (low oxidation-state metal ion) process have saved over 13,000 person-rem (cSv.) between 1986 and 1992, equivalent to $132 million using typical exposure costs.

Most applications have been carried out on BWR reactor water cleanup systems and recirculation piping systems, but several applications on PWR channel heads and heat exchangers have been reported, also. Typically, stainless steel or carbon steel components are decontaminated, although low alloy steels and nickel-based alloys, such as Inconel 182 and 600, are included frequently in the flow path for the decontamination solvents.

Most proprietary decontamination reagents consist of mixtures of organic acids and/or chelating agents, such as citric acid, oxalic acid and EDTA (ethylene diamine tetra acetic acid). These mildly reducing agents are often used in multi-step processes with oxidizing steps, such as alkaline potassium permanganate (AP), nitric acid/permanganate (NP) or permanganic acid. The LOMI process uses a mixture of vanadous formate and picolinic acid, which is more strongly reducing than the purely organic processes. LOMI is also used in combination with AP or NP steps, particularly for removing chromium-rich oxide films formed in the reducing coolant chemistry of PWRs and in some cases in BWRs operating with hydrogen addition (ref 1).

Examination of Table 1 indicates that different processes are favored for different applications. LOMI is the process of choice for BWR recirculation piping (where intergranular stress corrosion cracking is a major concern), while the regenerable CANDECON and CITROX processes are favored for systems with high surface/volume ratios, such as heat exchangers.

All the above reagents are currently used in relatively dilute formulations which give decontamination factors of 5 - 15, adequate for operating plants. However, changes in water chemistry can make the radioactive oxide films more tenacious and difficult to dissolve. The use of hydrogen water chemistry combined with zinc injection in BWRs has the potential to produce an oxide containing more chromium and zinc than typical; such oxides have proved to be difficult to dissolve in the past.

These dilute reagents have the advantage of not completely removing the inner, high-chromium, protective film on stainless steel surfaces. As a result, recontamination rates are not excessive, and in fact chemical passivation is not so effective on decontaminated surfaces as on fresh surfaces found on newly replaced components. For this reason, passivation techniques are not applied after chemical decontamination. In contrast, the strong chemical processes used 15-20 years ago resulted in a rough oxide-free surface that recontaminated more quickly than even newly-installed components, often resulting in higher radiation fields after one cycle of subsequent operation than measured before the decontamination.

As discussed above, these dilute reagents are not able to completely remove all the radioactivity in the oxide film, as required for free release of decommissioned components. For decommissioning applications, stronger oxidizing reagents (such as cerium compounds) or electrochemical processes are required. Typically these processes remove some of the base metal, which is acceptable for replaced components but obviously not for components to be returned to service.

Corrosion Issues

Concerns about intergranular stress corrosion cracking, particularly of sensitized Type 304 stainless steel in BWRs and of mill-annealed alloy 600 in PWRs, have lead to comprehensive corrosion testing with decontamination reagents. These tests included measurements of general corrosion, galvanic corrosion and crack growth measurements with stressed specimens.

For BWRs, accelerated corrosion is generally observed only when oxalic acid is present, and even then only in isolated cases (ref 2). Tests on irradiated materials, subject to irradiation-assisted stress corrosion cracking, showed no adverse effects. A detailed evaluation of all corrosion data by General Electric concluded that the LOMI process was qualified for use throughout the entire reactor system (ref 3). Insufficient data (particularly crack growth measurements) were available for other processes, and the same conclusion was reached for the alkaline permanganate oxidizing step. Thus, while no adverse effects had been reported, unrestricted endorsement of AP for complete reactor coolant system use has not been obtained. Nitric permanganate (NP) is not recommended, as a result of accelerated cracking with low alloy steels. There is a need for improved oxidizing pretreatment processes, such as permanganic acid used in the CORD process (ref 4), to be qualified for BWR use.

As part of the qualification program for PWR full system decontamination, corrosion characteristics of 40 materials in AP/LOMI and AP/CANDEREM have been established. These tests covered three cycles of normal decontaminations and one cycle of decontamination under fault conditions, corresponding to the most aggressive situation that could result from loss of process control. Typically, the one fault cycle gave as much corrosion as three normal cycles. Both processes were deemed acceptable for at least one application (ref 5).

Waste Processing

Radioactive waste processing and disposal has become the main impediment to more extensive use of decontamination technology. Improvements in the formulation and application of decontamination processes have been made to reduce waste volumes. For example, improved ion exchange resins, such as ion-specific resins are being developed. For LOMI, two changes have already been implemented: the use of "low formate" reagent and a reduction in the amount of picolinic acid used to maintain the dissolved metals in solution. With all such improvements the gains achieved are economically worthwhile but limited.

Several more radical technical developments in this area are providing alternatives to the current practices of disposing of ion exchange resins in high-integrity containers, or solidified in cement. Low temperature resin oxidation (ref 6) and Vitrification techniques are methods currently under development that have the objective of substituting a compact quantity of chemically-inert residue for the relatively large volume of ion exchange resin that results from current practice. Vitrification converts ion exchange resin into a glass matrix, suitable for long-term storage or disposal. Wet oxidation processes use hydrogen peroxide to oxidize organic material, leaving a residue which is readily incorporated in cement.

Electrochemical processes, such as the ELOMIX (electrochemical LOMI ion exchange) process, hold the promise of revolutionizing decontamination waste processing (ref 7). The objective of the ELOMIX technique is to reduce the volume of waste arising from the LOMI process by continuously removing the radioactive elements from solution using an electrochemical cell.

Ion exchange resin is used as an intermediate, rather than a final, waste form, and is continuously regenerated by the passage of electric current. The radioactivity is converted to a particulate metallic deposit which can be transported hydraulically to a vessel for encapsulation.

There are three main benefits of the process: smaller waste volume, inorganic waste form and regeneration of the chemical reagents; the latter reduces chemical costs, while the other two facilitate long-term storage, allowing most of the radioactivity to decay before disposal. Small scale field-tests during routine decontaminations at Dresden and

River Bend BWR power plants have demonstrated the feasibility of the concept, and provided parametric data required for the design of full-scale electrochemical cell modules.

PWR Full System Decontamination

Complete reactor coolant system decontamination has long been used in pressure tube reactors (CANDU and SGHWR) but has not been applied to large commercial reactors in the United States. However, full-system decontamination offers several important advantages: lower background fields, more effective decontaminations, and reduced recontamination rates (ref 8).

Significant strides in the area of PWR full-system decontamination have been made recently, with a plant demonstration planned at Con Edison's Indian Point 2 (IP2) station in 1995. Radiation fields at the plant have increased to a point where they are above the industry average. Numerous efforts to address the radiation exposure have not yielded the desired results.

In 1988 a qualification program for the chemical decontamination of the entire RCS of a Westinghouse PWR began. This qualification program was successfully completed in 1991. The LOMI and CAN-DEREM processes, with an alkaline permanganate (AP) conditioning step, were qualified for use as long as the fuel was removed.

An estimation of recontamination rates based on industry experience and computer analysis indicated a full-system decontamination would achieve a DF of 5 and that the benefit would last for 5 operating cycles, or at IP2, about 10 years. Potential exposure savings were then calculated for a range of plants. The exposure that could be avoided over 5 operating cycles ranged from 1000 rem to 3500 rem.

The measured recontamination rates of previously decontaminated subsystems at IP2 are well below what had been projected in the full-system decontamination report. Two factors could be that IP2 has maintained the higher pH values recommended for the RCS and has worked to replace cobalt sources in the equipment. If the recontamination rates were to continue at the low levels, the radiation levels may never return to the original levels and the exposure saved could be even greater than originally estimated.

Development of the procedures and equipment for the decontamination are well advanced. A 6-day schedule for implementation of the decontamination process has been established. CAN-DEREM and AP will be alternated during the process.

The IP2 full-system decontamination will be done with the fuel removed. A separate program started in 1989 to qualify nuclear fuel for full-RCS decon application. This program involves the chemical decontamination of actual fuel assemblies in a specialized canister at the V. C. Summer PWR plant, using the same dilute chemical solvent parameters as were employed in the full-RCS qualification program.

To take account of current generation fuel and future generation fuel designs, one assembly of Westinghouse Vantage 5 and one assembly of Vantage-Plus type fuel were exposed to CAN-DEREM solvents and one assembly each was exposed to LOMI solvents.

The qualification program necessitated the design and fabrication of a specialized decon test loop at the same conditions of flow, temperature, pressure, and chemical environment as would exist for the fuel in the reactor vessel during full-RCS decontamination. The specialized loop was designed, fabricated, tested, and installed in 1991. Extensive TV visual and eddy current cladding oxide thickness inspections were performed before and after exposure to solvents with the same inspections planned after one full cycle of operation. That operating cycle was completed in March 1993 and inspections were performed on the non decontaminated control assemblies and the fuel assemblies subjected to decontamination treatment. Preliminary evaluation of the cladding oxide thickness data shows no significant cladding corrosion performance differences between any of the assemblies. High-magnification TV visual examination of the grids, grid springs and assembly nozzles, and hold downsprings show no adverse effects.

A number of decontamination process application anomalies were observed. These anomalies resulted in recommendations for further study of boron control, ion exchange resin utilization, carbon dioxide generation, EDTA chemical analysis and decomposition, picolinic acid analysis and decomposition, vanadous formate oxidation, and reassessment of metals and radioisotopes removed. Other general recommendations included revisiting each facet of the study to extend the evaluations to include the "fuel-in" full-RCS decontamination scenario and submittal of a second Topical Report to the NRC for its approval.

BWR Full System Decontamination

Full-system decontamination applied to BWRs is less complex than for PWRs. Thus, the BWR programs have been smaller. In a collaborative CECO/EPRI project, the LOMI process was qualified and a safety review prepared. That work, which used Quad Cities as the reference plant, has now been extended to more modern BWR5 designs, such as LaSalle. In another project an engineering design study based on the Brunswick plant has been completed.

Full-system decontamination for BWRs could be economically attractive for removing deposits in the lower parts of the BWR cores to aid inspection/repair or to remove radioactive material that could be redistributed to out-of-core areas on switching to hydrogen water chemistry. It is interesting to note that, whereas originally the main concern with BWR decontamination was the potential for increased corrosion resulting from attack by the decontamination chemicals, full-system decontamination is now seen as potentially helpful in overcoming corrosion problems with core internals.

The LOMI process has already been tested successfully on BWR fuel in a test at Quad Cities BWR in 1986. This project was similar in many ways to the PWR fuel test described above. However, far more radioactive material was removed than anticipated, thus reducing the probability of future applications. Development of the ELOMIX electrochemical ion exchange process is continuing and this could well change the economics of all decontamination applications in the future.

Conclusions

This paper has identified a number of challenges that have been overcome, such as corrosion qualification, others where significant progress has been made (waste processing) and the future challenges of full system decontamination. Chemical decontamination has been one of the most significant causes of the reduction in radiation fields (and hence exposures) that U.S. plants have achieved in recent years. Looking ahead, further advances in decontamination technology will provide enhanced options as the industry faces the challenge of controlling radiation exposures during major repair/replacement work.

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TABLE 1

SUMMARY OF U.S. CHEMICAL DECONTAMINATION SEPTEMBER 1991 - MARCH 1994

Date	Plant	System	Vendor	Process	#Steps	Avg DF	Person-Rem Avoided
9/91	Brunswick 2 BWR	RRS	PNS	LOMI	2	10.7	331
10/91	V.C. Summer PWR	Fuel Test	W	LOMI	5	-	N/A
		Fuel Test	W	CANDEREM	4	-	N/A
10/91	Hatch 1 BWR	RWCU	W	LOMI	6	1.8	}700
		RRS	W	LOMI	4	8.4	
1/92	Browns Ferry 3 BWR	RWCU	PNS	LOMI	3	13.4	} 1000
		RRS	PNS	LOMI	3	62	
		FPC	PNS	CITROX	3	8.4	
1/92	Indian Point 1 PWR	SGS	W	CITROX	5	3.7	88
1/92	Quad Cities 2 BWR	RWCU	W	LOMI		2.9	} 513
		RRS	W	LOMI		6.0	
2/92	Browns Ferry 2 BWR	FPC	PNS	CITROX	3	8.1	N/A
2/92	Browns Ferry 1 BWR	FPC	PNS	CITROX	3	7.0	N/A
2/92	D.C. Cook 1 PWR	RTD Bypass	W	CANDEREM	5	1.6	N/A
2/92	FitzPatrick BWR	RRS	PNS	LOMI	3	13.5	350
3/92	Perry BWR	RWCU	W	LOMI	1	11.6	N/A
3/92	ANO-1 PWR	Subsystems	PNS	CITROX	3	5	80
3/92	Duane Arnold BWR	RRS	PNS	LOMI	1	4.5	65
5/92	WNP-2 BWR	RRS	PNS	LOMI	3	4.1	300
5/92	Grand Gulf BWR	RRS	PNS	LOMI	3	16	} 138
		RWCU	PNS	LOMI	4	36	
5/92	River Bend BWR	RWCU	W	CANDEREM	3	16.8	} 504
		RRS	W	LOMI	1	7.6	
5/92	D.C. Cook 1 PWR	RTD Bypass	PNS	CITROX	6	15	N/A
9/92	ANO-2 PWR	Regen HEX	PNS	CITROX	3	11.2	11

Table 1 (Cont.)

SUMMARY OF U.S. CHEMICAL DECONTAMINATION SEPTEMBER 1991 - MARCH 1994

Date	Plant	System	Vendor	Process	#Steps	Avg DF	Person-Rem Avoided
12/92	Brunswick 1 BWR	RHR	PNS	CITROX	1	26	>200
2/93	Dresden-2 BWR	RWCU RRS	W W	CANDEREM LOMI		2.3 4.3	N/A
3/93	Browns Ferry 2 BWR	RRS RWCU	PNS PNS	LOMI LOMI	3 3	70 20	N/A
3/93	Quad Cities 2 BWR	RWCU RRS	W W	CANDEREM LOMI	- -	2.1 8.3	N/A
4/93	Brunswick 1 BWR	RRS	PNS	LOMI	3	9.5	N/A
5/93	Susquehanna 1 BWR	Fuel Pool Systems	W	CANDEREM	N/A	5-6	N/A
7/93	Susquehanna-2 BWR	Fuel Pool Systems	W	CANDEREM	N/A	5-6	N/A
7/93	Duane Arnold BWR	RRS	PNS	LOMI	1	3.2	N/A
10/93	Clinton BWR	RWCU RRS	PNS PNS	CITROX LOMI	1 1	10.8 8.0	N/A N/A
12/93	Cook-1 PWR	Various	PNS	CITROX or CANDEREM	N/A	2-3	N/A
2/94	Perry BWR	RRS	W	LOMI	N/A	N/A	N/A
2/94	Brunswick BWR	RHR RRS	PNS PNS	CITROX CITROX	N/A	N/A	N/A
3/94	Dresden BWR	RRS RWCS	PNS	CITROX	N/A	N/A	N/A
3/94	Quad Cities 1 BWR	RWCU/R HR RRS	W W	CANDEREM LOMI	N/A	N/A	N/A
3/94	LaSalle BWR	RRS	PNS	LOMI	N/A	N/A	N/A

Introduction

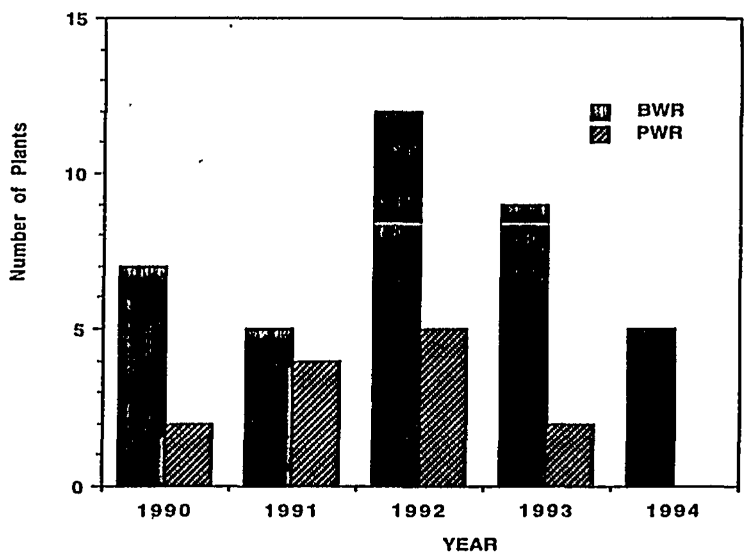
- Recent US chemical decontamination experience
- Technical Issues:
  - decontamination effectiveness
  - corrosion concerns
  - waste disposal
- New Developments:
  - full reactor coolant system decontamination
  - low level waste processing

Recent Decontamination Experience

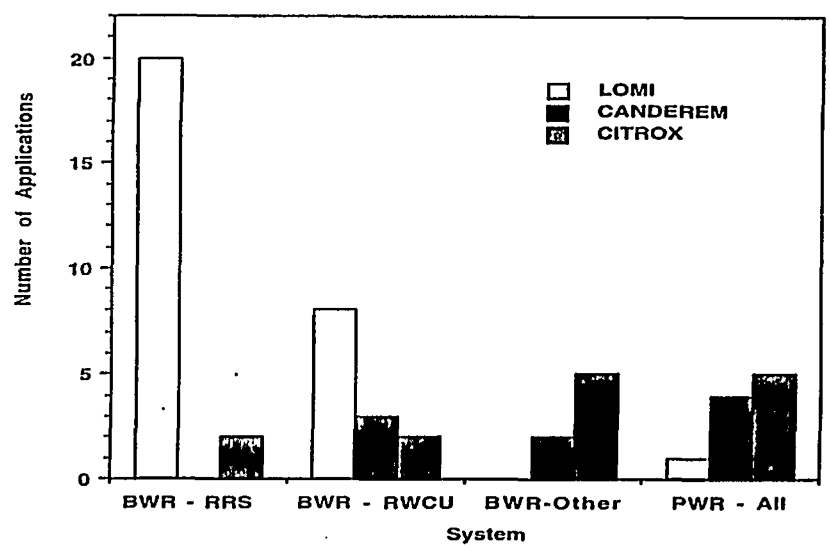
Part-system chemical decontamination is routine at many plants:

	Number of applications 1991-1994
BWR recirculation piping systems	22
BWR reactor water cleanup systems	13
BWR - other	7
PWR - all systems	10
Total	52

U. S. Plants using Chemical Decontamination



Chemical Decontaminations 1991-1994 (U.S. Applications by PNS and Westinghouse)



Corrosion Issues

- Historically, corrosion has been a major concern for US utilities, because of IGSCC experience with Type 304 Stainless Steel and mill-annealed Alloy 600
- Hence BWR requiremeent to show zero increase of IGSCC rates for pre-existing cracks:
  LOMI endorsed by General Electric
- No evidence from over 100 applications of any adverse corrosion effects with any dilute process used at US power plants

Recontamination

- Recontamination rates are typically low:

  BWR recirculation piping - 2 cycles or more to return to pre-decon fields

  PWR recontamination rates are even lower

Low Level Waste Issues

- Radioactive waste disposal is a growing impediment to more extensive use of decontamination in USA
- Several advanced processes under development:

  Wet oxidation of ion exchange resins

  LLW vitrification

  Electrochemical ion exchange
- Goal is small volume of radioactive residue in a stable waste form

Wet Oxidation of Ion Exchange Resins

- Low temperature process
  (100C with hydrogen peroxide, 250C with wet air)
- Removes organics
- Volume reduction factor of 3 -10
- Residue can be incorporated in cement or stored dry
- Used in Europe, Japan and Canada

LLW Vitrification

- Vitrification of HLW in widespread use in France and USA
- Some technical issues with vitrification of wastes with high organic content, e.g. decon wastes
- Major advantage: stable, glass-like waste form
- Also: volume reduction factor of 10-50
- EPRI and VECTRA (formerly Pacific Nuclear) plan demonstrations of advanced process in 1995

Electrochemical Ion Exchange

- On-line process replacing conventional ion exchange
- Residue is chemically inert, metallic material
- Volume reduction factor of about 10
- Field tested on small scale during LOMI plant decontaminations at Dresden and River Bend BWRs in USA
- Developed by EPRI in association with Westinghouse
- Not yet ready for full scale application

Full Reactor Coolant System Decontamination

- Full system decontamination (FSD) with fuel removed has been qualified for use at GE BWRs (LOMI process) and W PWRs (AP-CANDEREM and AP-LOMI)
- First application planned at Indian Point-2 PWR in March 1995 (AP-CANDEREM)
- Currently there are no plans for a BWR application, but potential benefits for reactor internal work as at Oskashamn in Sweden (1993: CORD process)

SYSTEM DECONTAMINATION AS A TOOL TO CONTROL RADIATION FIELDS

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ABSTRACT

Since chemical decontamination of the Reactor Coolant Systems (RCS) and subsystems has the highest potential to reduce radiation fields in a short term this technology has gained an increasing importance. The available decontamination process at Siemens, i.e. the CORD process, will be described. It is characterized by using permanganic acid for preoxidation and diluted organic acid for the decontamination step. It is a regenerative process resulting in very low waste volumes. This technology has been used frequently in Europe and Japan in both RCS and subsystems. An overview will be given i.e. on the 1993 applications. This overview will include plant, scope, date of performance, system volume special feature of the process removed activities, decon factor time, waste volumes, and personnel dose during decontamination. This overview will be followed by an outlook on future developments in this area.

INTRODUCTION

The chemical decontamination process CORD (Chemical Oxidation Reduction Decontamination) of the Siemens Power Generation Group has successfully demonstrated in recent years its dose rate reduction capabilities during large scale technical applications. Table 1 for example summarizes the decontamination projects performed by Siemens in 1993 in Europe and in Japan. It is shown that in this calendar year 33 field applications at components and subsystems were performed. In addition, two major studies for Full System Decontamination (FSD) were awarded to Siemens.

A general experience related to decontamination is: the greater the surface of the system or subsystem , the greater the effect regarding

- Dose rate reduction, especially the area dose rate
- Recontamination.

This can be proven for example by the decontamination and recontamination results generated in the Wuergassen power station as shown in Figure 1.

Table 1. Siemens Decontamination Projects Performed in 1993

Plant	Type of Reactor	OEM	Component / System
Fukushima 2	BWR	GE/Toshiba	Regenerative Heat Exchangers
Fukushima 3	BWR	GE/Toshiba	Regenerative Heat Exchangers
Fukushima 5	BWR	MHI/Westingh.	Regenerative Heat Exchanger
Cofrentes	BWR	GE	Recirculation System Recirculation Pump Reactor Water Clean Up System
Oskarshamn 1	BWR	ASEA	Reactor Water Clean Up System Residual Heat Removal System Feed Water System
Grafenrheinfeld	PWR	KWU	Reactor Coolant Pump
Isar1	BWR	KWU	Reactor Water Clean Up System Bearing Water Supply System for Internal Axial Flow Pumps
Krümmel	BWR	KWU	Fuel Pool Cooling System Reactor Water Clean Up System
Krümmel	BWR	KWU	RPV-Nozzles
Philippsburg 1	BWR	KWU	Reactor Water Clean Up System Bearing Water Supply System for Internal Axial Flow Pumps
Isar 1	PWR	KWU	Reactor Coolant Pump
Biblis B	PWR	KWU	Reactor Coolant Pump
Hamaoka	BWR	GE/Toshiba	Recirculation Loops
Fukushima 6	BWR	GE/Toshiba	Reactor Water Clean Up Pumps
Trillo	PWR	KWU	Reactor Coolant Pumps
Kahl	BWR	KWU	Full System Decontamination
Brunsbüttel	BWR	KWU	Internal Axial Flow Pumps Reactor Water Clean Up System Bearing Water Supply System for Internal Axial Flow Pumps
Biblis A	PWR	KWU	Reactor Coolant Pump
Beznau 1	PWR	Westinghouse	Main Loop Sections
Doel 3	PWR	Westinghouse	Main Loop Sections
Mihama 2	PWR	MHI/Westingh.	Main Loop Sections
Oskarshamn 1	BWR	ASEA	Full System Decontamination Study
Loviisa	WWER	AEE	Full System Decontamination Study

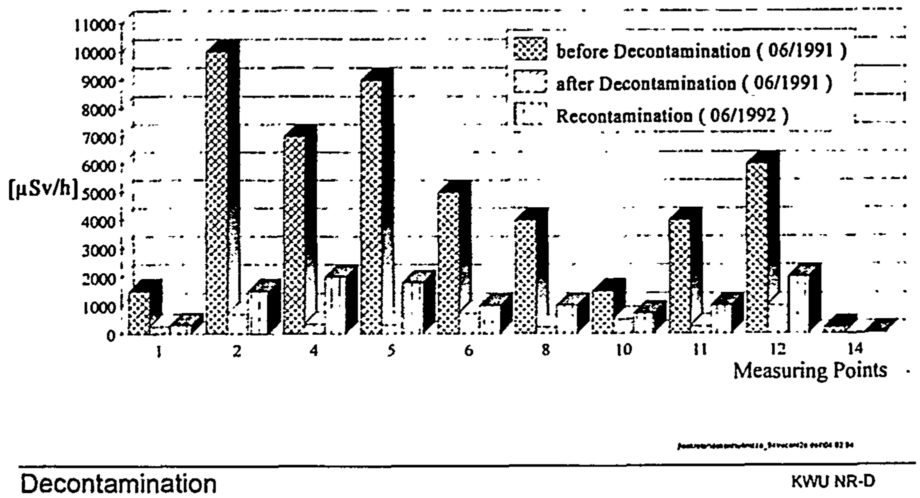
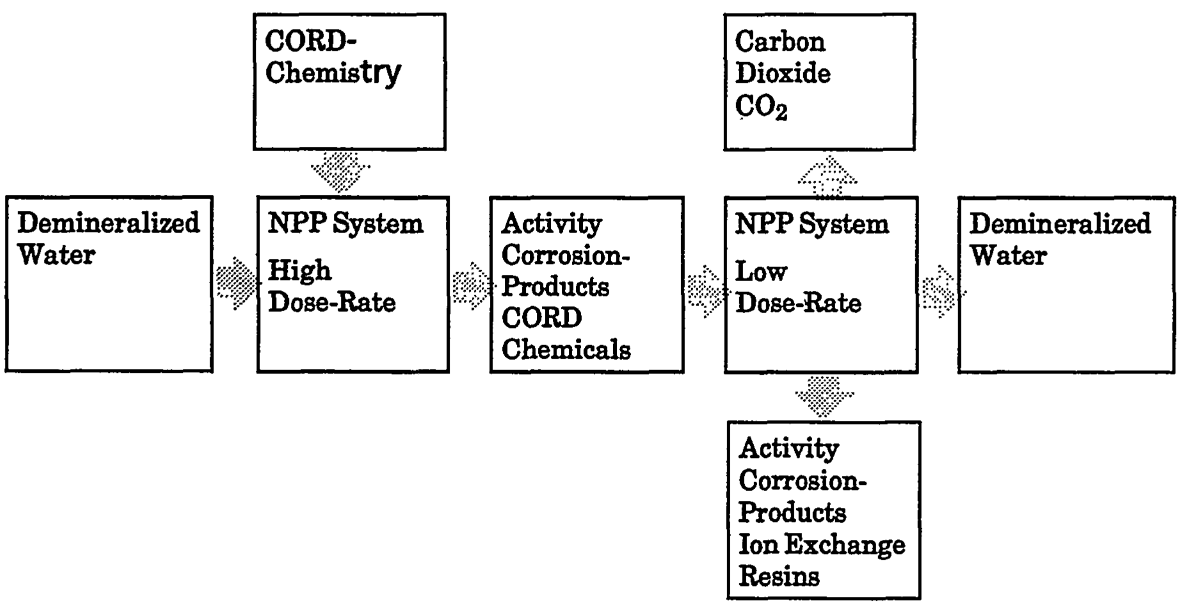


Figure 1. NPP-Würgassen
Recontamination of the Recirculation Loop No 2
after CORD-Decontamination

THE CORD/UV CONCEPT

A description of the CORD process is given in the paper of Wille and Sato.[1] The present paper shall describe the most updated version of the CORD process and some application features. This combination of the chemistry of the process and the application technology results in the CORD/UV-concept as described in the flow-chart as shown below.

CORD/UV - Concept



The reactor system having high radiation field will be filled with demineralized water (one fill) and the CORD chemicals will be injected according to the work procedure. At the end of one CORD-phase (preoxidation + decontamination step) the fluid will contain the activated corrosion products, inactive corrosion products and the CORD chemicals. The next part of the overall concept is to decompose the CORD chemicals into water and carbon dioxide while removing the activity and the corrosion products via ion exchange resins. Through this procedure demineralized water will be left inside the decontaminated system having a water quality close to the make-up water that was filled originally into the system.

RECENT FIELD EXPERIENCE

Recent field experiences with the CORD/UV-concept are summarized in detail in Attachments 1 to 4. They describe the detailed results of

- Oskarshamn 1 — RWCU, RHR, Feedwater System
- VAK — FSD
- Hamaoka — RRL (both loops simultaneously)
- Oskarshamn 1 (OKG-1) — FSD.Because it is not possible to describe all these applications in all aspects, emphasize shall be given to the latest project, the FSD at OKG-1 (see also Reference).[2]

The decontamination measures at OKG 1 became necessary in order to permit extensive inspection and repair work in the reactor pressure vessel. Prior to the decision to decontaminate, the operator of OKG-1 as well as the Swedish Licensing Authorities requested extensive proof of qualification. This means in advance decontamination tests were done with original parts taken from the activated area of the reactor pressure vessel. Bent beam samples of these decontaminated parts (AISI 304 with a 316 weld) were placed into an autoclave and then again exposed to the operating conditions of a boiling water reactor. According to a previously agreed examination sequence, the samples were finallly microsectioned and metallographically examined for surface changes by Studsvik (Sweden).

Based on the positive findings, the CORD process was qualified for RPV decontamination by the licensing authority ands selected for application by OKG-1.

The total system volume amounted to approximately 160 m^3. The process was applied in such a way that the decontamination could be carried out with only one fill of demineralized water and also with plant internal equipment (pumps, heaters, ion exchangers) (see Figure 2).

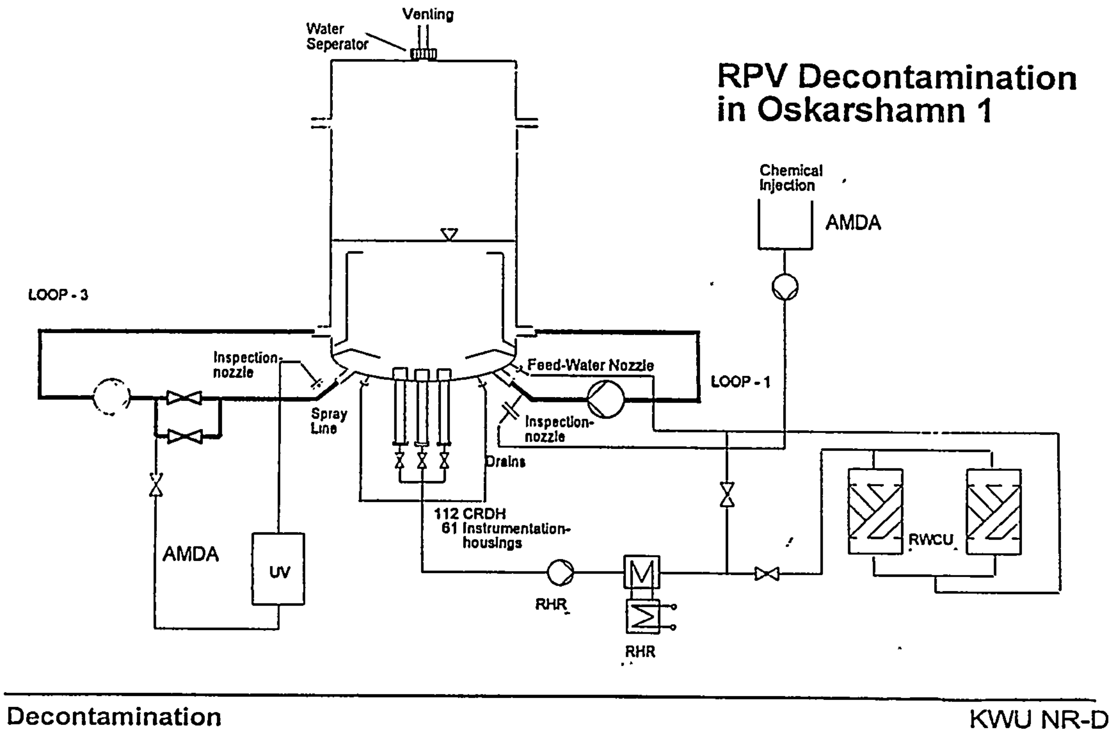
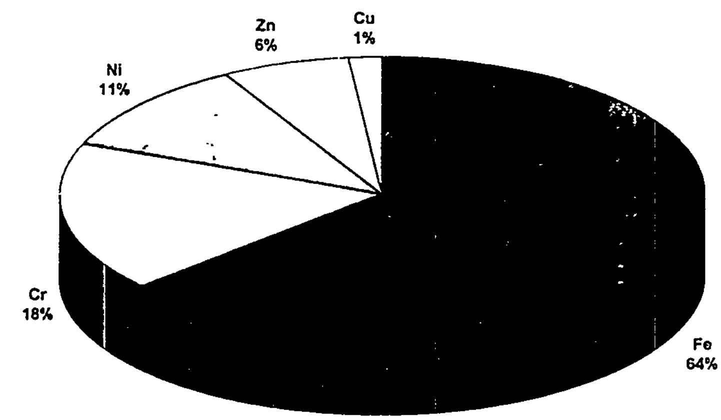


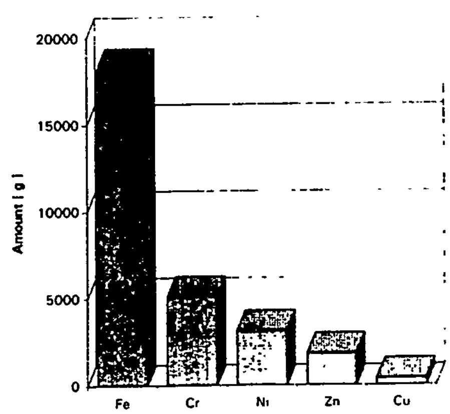
Figure 2. Full System Decontamination CORD/UV Technique

Additional external AMDA (Automated Mobile Decontamination Appliance) components were only supplied by Siemens for chemical injection and for the oxidative decomposition of the decontamination chemicals. The system water had again demineralized water quality (1.5 µS/cm) after completion of the decontamination process. 2.0 m^3 of ion exchange resin in the reactor water cleanup system adsorbed the removed activity (approx. 2.3 E12 Bq) and the dissolved metal ions (approx. 30 kg Fe, Cr, Ni). During decontamination, an oxidative treatment decomposed the decontamination chemicals completely to CO_2, i.e. no ion exchange resins were required for the removal of the decontamination chemicals (see Figures 3 to 6).



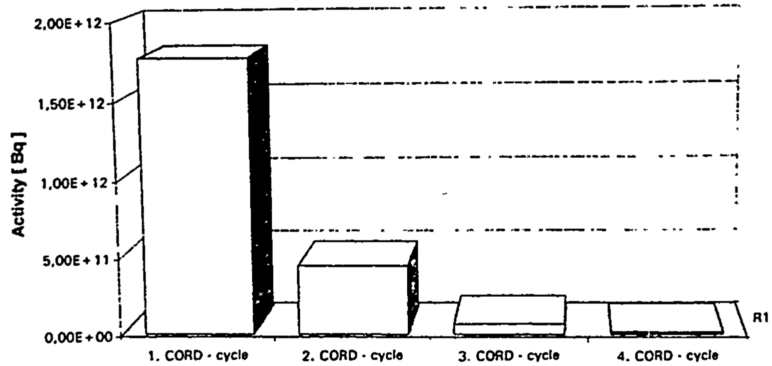
Decontamination KWU NR-D

Figure 3. NPP Oskarshamn 1 - 1994
Full-System Decontamination
Relation of removed Cations



Decontamination KWU NR-D

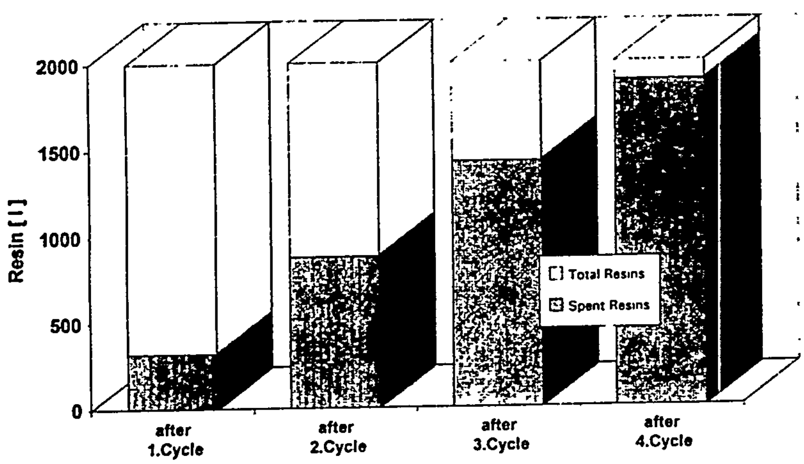
Figure 4. NPP Oskarshamn 1 - 1994
Full-System Decontamination
Cation output after 4 CORD-cycles



Decontamination

KWU NR-D

Figure 5. NPP Oskarshamn 1 - 1994
Full-System Decontamination
Activity Output depending on the CORD-Cycles



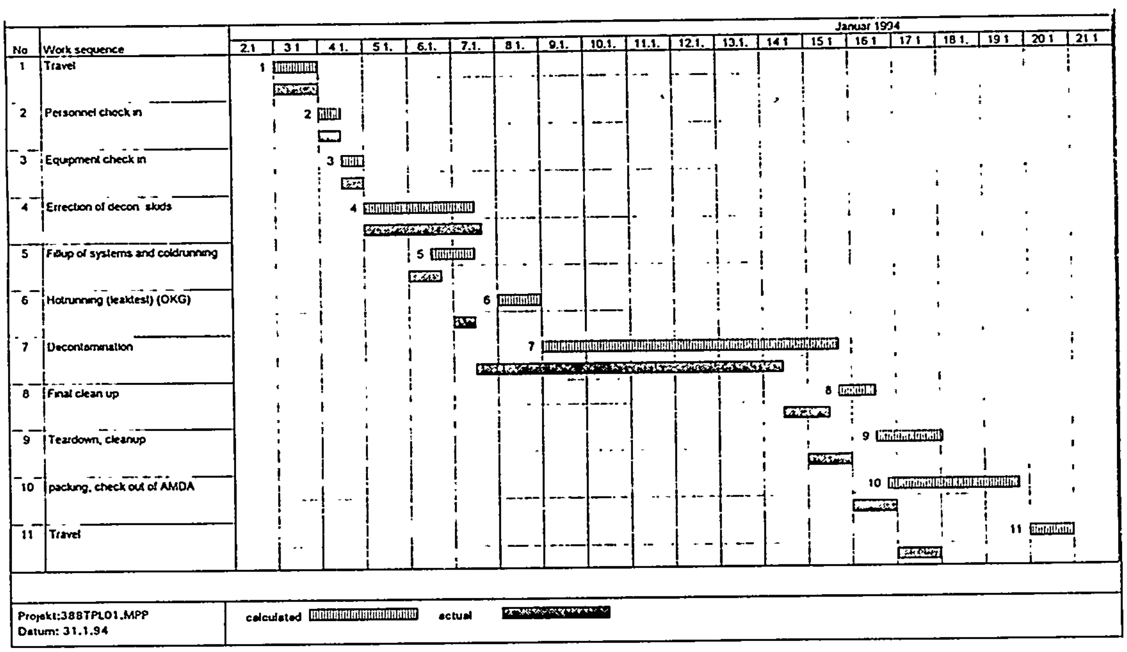
Decontamination

KWU NR-D

Figure 6. NPP Oskarshamn 1 - 1994
Full-System Decontamination
Total Waste Generation

Approximately 20 OKG-1 employees and 9 Siemens employees were taking part during the decontamination. All interfacing and modification work in the power plant was done by OKG-1 personnel. A total of approx. 600 new temporary external interface points or flange connections were required. Siemens AG was responsible for installation of the AMDA components and for performance of the overall decontamination work. The excellent preparatory work done by OKG-1 and the very close and amicable team work between OKG-1 and Siemens personnel was a very important factor towards the excellent decontamination result. This contributed clearly towards the substantial shortening of the time frame scheduled (see Table 3).

Table 3. NPP Oskarshamn 1 Decontamination of the RPV and the 4 Loop's



After decontamination, the area of the RPV to be examined was again cleaned with high pressure water. First measurements taken after the decontamnation resulted in a smearable residual activity of approx. 4 Bq/cm^2 and a dose rate of only 15-20 µSv/h. This extreme low dose rate permits now inspection and repair work within the reactor pressure vessel of OKFG-1 without serious time limitations for the personnel.

Apart from decontaminations performed around the world, all full systrem decontaminations (MOL/Belgium, VAK/Germany, OKG/Sweden) done within the last 3 years were performed with the CORD technology.

In 1993, Siemens received also the contract for a full system decontaminatuion of the NPP Loviisa (Finland) with the CORD process. It is also planned for Loviisa to perform the decontaination as far as possible with plant internal systems. At this time, the decontamination is scheduled to take place in August 1994. The total system volume is expected to be approx. 300 m^3 and approx. 17000 m^2 surface area will exposed to the decontamination solvent

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Author Biography

Rolf Riess is Senior Director for the Power Plant Chemistry Division of Siemens AG KWU in Erlangen, Federal Republic of Germany. The division is responsible for all aspects of Power Plant Chemistry including research and development, and service activities like radiation control, decontamination and steam generator chemical cleaning. Before joining Siemens, Dr. Riess was a scientist for one year at the Institute of Nuclear Chemistry at the Technical University in Darmstadt, FRG. He received a Ph.D. degree in Chemical Engineering from the Technical University of Darmstadt, Federal Republic of Germany.

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PAPER 10-2
DISCUSSION

Egner: You touched on recontamination after the decontamination, which I think is very, very important. Have you, in your methods, some way of passivation, or what do you recommend for treatment?

Riess: We have no method of passivation, but the current thinking on our side is that it would be very useful to consider trace element injection, as an example, zinc injection, to further improve the situation in the plant to avoid that, but we have no special treatment as passivation method.

SIEMENS

Plant : NPP Oskarshamn, Unit 1, Sweden
442 MW, BWR, 1972

System : Reactor Water Cleanup Sytem, Feedwater System and Residual Heat Removal System

Scope:	Personnel dose rate reduction for replacement and service work
Date of Performance:	From end of May until beginning of June 1993
System Volume:	14.5 m^3
Process:	CORD
Particulars:	In-situ wet oxidation of the decon chemicals.

Results:

Removed Activity:	3.5 E 11 Bq		
Decon Factors (contact):	Measuring points:	120	
	Average DF:	16.9 (full)	21.1 (empty)
Decon Factor Area:		10.4 (full)	8.6 (empty)
Time:	72 hours		
Waste Treatment:	Continuous purification with plant internal ion exchange resins Total decomposition of oxalic acid by wet oxidation (photo reactor) Final cleanup through mixed bed		
Waste Volume:	14.5 m^3 water with a conductivity of 1μS/cm (demin. water quality) 225 l Ionexchange resin		
Personnel Exposure:	20.4 mSv		
Client Contact:	- Mr. J. Eriksson (Health Physics) - Mr. K. Ingemarsson (Decontamination) - Phone: Sweden +46 491 860 00		

Plant : **NPP Kahl (VAK)**
16 MW, BWR, 1961

System : **Full System Decontamination**

Scope: Reduction of radiation exposure during dismantling
Reuse of the removed materials by melting

Date of performance: 5/92 and 11/93

System-Volume: 58 m^3

Process: CORD/UV-Technique

Results:

Removed Activity: 6.23 E12 Bq

Decon Results: The results obtained, show that it can be expected that large sections of the primary loop and the auxilliary systems can be reused by melting. Due to the remaining low residual dose rates the personell dose during dismanteling can be reduced to a large extend.

Dose Rates (μSv/h):

	before decon	after decon
RPV cover	30 000	250
other components	400-4000	20-130

Time: 24 days

Waste Treatment: Decomposition of the used decontamination chemicals to CO_2 and water
Removal of the corrosion products, activity and manganese by ion exchange

Waste Volume: Corrosion products: Fe 124 kg, Cr 10 kg, Ni 3 kg
approx. 4.4 m^3 resins and 10 m^3 contaminated liquid
(further volume reduction foreseen)

Personnel Dose: 46 mSv

Client Contact: Mr. Pachl (project manager)
Phone: 41-6188-4990

SIEMENS

Plant : **NPP Hamaoka, Unit 1, Japan**
516 MW, BWR, 1976

System : **Recirculation Piping (Loop 'A' and Loop 'B')**

Scope:	Dose rate reduction for replacement
Date of Performance:	November / December 1993
System Volume:	16 m^3 (in-place) and 5 m^3 (out-place)
Process:	CORD
Particulars:	Recirculation technique, main pipe sections plus pump housings (loop A and B) were in-place decontaminated. Removed pipe and safe end sections were out-place decontaminated In-situ wet oxidation (photo reactor) of the decon chemicals. TOC release rate $\leq$ 1 ppm

Results:

Removed Activity:	4.32 E11 Bq (in-place) and 9.71 E10 (out-place)	
Decon Factor:	Average DF (in-place) Average DF (out-place):	180 (loop A), 97 (loop B) 70
Decon Factor Area:	not known	
Time:	3 days	
Waste Treatment:	continuous purification by ion exchange resins total decomposition of chemicals by wet oxidation (photo reactor) no solvent discharge	
Waste Volume:	1150 l of ion exchange resin (in-place) 725 l of ion exchange resin (out-place)	
Personnel Exposure:	66 mSv (in-place)	100 mSv (out-place)
Client Contact:	- Mr. Y. Sato (Engineering Manager), Toshiba Corp.	

SIEMENS

Plant : **NPP Oskarshamn, Unit 1, Sweden**
442 MW, BWR, 1972

System : **Full-System-Decontamination**

Scope:	Personnel dose rate for inspection and repair work in the Reactor Pressure Vessel
Date of Performance:	January 1994
System Volume:	160 m^3
System Surface:	1500 m^2
Process:	CORD / UV
Particulars:	In-situ wet oxidation of the decon chemicals. No resins required for the removal of decon agents

Results:

Removed Activity:	2.3 E 12 Bq
Remaining Activity:	Smear Test after Hydrolancing 4 Bq/cm^2 in RPV
Decon Factors (contact):	Components (contact) > 1000 Reactor Pressure Vessel 200 - 1300
Decon Factor Area:	≥ 20 Reactor Pressure Vessel
Time:	150 hours
Waste Treatment:	Continuous purification with plant internal ion exchange resins Total decomposition of oxalic acid by wet oxidation (photo reactor) Final cleanup through mixed bed
Waste Volume:	160 m^3 water with a conductivity of < 1.5 $\mu S/cm$ (demin. water quality) 2500 l Ionexchange resin
Personnel Exposure:	5 mSv
Client Contact:	- Mr. J. Eriksson (Health Physics) - Mr. K. Ingemansson (Decontamination) - Phone: Sweden +46 491 860 00

FULL REACTOR COOLANT SYSTEM CHEMICAL DECONTAMINATION QUALIFICATION PROGRAMS

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INTRODUCTION

Corrosion and wear products are found throughout the reactor coolant system (RCS), or primary loop, of a PWR power plant. These products circulate with the primary coolant through the reactor where they may become activated. An oxide layer including these activated products forms on the surfaces of the RCS (including the fuel elements). The amount of radioactivity deposited on the different surfaces varies and depends primarily on the corrosion rate of the materials concerned, the amount of cobalt in the coolant and the chemistry of the coolant. The oxide layer, commonly called crud, on the surfaces of nuclear plant systems leads to personnel radiation exposure. The level of the radiation fields from the crud increases with time from initial plant startup and typically levels off after 4 to 6 cycles of plant operation. Thereafter, significant personnel radiation exposure may be incurred whenever major maintenance is performed. Personnel exposure is highest during refueling outages when routine maintenance on major plant components, such as steam generators and reactor coolant pumps, is performed. Administrative controls are established at nuclear plants to minimize the exposure incurred by an individual and the plant workers as a whole.

A critical objective for the U.S. nuclear industry is the reduction of personnel exposure to radiation, both for continued safe operation of currently licensed plants and for long term acceptance by the public of the nuclear option for power generation. While reductions in personnel exposure to radiation have been achieved through the industry's aggressive radiation management programs, increased plant maintenance and high radiation fields at many sites continue to raise concerns. Unexpected maintenance problems and major equipment replacements have resulted in significant personnel exposure for many plants. In addition, it is likely that new regulations will be forthcoming which could reduce the allowable limits for quarterly and annual exposure. Such restrictions would have a significant impact on productivity and performance. To alleviate the radiation exposure problem, the sources of radiation which contribute to personnel exposure must be removed from the plant. The only economically feasible way of significantly reducing the source term of a pressurized water reactor (PWR) is to chemically decontaminate the entire primary system.

Full RCS Chemical Decontamination Qualification Program - Description and Results

Beginning in 1989 and continuing through 1993, the Electric Power Research Institute (EPRI) and 11 PWR utilities sponsored a program to verify the technical acceptability of using dilute chemical solvent processes for primary system decontamination. Two processes, AP/CAN-DEREM and AP/LOMI, were qualified for use in the RCS of a W PWR. This laboratory research and engineering evaluation program was completed in 1992 and a topical report was approved by the Nuclear Regulatory Commission (NRC) in 1993.

The purpose of the program was to define and complete a systematic evaluation of the major issues that need to be addressed for the successful decontamination of the entire primary system of a Westinghouse pressurized water reactor system with all fuel removed. The workscope of the overall program was large and encompassed a broad spectrum of engineering evaluations, materials and chemistry evaluations, radiological assessments, and equipment designs.

The program was structured in three major phases: Phase 1 - Initial Parametric Studies, Phase 2 - Decontamination Process Qualification and Detail Engineering Evaluations, and Phase 3 - Detailed Design and Implementation.

Phase 1 of the Program constituted the initial parametric studies to address the major issues related to full RCS decontamination. This evaluation included critical process screening of two commercially available chemical decontamination processes. This evaluation also included consideration of decontamination application with the fuel in and fuel out and basic configuration of the reactor coolant system and auxiliary systems for decontamination application.

Phase 2 of the program entailed detailed engineering and testing evaluations to verify the technical feasibility of applying the two chemical decontamination processes to the generic Westinghouse full reactor coolant system. Phase 2 was divided into seven tasks, as follows:

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

Phase 3 of the program encompasses the detailed design and design implementation effort required to perform the chemical decontamination on a specific demonstration plant. The generic plant evaluation data generated in Phases 1 and 2 is being utilized to perform the plant-specific design effort for a demonstration plant application. Details of this Full RCS Chemical Decontamination National Demonstration, which will be conducted at Con Edison's Indian Point 2 plant in 1995, will be provided in the paper which follows.

All of the tasks described in Phases 1 and 2 above have been completed. The comprehensive process testing program and extensive engineering evaluations of the results of the tests clearly indicate that there are no significant detrimental effects of the chemicals employed in the two proprietary processes tested on primary system materials and components. For most materials of construction, the expected corrosion rates are very low and there is no evidence of intergranular attack (IGA) or stress-corrosion cracking (SCS).

As a result of noted material effects on chrome-plated surfaces, 410 SST and stellite materials, however, certain minimal critical system and component recommended pre- and post-decon inspections and equipment modifications are required during a Full RCS Chemical Decontamination. These highly encouraging results are reported in the Phase 2 Qualification Program Final Report for the subject program.

Based on the success of the Phase 2 Qualification portion of the program, EPRI and Con Edison working with Westinghouse and others are currently completing the detail design and fabrication of equipment and facilities as well as defining the detail operational procedures to perform a first Full RCS Chemical Decontamination in

1995. Efforts are well underway to form a National PWR Demonstration Program for the purpose of funding and managing this important demonstration project.

Fuel Decontamination Qualification Program at V. C. Summer - Description and Results

As noted above, the Indian Point 2 full system decontamination will be done with the fuel removed. Looking beyond this, Westinghouse, at the request of South Carolina Electric and Gas Company, developed a separate program in 1989 to qualify nuclear fuel for full reactor coolant system decon application.

This program involved the chemical decontamination of actual fuel assemblies in a specialized canister in the fuel handling building at the V. C. Summer nuclear station with the same dilute chemical solvent parameters as were employed in the full-RCS qualification program.

To take account of current generation fuel and also future generation fuel designs, one assembly of Vantage 5 and one assembly of Vantage-Plus type fuel were exposed to CAN-DEREM solvents and one assembly each was exposed to LOMI solvents.

Conduct of the Fuel Decontamination Qualification Program (FDQP) necessitated the design and fabrication of a specialized qualification test loop for decontamination of each fuel assembly at the same conditions of flow, temperature, pressure and chemical environment as would exist for the fuel in the reactor vessel during full-RCS decontamination. The specialized loop was designed, fabricated, tested and installed in the fuel handling building at V. C. Summer during 1991.

In August 1991, the two Vantage-Plus assemblies were decontaminated and in October 1991, the two twice-burned Vantage 5 assemblies were decontaminated. All four assemblies plus two twice-burned control assemblies underwent extensive TV visual and eddy current cladding oxide thickness inspections before and after being exposed to the solvent processes, with the same inspections planned after one full cycle of operation in the V. C. Summer plant.

The Summer operating cycle was completed on 6 March 1993 and TV visual and eddy current cladding oxide thickness inspections were performed on the fuel assemblies which had been subjected to decontamination treatment and the non-decontaminated control assemblies. Evaluation of the cladding oxide thickness data showed no significant cladding corrosion performance differences between any of the assemblies. High magnification TV visual examination of the girds, grid springs and assembly nozzles and hold down springs showed no adverse effects of the decontamination processes.

Overall, the fuel material post-decontamination inspection results showed no deleterious effects, with excellent decontamination effectiveness. Preliminary evaluation of the cladding corrosion oxide thickness measurements on the decontaminated and control assemblies indicates that the decontamination treatments have had no adverse affect on the post-decontamination cladding corrosion behavior. In addition, a number of decontamination process application anomalies resulted in recommendations for further study of boron control, ion exchange resin utilization, carbon dioxide generation, EDTA chemical analysis and decomposition, picolinic acid analysis and decomposition, vanadous formate oxidation and reassessment of metals and radioisotopes removed.

Other general recommendations included revisiting each facet of the study to extend the evaluations to include the "fuel-in" full-RCS decontamination scenario and submittal of a second Topical Report to the US Nuclear Regulatory Commission for its approval. To this end, W has developed a Fuel-In Full RCS Chemical Decontamination Qualification Program, which has as its objective, NRC approval of a second comprehensive Topical Report for the Fuel-In Case. Completion of this work and subsequent NRC approval will provide utilities with another key option which provides significant additional benefit in terms of reduced critical path time and lower recontamination rates during a Full RCS Chemical Decontamination. Development of the work scope and schedule for this program has been completed and W is in process of obtaining necessary funding to complete this program. It is anticipated that key results will be available early in 1995.

Author Biography

Phillip E. Miller is Manager of Plant Application in the Nuclear Technology Division of Westinghouse Electric Corp. Mr. Miller acted as Program Manager for the two programs completed as described in this paper and will also be in charge of the Fuel-In Qualification Program. He has 22 years experience with Westinghouse and has managed numerous large, multidisciplined projects. Mr. Miller holds a B.S. Degree in Civil Engineering and has received 9 U.S. Patents to date, many of them in chemical decontamination related technologies.

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Full RCS Chemical Decontamination Programs Programs to be Discussed

- Fuel-out Qualification Program
- Con Edison/EPRI Fuel-Out National Demonstration Program
- V. C. Summer Fuel Qualification Program
- Fuel-In Qualification Program

Introduction

- Condition caused by Co-58 and Co-60 buildup in the primary system
- Nature and tenacity of "crud" is different in PWR versus BWR
- Amount of crud on fuel assemblies is much higher in BWR
- Subsystem decontaminations have become common in the U.S. in both BWR and PWR plants

Cost/Benefit for Various RCS Component / Subsystem and System Chemical Decontaminations

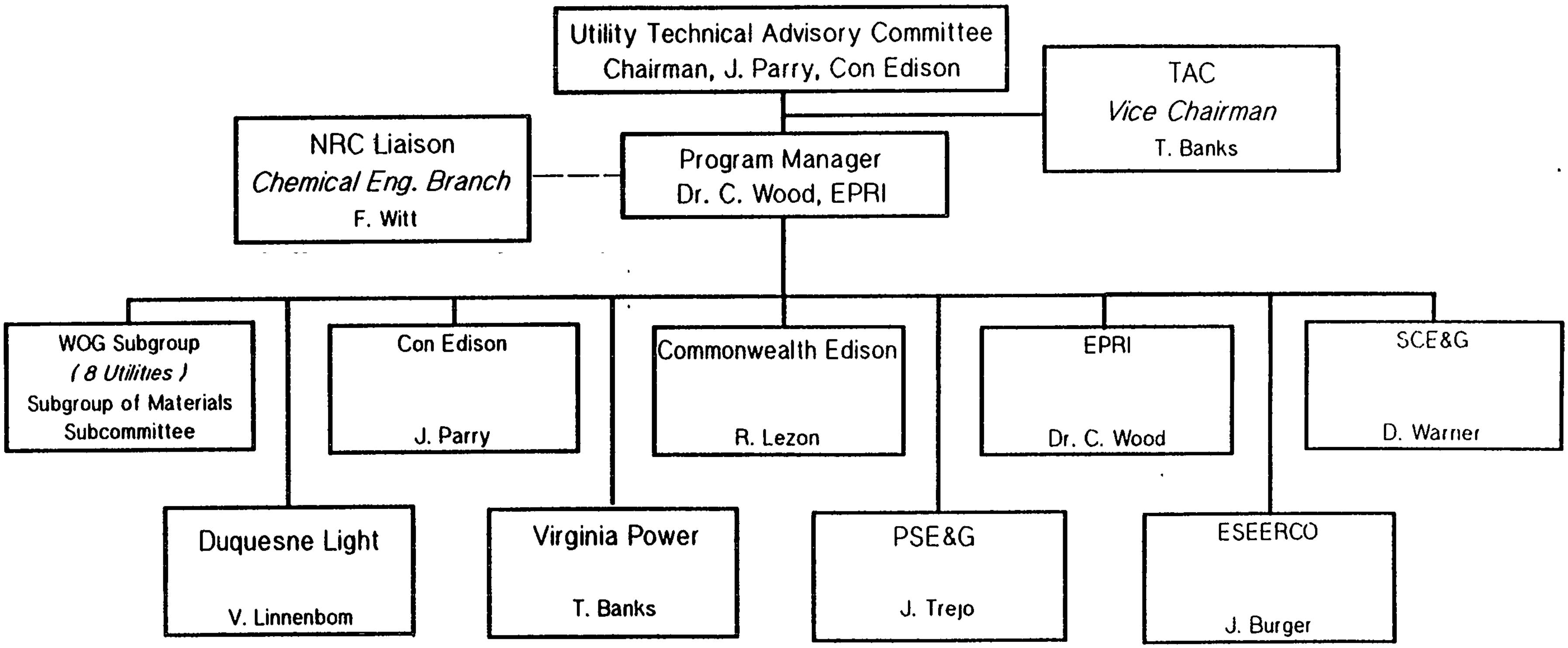
Component/ Subsystem/ System	% of RCS Surface Area (Ft^2)	Range of Potential Exposure Savings ★ (Man-Rem)	Range of Cost To Implement ($K)
RHX	0.1%	10 - 30	250 - 350
RTD Bypass	-0.1%	30 - 50	200 - 300
CVCS	0.1%	45 - 60	350 - 500
RHR	1.5%	65 - 80	350 - 700
S/G Channel Heads	2%	200 - 300	750 - 1000
Primary Loop	3.6%	500 - 900	1000 - 1500
Full RCS System	100%	1500 - 2500	10M - 20M

★ *Assumed over next 5 operating cycles*

Program Highlights

- Three-phrase program
- Phase 1 completed in 1988
- Phase 2 completed in January 1991
  - Funded by ten utilities, WOG, ESEERCO and EPRI
  - Included seven discrete work tasks
- Program assumed removal of nuclear fuel

FULL RCS
CHEMICAL DECONTAMINATION PROGRAM
TECHNICAL ADVISORY COMMITTEE ORGANIZATION CHART



NOTE: (1) EPRI REPRESENTATIVE (C. WOOD) MAINTAINS REQUIRED INTERFACES WITH COMPANION PROGRAMS WITH CEOG AND B&WOG AND ALSO TECHNOLOGY TRANSFER WITH BWR PROGRAM

Technical Highlights

- Processes considered were AP CAN-DEREM and AP LOMI
- Process qualification for three applications
- No prefilming of specimens
- Particulate testing completed
- Wear and friction tests conducted for both processes
- Off-normal tests conducted for both processes
- Program included CE specimens
- Processes qualified at boron concentrations of 0-650 ppm

Full Reactor Coolant System Chemical Decontamination Program

Phase 2 Decon Process Qualification and Detailed Engineering Evaluation

Task 1 Material Corrosion and Compatability Test Program

Task 2 Fluid Systems Evaluations

Task 3 RCS Equipment Evaluations

Task 4 Waste Management

Task 5 Radiological Evaluations

Task 6 Nuclear Safety Evaluations

Task 7
- Develop Decon Methodology
- Cost and Schedule Estimates
- Decon Equipment Conceptual Design

Full RCS Chemical Decontamination

Phase 1 Conclusions

- Technically feasible
- Perform first full system decon without fuel in the reactor vessel
- Reactor coolant pumps would provide temperature and flow
- Pressure controlled by nitrogen blanket in pressurizer
- Need cleanup flow of 1000-1500 gpm
- Expected exposure savings up to 2500 man-rem

Full Reactor Coolant System Chemical Decontamination

Results of Engineering Evaluations

- Qualification Program results extremely positive
- Minimal effects of chemical reagents on most primary system materials and components
- Certain pre- and post-decon inspections and modifications recommended

Full RCS Chemical Decontamination

Recommended Pre- and Post-Decon Inspections and Modifications

- RCP Vibration Monitoring
- Control Rod Driveline Removal
- CRDM Latching Functionality Test
- System Pump and Valve Leak and Wear Inspections
- Instrumentation Removal
- RHR Mechanical Seal Inspection
- RCP Seal Replacement

Con Edison / EPRI National Demonstration

- Con Edison's Indian Point Unit 2 selected as National Demonstration Site
- National Demonstration is scheduled for 1995 outage
- Program to be conducted under EPRI tailored collaboration
- Significant efforts are underway by various firms in support of National Demonstration (e.g., Westinghouse, Pacific Nuclear, General Physics, etc.)

V.C. Summer Fuel Decontamination Qualification Program

Objective:

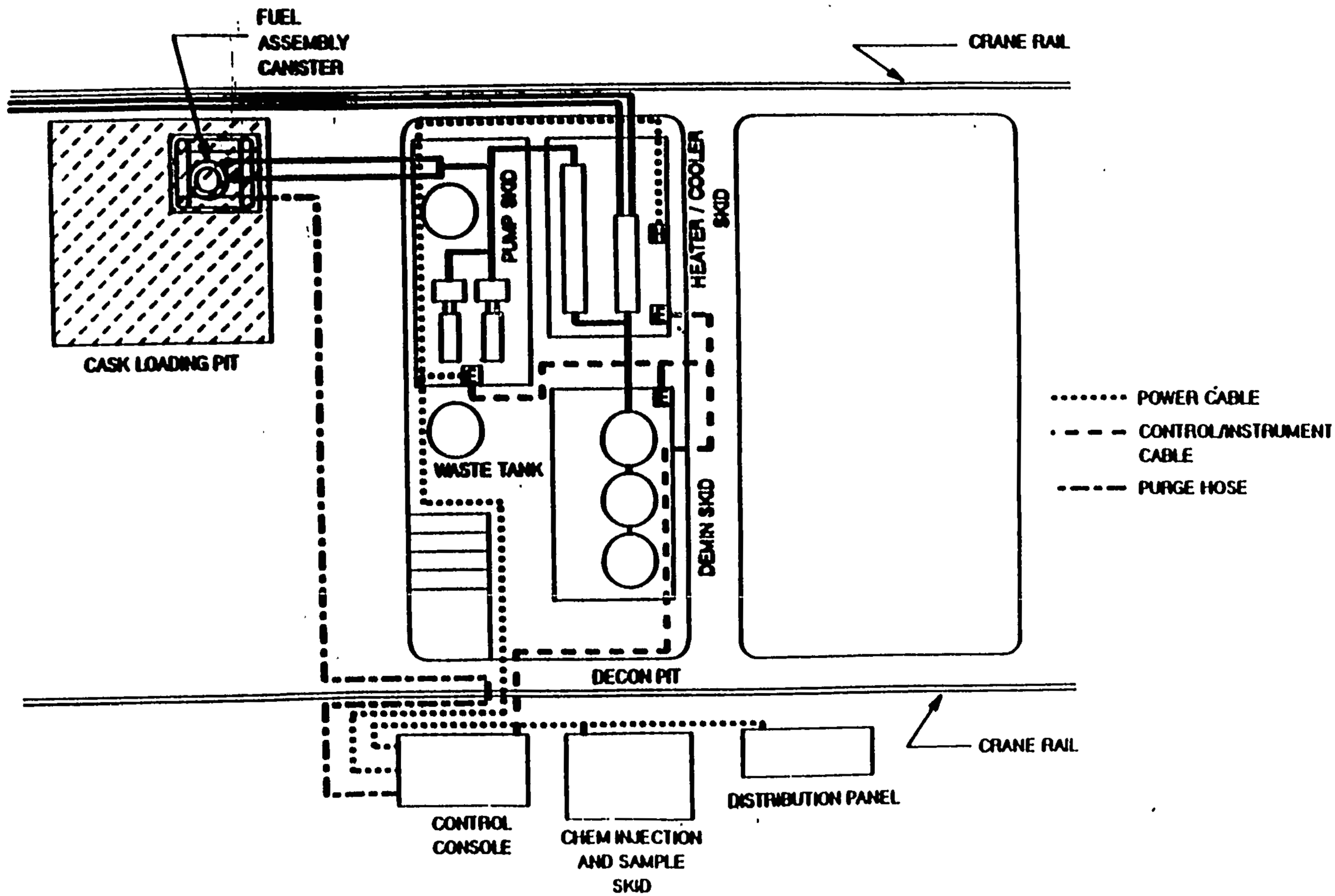
To perform a fuel decontamination qualification program at the V.C. Summer site which will qualify full RCS chemical decontamination with fuel in-place while maintaining existing warranties and incorporating improved fuel technologies.

Fuel Decontamination Qualification Program

Program Basis:

- General corrosion data for Zircaloy 4 and Zirlo was obtained from the full RCS decontamination program
- Fuel decon demonstration performed on four fuel assemblies
- Two fuel assemblies decontaminated using CAN-DEREM and two assemblies decontaminated using LOMI.
- Four deconned assemblies and two control assemblies reinserted in V.C. Summer plant for one more cycle with inspections at next outage.

V.C. SUMMER SITE FUEL DECON EQUIPMENT ARRANGEMENT



Fuel Examination Summary

- Visual Exams
  - Light to locally moderate crud deposition
  - All fuel assembly components in good condition
  - All crud removed by decon treatment
- Cladding Corrosion Exams
  - Zirconium oxide thickness typical for 2 cycle of exposure
  - No unusual corrosion conditions observed

Principal Findings from Fuel Decontamination Qualification Program

- Both processes were very effective in removing crud (approximately 20 curies of Co-58 and Co-60 per assembly)
- Corrosion rates on fuel materials were very low. No significant effect on ZrO_2 layer.
- Observed depletion of EDTA in CAN-DEREM reducing step
- Did not observe detectable quantities of CO_2 as free gas
- Successfully demonstrated that corrosion rates of key materials can be monitored on-line during decon
- Post-decon cladding corrosion behavior unaffected by decontamination processes

Principal Conclusions from Fuel Decontamination Qualification Program

- We were underestimating the activity level in-core based on prior visual crud deposition and sampling data
- Now have an improved database for estimating the activity levels and isotopic inventory in operating plants
- As a result of EDTA depletion, will need to recalculate estimated waste volumes for Full RCS Decon
- Control of boron within the limits specified was cumbersome and difficult to achieve. Additional study is required to enhance and control techniques
- Full plant fuel-in decontamination should not be limited by fuel cladding corrosion performance

Fuel-In Full RCS Chemical Decontamination Qualification Program

Objective of Program

To develop and complete a comprehensive program which will lead to Qualification of Fuel-in Full RCS Chemical Decontamination and NRC Approval of the Fuel-In Topical Report. The program will be based on qualification of both LOMI and CAN-DEREM and will utilize the existing extensive database for Fuel-Out Qualification.

Brief History of Fuel-In Qualification Program

- Successful completion of Full RCS Decon Qualification Program (Fuel Out) and V.C. Summer Fuel Decon Qualification Program leads into Fuel-In Qualification
- Benefits of Fuel-In are significant in terms of critical path time savings and additional man-rem exposure savings
- In 1993, Westinghouse developed the scope for the Fuel-In Qualification Program and conducted a seminar for interested utilities in July.
- Based on utility feedback, the program was divided into four phases with priority being given to those tasks essential to obtaining NRC Approval (i.e., Phase A).

Description of the Four Phases of the Fuel-in Qualification Program

Phase A

Work Scope Definition

- Work scope of Phase A includes only those tasks that are essential to achieving an approved Topical Report for the Fuel-In Option.

Major work areas include the following:

I. Topical Report Preparation and Review

- Fluid Systems Evaluations
- NSSS Equipment Evaluations
- Waste Characterization
- Radiological Evaluations
- Safety Evaluations
- Topical Report Preparation and Defense
- Topical Report Review by NRC

Phase A (Cont.)

II. Boron Control Study/LOMI and CAN-DEREM

III. Inquiries to Developers of LOMI and CAN-DEREM Processes

Optional Scope

Fracture Mechanics Analysis of CRDM Drive Road Coupling

Phase B

Work Scope Definition

- Work scope of Phase B includes the study of certain Process Application Anomalies which were noted during the two previous qualification programs. Some of these anomalies are controversial and may be "refined" during the inquiries of Phase A, Task III. Resolution of these issues are not important to development and approval of the Topical by NRC but are instead process effectiveness/process control non-safety related issues.

The Process Anomalies are as follows:

- Effect of Boron on Ion Exchange Resin Requirements
- Ion Exchange Resin Utilization
- Carbon Dioxide Generation
- Field Chemical Analyses
- Radiolytic Decomposition

Phase C

Work Scope Definition

- Work scope of Phase C includes the Plant Specific Evaluations related to an actual site-specific application of the Full RCS Chemical Decontamination with the Fuel-In

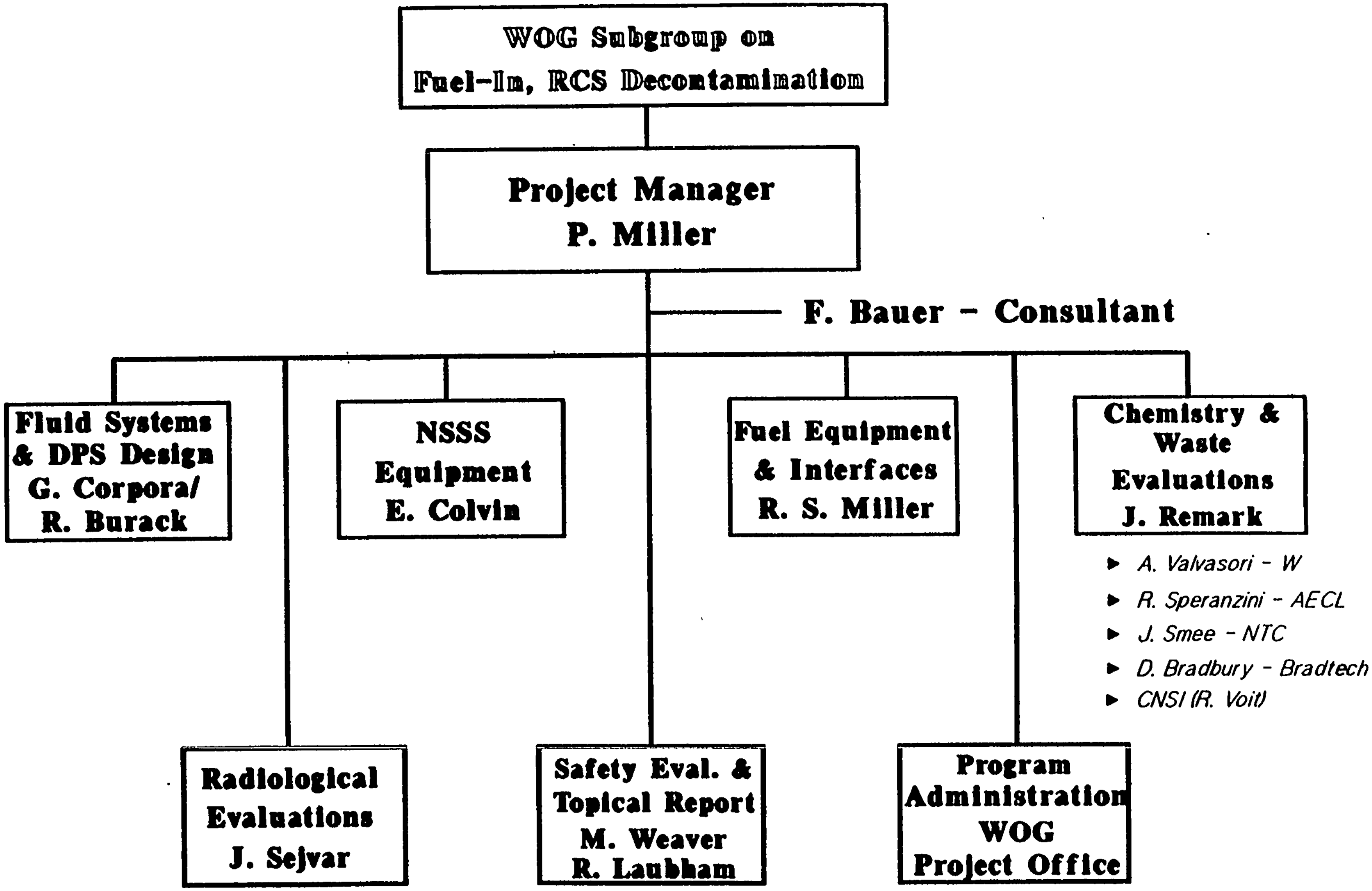
The Plant Specific Evaluations are as follows:

- Normal Operating Procedures
- Development of Check Off Lists
- Abnormal Operating Instructions
- Upgrade of Decon Process Equipment Design Basis

Phase D

Work Scope Definition

- Work scope of Phase D includes Waste Stabilization Certification and optional tasks as defined below:
  - Evaluation of Resin Requirements for LOMI
  - Evaluation of Resin Requirements for CAN-DEREM
  - Revise ORE Estimates for Fuel-In
  - Waste Stabilization Program
  - Resin Optimization
  - Vanadous Formate Oxidation


WOG Subgroup on Fuel-In, RCS Decontamination
Project Manager P. Miller
F. Bauer - Consultant
Fluid Systems & DPS Design G. Corpora/ R. Burack
NSSS Equipment E. Colvin
Fuel Equipment & Interfaces R. S. Miller
Chemistry & Waste Evaluations J. Remark
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R. Speranzini - AECL
J. Smee - NTC
D. Bradbury - Bradtech
CNSI (R. Voit)
Radiological Evaluations J. Sejvar
Safety Eval. & Topical Report M. Weaver R. Laubham
Program Administration WOG Project Office

Technical Evaluations to Extend Topical Report to Cover Fuel-In

Schedule in Months

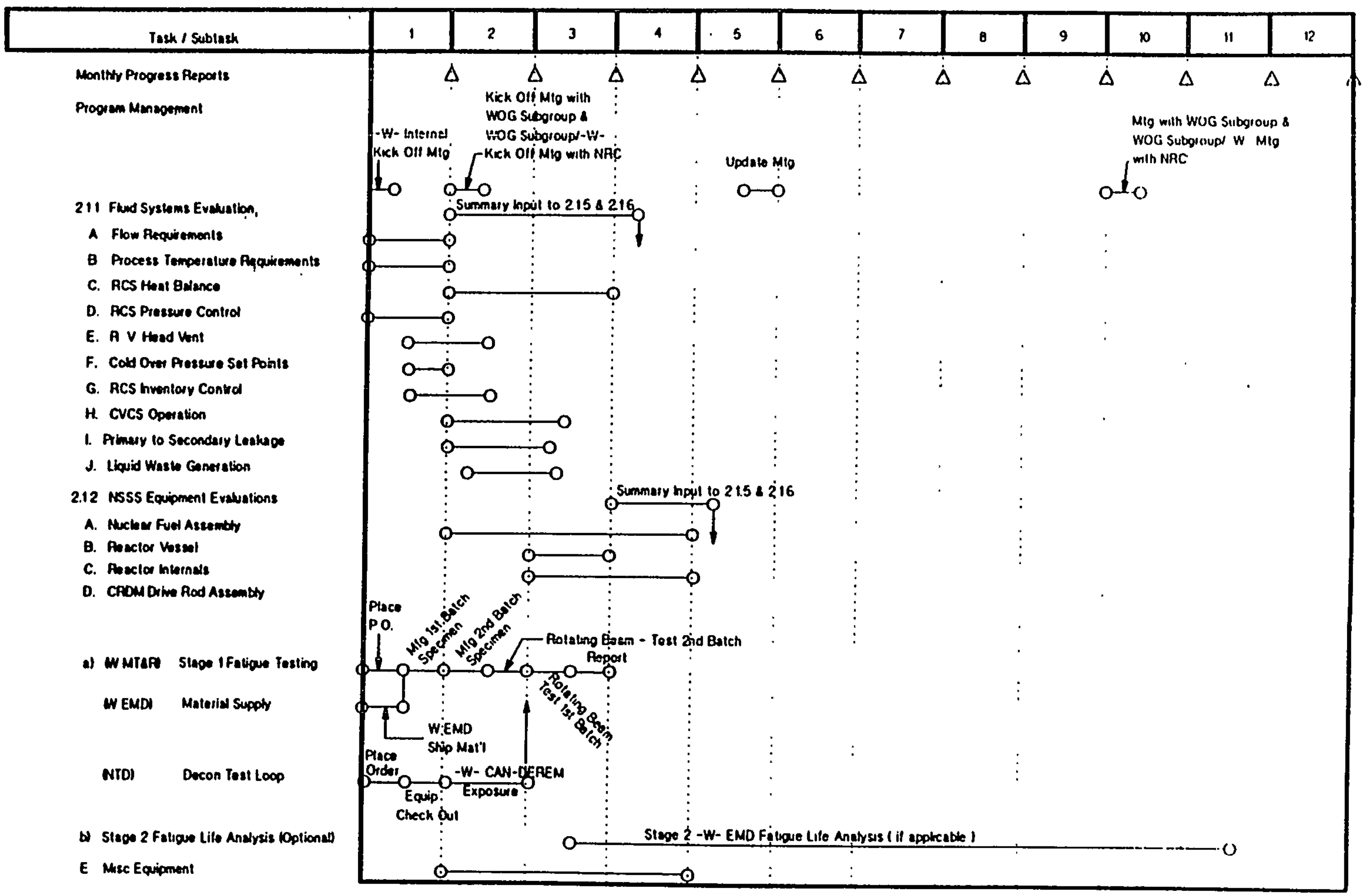

Figure 1 **Fuel In Full RCS Chemical Decontamination Program**
Phase A Program Schedule

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Page 1 of 2

Westinghouse Proprietary Class 2

Technical Evaluations to Extend Topical Report to Cover Fuel-In

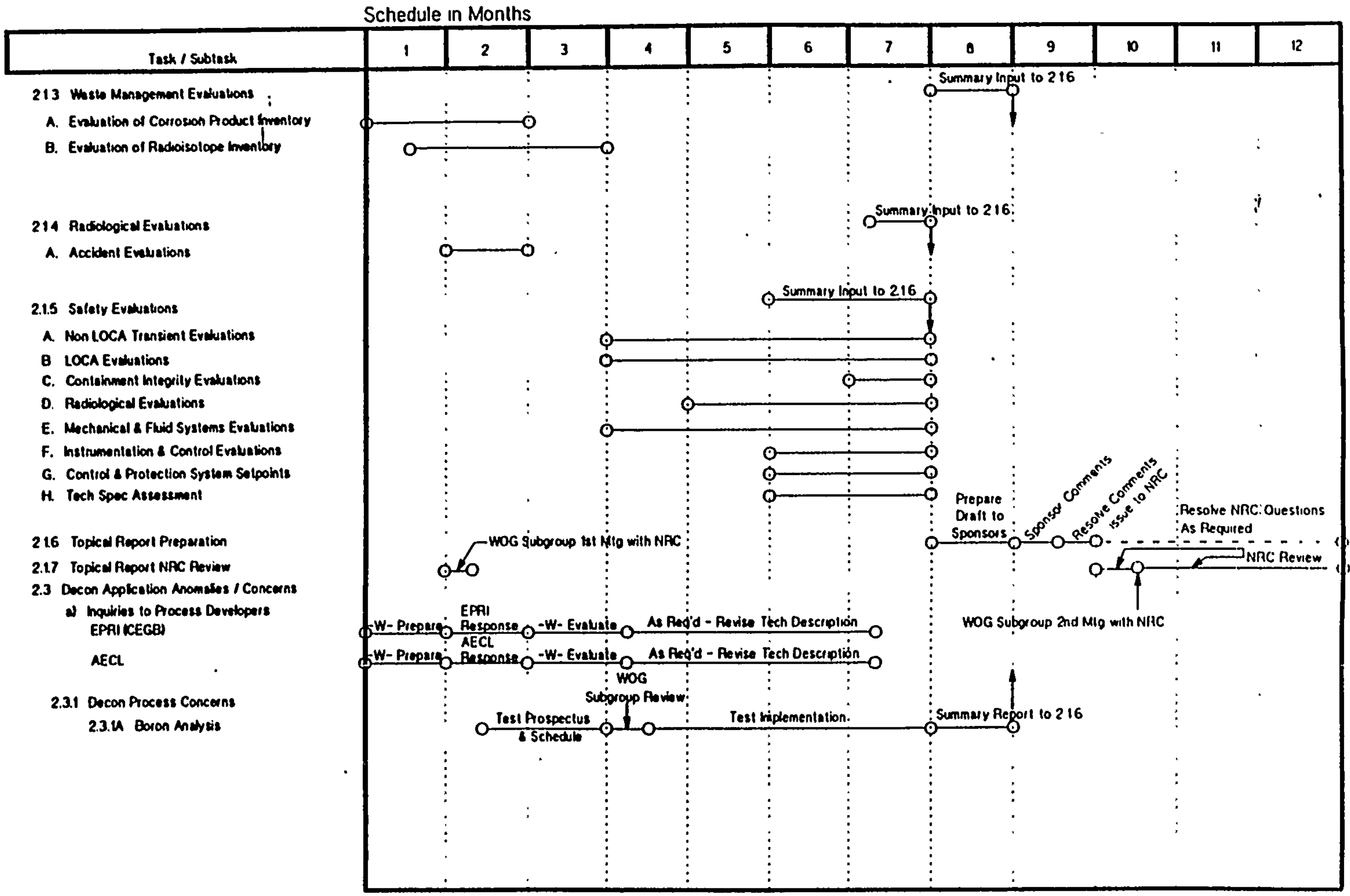

Figure 1 Fuel In Full RCS Chemical Decontamination Program
Phase A (Continued) Program Schedule

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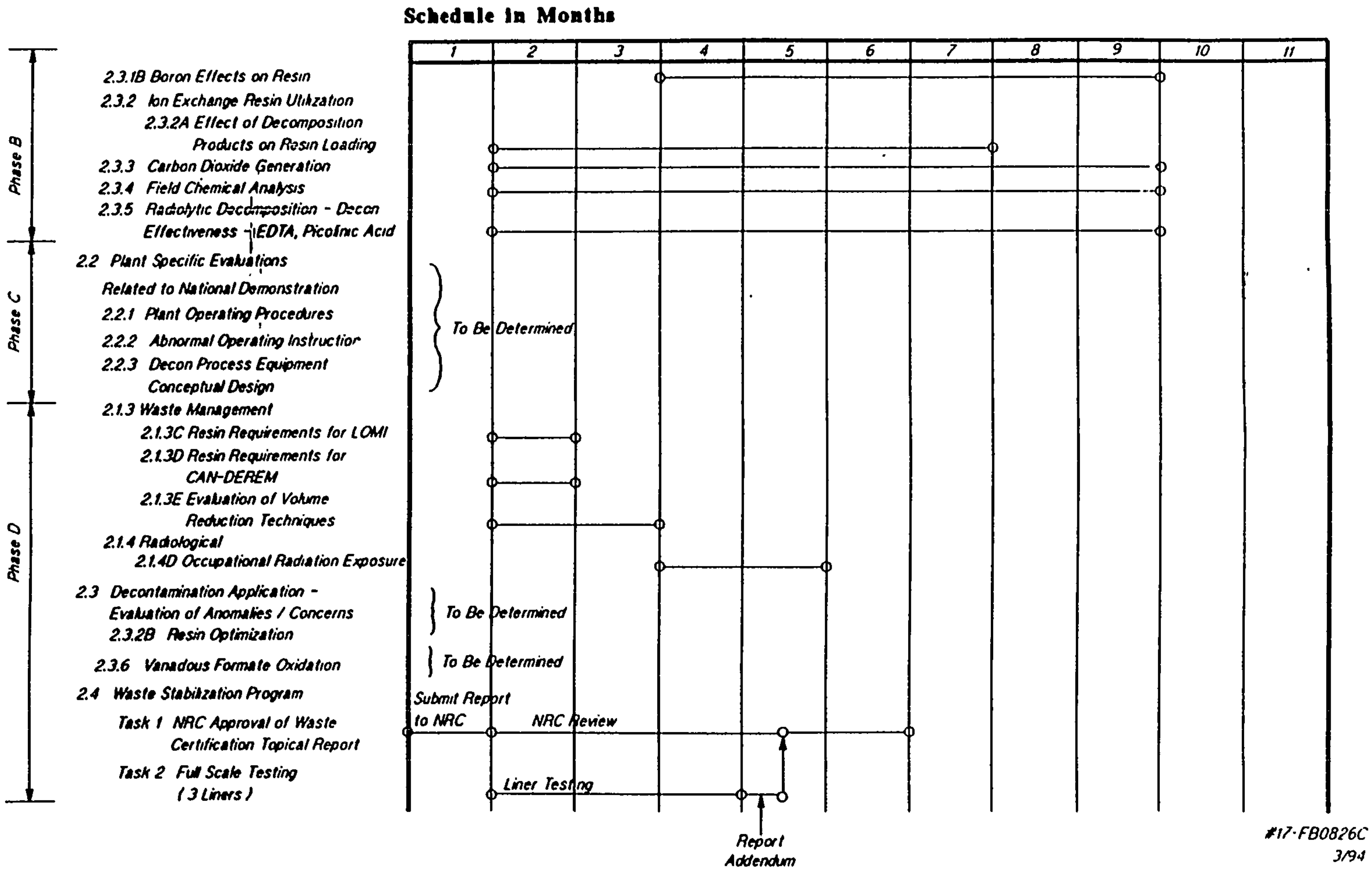
Page 2 of 2



Page 1 of 1

Figure 2 Fuel-In Full RCS Chemical Decontamination Program Phase B, C & D Program Schedule

NATIONAL DEMONSTRATION OF FULL REACTOR COOLANT SYSTEM (RCS) CHEMICAL DECONTAMINATION AT INDIAN POINT 2

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ABSTRACT

Key to the safe and efficient operation of the nation's civilian nuclear power plants is the performance of maintenance activities within regulations and guidelines for personnel radiation exposure. However, maintenance activities, often performed in areas of relatively high radiation fields, will increase as the nation's plants age. With the Nuclear Regulatory Commission (NRC) lowering the allowable radiation exposure to plant workers in 1994 and considering further reductions and regulations in the future, it is imperative that new techniques be developed and applied to reduce personnel exposure. Full primary system chemical decontamination technology offers the potential to be the single most effective method of maintaining workers' exposure "as low as reasonably achievable" (ALARA) while greatly reducing plant operation and maintenance (O&M) costs.

A three-phase program underway since 1987, has as its goal to demonstrate that full RCS decontamination is a viable technology to reduce general plant radiation levels without threatening the long term reliability and operability of a plant. This paper discusses research leading to and plans for a National Demonstration of Full RCS Chemical Decontamination at Indian Point 2 nuclear generating station in 1995.

BACKGROUND

The continued cost-effective operation of the 108 operating civilian nuclear power plants in the United States is an important part of our national energy strategy. To help implement this strategy, nuclear plant owners will need to reduce personnel exposure to radiation to the lowest level possible. In so doing, plant owners will improve the productivity of their work force and lower O&M costs for their plants. Thus far, the industry's aggressive radiation management programs have kept personnel exposure levels well within safe limits and, until recently, steadily reduced the average total exposure per plant, but, as nuclear plants age, maintenance and major equipment replacements can increase and contribute to increased collective exposure. Also, new governmental regulations have been issued that will reduce the allowable limits for an individual worker's annual exposure. The combination of the two factors could impair the cost-effectiveness of nuclear energy. Under these anticipated changes to the nuclear energy market, new methods to reduce personnel radiation exposure will likely be needed to maintain nuclear power as a commercially competitive option for the nation's current and future energy supply.

To deal with the twin problems of lower limits and increasing demand for personnel presence in radiation areas, the sources of radiation that contribute to personnel exposure must be reduced to the greatest extent practicable. While many approaches and technologies are being developed and implemented to address the problem, the decontamination of the entire primary system of a nuclear plant is the only method that offers the potential to reduce industry radiation exposure by an order of magnitude. Since 1975, Con Edison has been a leader in radiation management research and has had as a long-term goal the decontamination of the entire primary system at Indian Point 2, a pressurized water reactor (PWR) located in Buchanan, New York.

SOURCES OF PLANT RADIATION

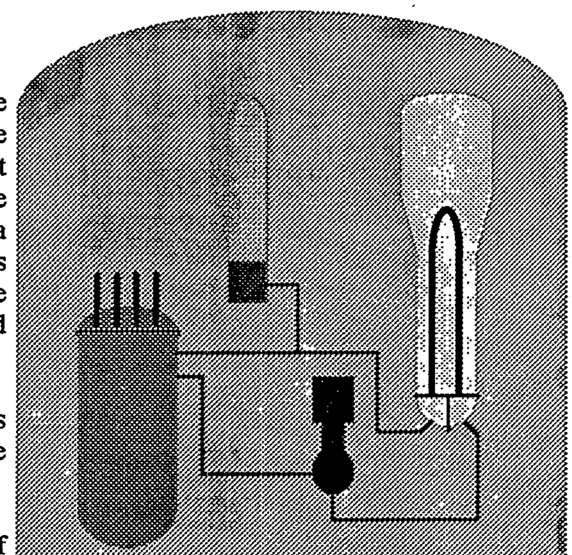


Figure 1. PWR Reactor Coolant System

Corrosion and wear products are found throughout the primary system of any nuclear power plant. In a PWR, the primary system is a separate loop called the reactor coolant system (RCS) (see Figure 1). These products circulate with the primary coolant, water, through the reactor vessel, where a small fraction become radioactive. A variety of radioisotopes are formed in this manner, but, for the most part, they are removed by filtration and demineralization in the chemical and volume control system (CVCS).

An oxide layer containing these activated products does form, however, on the surfaces of the RCS, including the fuel elements, CVCS, and other primary support systems.

The oxide layer of a PWR is a black spinel of primarily iron, chromium and nickel, formed in a slightly reducing chemistry with a pH less than 7.0. An analysis of the Indian Point 2 oxide layer shows that its composition is essentially:

$$(Ni_{1-x}\,Fe_x)\ (Cr_{1-y}\,Fe_y)_2O_4$$

with roughly 50% iron, 30% chromium and 20% nickel, although these may vary. Many radioisotopes of elements such as cobalt, manganese, zinc and antimony replace the iron, nickel and chrome, or otherwise become trapped, in the oxide layer matrix in small quantities. Two of the radioisotopes of cobalt (58 and 60) are typically the main contributors to the radiation fields in a PWR nuclear plant. The amount of radioactive material deposited on the different surfaces varies and depends primarily on the corrosion rate of the various plant materials, the chemistry of the primary water used as coolant, and the number of sources of cobalt. In a PWR plant, the oxide layer is rather tenacious, thereby making it difficult to remove the trapped radioisotopes from plant systems. As can be seen from table 1, it takes only a small degree of corrosion of base plant materials to create a significant radiation source on plant systems for long periods of time.

Table 1. Effect of Small Amounts of Radioactive Material

Average Cobalt in PWR	
Steam Generator Tubes	0.040%
1 Ounce	28.4 Grams
1 Gram of Cobalt 60	1,100 Curies
1 Curie of Cobalt 60	11 Rem/Hour at 3 Feet
Half-Life of Cobalt 60	5.2 Years
Personnel Exposure Limit	5 Rem/Year

As maintenance is performed on plant systems, personnel are exposed to radiation, primarily gamma, emitted from the radioactive material in the oxide layer. Radiation fields from the oxide layer increase with time from initial plant start up and level off after several years of plant operation.

Personnel exposure to radiation is highest during refueling outages, when routine maintenance is performed on such major plant components as steam generators and reactor coolant pumps. In fact, radiation fields from the steam generators are usually the largest single contributor to PWR personnel radiation exposure. The reason for this is that steam generators can require extensive maintenance and inspection, and the radiation fields inside the steam generator where this maintenance must be performed can be as high as 40 rem per hour.

DECONTAMINATION RESEARCH HISTORY

Because the PWR oxide layer is tenacious, mechanical methods are relatively ineffective at removing the oxide and trapped radioisotopes. Research has therefore focused on chemical decontamination processes that are effective at dissolving the oxide layer. When the oxide layer is dissolved, the radioisotopes are again returned to solution in the reactor coolant, where they can now be removed by filtration and demineralization. In the mid-1970s, with the support of the Department of Energy, Con Edison began research on ways to remove the radioactive material by dissolving the cobalt-containing primary system oxide layer.[1] This research effort is important from a historical basis in that it had a different objective than much of the chemical decontamination research, done prior to that time, which had focused on decommissioning. Unlike decontamination methods for decommissioning, where the post-decontamination plant equipment condition is a relatively minor concern, methods for operating reactors must not impact the life of the decontaminated equipment. The project identified some of the more important issues of PWR decontamination, while a the same time laying some of the ground work for future research programs. The issues identified include:

- The first plan of how to perform a chemical decontamination of a PWR primary system and be able to restart the plant,
- A comparison of various chemical techniques to perform an effective decontamination, and
- A screening study to conduct some limited material/chemical tests.

Concentrated chemical solvents, typically 1-10% in concentration, referred to as "hard" decontaminations, are often used for decommissioning. This early work led to a focus by researchers on dilute chemical solvents typically less than 0.5% in concentration, referred to as "soft" decontaminations, for operating reactors. Soft decontamination solvents are weak acids such as citric acid.

During the late 1970s and early 1980s, Con Edison and many organizations, including the Empire State Electric Energy Research Corp. (ESEERCO), conducted extensive research programs to investigate the compatibility of dilute chemical solvents with the materials used in nuclear plant systems. Con Edison's and ESEERCO's efforts[2,3] centered around the AP/Can-Derem[a] process, a modified version of a process originally developed by Atomic Energy of Canada, Limited (AECL) of Chalk River, Canada, for the heavy water CANDU reactors. The process, which will be used for the first full primary system decontamination, has been used many times for component decontaminations, which are characterized by the small volume of equipment to be cleaned. The reactor coolant system of a PWR is much larger in volume and scope than the typical component decontamination. The AP/Can-Derem process was selected because it has several advantages over other decontamination processes in that it produces less waste, can be easily controlled on a system as large as a reactor coolant system, is benign with regard to material corrosion and yet achieves high decontamination factors. Decontamination factor, the ratio of radiation fields before to after a decontamination, is the measure used in the industry to assess results. At the present time Vectra Technologies, Richland, Washington and Westinghouse Electric Corporation, Pittsburgh, Pennsylvania are two vendors licensed in the United States by AECL to provide the AP/Can-Derem process.

AP/CAN-DEREM DILUTE CHEMICAL SOLVENT PROCESS

Dilute acids are used in most decontamination processes applied in the civilian reactor industry to dissolve

[a] Can-Derem is a trademark of AECL

the oxide layer. For example, a simple one-step acid dissolution approach is used for the CANDU reactors and U.S. boiling water reactors (BWR). Since a PWR oxide layer contains a high chromium content, dilute chemical solvent processes for such plants must contain two diverse steps. The high chromium content (20-40% Cr as Cr^{+3}) found in a PWR oxide layer acts as a barrier to dissolution of the iron and nickel. An oxidizing step is therefore used to dissolve the chromium portion of the layer. While the following is a simplification of the complex reactions associated with such processes, the basic steps are illustrated.

The Alkaline Permanganate (AP) oxidizing step (see Figure 2), often referred to as a pretreatment, is used to convert insoluble Cr^{+3} in the PWR oxide layer to its soluble valence state, Cr^{+6}. The Can-Derem dissolution step (see Figure 3) is then applied to dissolve the iron and nickel oxide portion and release the majority of the radioisototpes to solution. The reaction steps can be expressed as:

Oxidation step - chromium oxide plus an oxidizing agent form soluble chromium in the form of chromate, typically:

$$Cr_2O_3 + 2MnO_4^- + 2OH^- \dashrightarrow 2CrO_4^{-2} + 2MnO_2 + H_2O$$

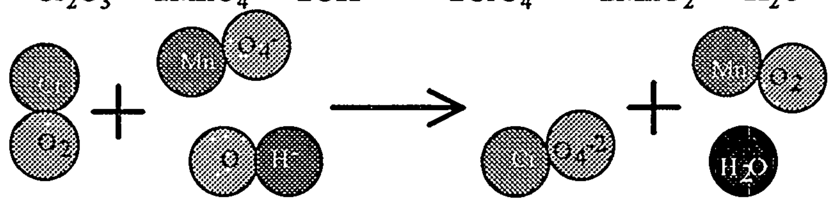


Figure 2. Alkaline Permanganate (AP)

Dissolution step - iron oxide plus a dilute acid form soluble iron and water, typically:

$$Fe_3O_4 + 8H^+ \dashrightarrow Fe^{+2} + 2Fe^{+3} + 4H_2O$$

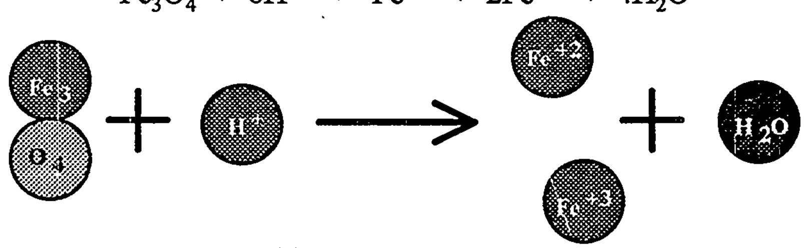


Figure 3. Can-Derem

The number of steps and their sequence of application depends on the composition of the deposit layer, the concentration of the chemicals, and the relative effectiveness of each step.

When applied to a plant component or system, the AP/Can-Derem is a multi-step dilute chemical decontamination process with alternating oxidizing and dissolution steps. Five steps are commonly used in PWR applications to dissolve alternately the chromium and iron/nickel. By alternating steps, the most effective dissolution of the oxide layer can be achieved in the shortest amount of time. For the oxidizing step, Alkaline Permanganate (AP), the process is a reagent solution of dilute (0.1%) potassium permanganate ($KMnO_4$) with 0.01% sodium hydroxide (NaOH) to adjust the pH to between 10 and 11.

AP pretreatment is very effective on inconel surfaces. Reagent additions are used to maintain the AP concentration and pH as the chemicals are circulated through the plant systems. No significant activity or oxide removal occurs during the AP pretreatment. At the conclusion of the pretreatment steps oxalic acid is added to decompose the remaining permanganate and acidify the solution prior to the dissolution step. Virtually no radioisotopes are released during the AP step clean-up.

The dissolution step is a 0.1% solution of Can-Derem, an organic acid/chelant mixture composed of citric acid and ethylenediaminetetraacetic acid (EDTA), at a resultant pH of 2.3-3. The dissolved oxygen concentration of the solution (reactor coolant for a full RCS decontamination) is adjusted to less than 0.2 ppm by the addition of hydrazine (N_2H_2) prior to the introduction of the Can-Derem reagent.

Table 2 is a set of conditions and steps shown to be a good sequence for effectively obtaining decontamination factors in excess of 5 on 304SS, Inconel 600 and other PWR primary system materials:

Table 2. Optimized AP/Can-Derem Process

Sequence: Can-Derem, AP, Can-Derem, AP, Can-Derem

Step	Reagent	Concentration	Time	Temperature
1:	Can-Derem	0.1%	24 hours	120C
2:	AP	0.1%	12 hours	95C
3:	Can-Derem	0.1%	24 hours	120C
4:	AP	0.1%	12 hours	95C
5:	Can- Derem	0.1%	24 hours	120C

In the AP/Can-Derem process, the acidic solution removes the vast majority of the surface oxides and radionuclides which are then subsequently captured by cation exchange resin. As the cation resin captures the dissolved metals, it regenerates the decontamination solution for reuse. Each Can-Derem step may be continued as long as contaminants are still being removed. Decontamination is terminated by isolating the cation demineralizers and using mixed cation and anion resin for system cleanup. The anion resin removes the chemical reagents themselves, and the cation resin removes any remaining dissolved metals.

The primary advantages of AP/Can-Derem with regard to ease of field application are its online process control, chemical stability, and regenerative nature, which reduces waste generation.

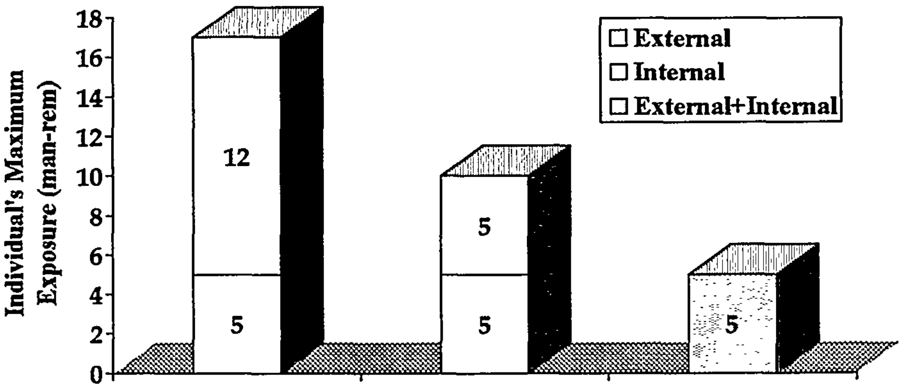
Laboratory research and more than a hundred decontaminations of components and systems in the field have shown that a variety of components and systems can be decontaminated with AP/Can-Derem without adverse impact on the long-term reliability of the equipment. Con Edison has already performed decontaminations at Indian Point 2 of its regenerative heat exchanger in 1989, CVCS in 1991, residual heat removal system (RHRS) in 1993 and retired Unit 1 steam generators in 1992, all with the AP/Can-Derem process.

NEED FOR FULL PRIMARY SYSTEM DECONTAMINATION

The International Committee on Radiation Protection and the National Committee on Radiation Protection have recommended personnel radiation exposure limits that are, on average, less than half current limits (see Figure 4). These recommendations will affect nuclear plant owners in several ways. Higher radiation exposure and lower limits on exposure will likely make it more difficult to obtain skilled labor as workers expend their yearly allotments. This would be particularly likely to arise if a refueling outage were to occur late in the calendar year, when annual radiation exposure allotments have been depleted. Such a situation would leave utilities with fewer resources to respond to unanticipated maintenance requirements.

Along with lower exposure limits, federal regulations were revised to make the ALARA concept law. The concept means that personnel exposure to radiation must be kept "as low as reasonably achievable." This will

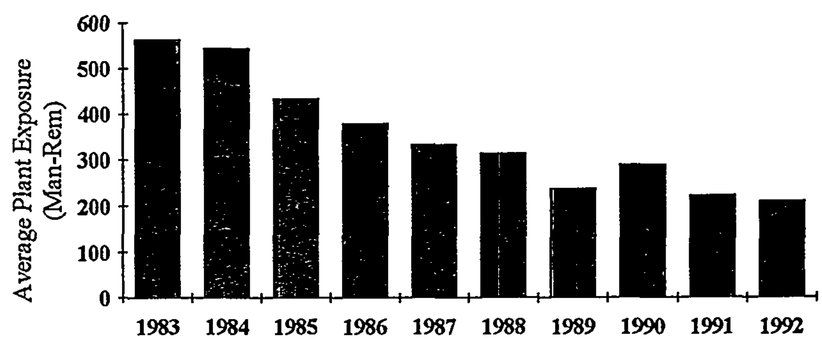
Figure 4. Exposure Limit Trend



enable the Nuclear Regulatory Commission (NRC) to set strict rules to ensure that utilities make personnel exposures ALARA.. Figure 5 shows that over the last few years average exposure for s (PWRs) has leveled off. This is because the prospects for new exposure reducing technology are limited.

Additionally, radiation exposure contributes to lost productivity. More and larger crews are needed and work is less productive in a radioactive environment. The time that crews can stay in the radioactive environment is shorter and set-up and clean-up requirements for manpower and materials are much higher than they would be for the same work in a clean environment. A study performed in 1992 indicated that a 10-50% increase in operation and maintenance costs will be incurred by utilities due to the above factors.[4]

Figure 5. PWR Exposure Trend



While new radiation field reduction techniques (such as high pH chemistry and specifying low cobalt alloys for replacement parts) have helped, a more substantial reduction in radiation fields is needed to help nuclear plant owners comply with new regulatory requirements, enhance worker productivity, and help keep nuclear power operating costs competitive. Full primary system decontamination is the only new technology with the potential to meet this need for a large industry-wide improvement.

The step from component to full primary system decontamination is, however, technologically a large one. The systems involved, the technical issues to be addressed, and the logistics for performing such a large-scale chemical decontamination are very complex.

QUALIFYING THE PROCESS

In 1987, Con Edison spearheaded a three phase program whose goal was demonstrating for the first time chemical decontamination on the entire primary system of a U.S. PWR plant. The first two phases, Feasibility and Qualification, are now complete. Phase 3, Demonstration, is in the engineering and fabrication stages.

ESEERCO, ten other utilities that own PWRs, and the Electric Power Research Institute have participated in the program thus far, along with Westinghouse and other decontamination service vendors.

The objective of the Feasibility phase was to determine the technical acceptability of dilute chemical solvent processes for primary system decontamination of a typical PWR plant. Studies of the Westinghouse PWR primary systems were conducted to establish the conditions, parameters, and criteria for a test program to qualify the solvents for the full RCS.[5] One important finding was that existing PWR plant systems can be used to control the solvent process on the full primary system. A temporary decontamination support system will be needed on site to feed the chemicals into the plant and receive and process the decontamination waste stream. Another important finding is that full primary system decontamination without fuel is highly cost beneficial, although leaving the fuel in place in the reactor would provide some additional savings. For the first demonstration of full primary system decontamination, fuel will be removed from the reactor vessel. Ultimately, once experience has been gained with full primary system decontaminations, fuel will likely be included. Extensive surveys were also conducted to establish a complete list of materials that would be exposed to the chemical solvents, and a series of tests was designed to address all possible issues for each material at the flow and chemistry conditions expected in the RCS.

In the Qualification phase, more than 250 specimens of more than 80 different materials were tested. Tests were also performed with the chemical concentrations and temperatures slightly increased. This results in a more corrosive test and demonstrates that the process could still be applied with a margin of safety under hypothetical fault conditions.

The tests were designed to establish a technical basis for performing at least three applications of the process during the remaining life of a plant. The tests were done in two test loops constructed at Westinghouse laboratories in Churchill, Pennsylvania. Engineers at Westinghouse, the original supplier of the reactor system at Indian Point 2, evaluated the test results. These evaluations were compiled, recommendations were made for application in the field, and a safety assessment was performed. Westinghouse prepared reports on the successful qualification for submittal to the NRC.[6,7] The NRC has reviewed the reports and issued a letter authorizing their use within an approved framework.

THE NATIONAL DEMONSTRATION

A joint effort is needed to assure that the broadest benefit is achieved by the entire industry from the first full RCS decontamination of an operating commercial nuclear power plant in the United States. An R&D consortium of utilities and other organizations including EPRI and ESEERCO has been organized to sponsor a national demonstration of the technology at Con Edison's Indian Point 2 in 1995. Participants will gain direct experience through a technical advisory group and will be provided with detailed information such as process and material test results, specifications and safety evaluations. This information and the experience gained should lower substantially the cost of their own subsequent decontaminations. The cooperative effort will enhance technology transfer by assuring that input is received from experts throughout the industry and by promoting exchange and publication of the results. For more information about the program or the technical advisory group contact Stephen Trovato, (212) 460-2090, or John Parry, (914) 526-5038, of Con Edison.

The National Demonstration is currently on schedule for Indian Point 2 in February 1995. Vectra Technologies (formerly Pacific Nuclear) has been selected to conduct the decontontamination. A temporary process system has been designed that will have minimal impact on the existing plant. This majority of this system will be located in a small room of the Indian Point 2 primary auxiliary building (PAB) normal used for temporary waste storage. The process system will be controlled remotely from outside the PAB with communication to the plant central control room for coordination with plant operation. The PAB contains much of the equipment for the RCS support systems, such as the RHRS pumps and heat exchangers. The proximity of the room selected for placement of the temporary process system to the RHRS facilitates the plant to process system tie-in. Decontamination chemicals, clean demineralizer resins and clean water for sluicing resins will be supplied from equipment placed outside the PAB. Demineralizers will be placed inside the PAB to provide adequate shielding of removed radioactive material. Feed of chemicals to the primary system and removal of system water for cleanup will be through the RHRS. From the RHRS the solvents travel through the RCS and the CVCS. While only one reactor coolant pump is required to circulate the solvent, up to three may be operated to provide an even decontamination of the reactor coolant loops. The reactor coolant pumps generate more than enough heat to maintain the temperature required for the solvents to work. The RHRS heat exchangers, which provide cooling, are used to balance the primary system temperature. Once the decontamination has been completed the spent resins are sluiced to high integrity containers, and dewatered for eventual storage or burial.

Major components of the decontamination process system were procured in 1993 and fabrication and assembly are nearing completion. A five week factory acceptance test of the process system is scheduled for Spring 1994. The decontamination process system operating procedure has been drafted and plant operating procedures have been written. A test plan that defines the data to be captured, its form, frequency and accuracy has been developed. Design work on the low level waste cask storage modules has also been completed. Numerous supporting analyses were conducted to support the operation of the plant in "decon mode" and the preparation of a safety evaluation including:

- Required reactor vessel head closure studs
- Off-site radiation exposure levels
- Major component cladding cracks (if existing)

Plant system walk-downs have been completed and plant modification packages prepared. A program was put in place to identify all potential dead legs and methods to flush them after the decontamination. Outage task planning is now in progress.

To prepare for the demonstration in 1995, Con Edison performed many activities during the 1993 refueling outage of Indian Point 2. For example, the RHRS was modified to provide the connection points for the temporary decontamination support system. Another change was to test a new low pressure seal, to be used when the reactor in-core instrumentation is pulled out of the vessel, to upgrade the existing seals to the operating pressure anticipated during the decontamination. Several tests of plant equipment were conducted to very their ability to meet "decon mode" operating conditions including component cooling, charging flow and the reactor coolant pump seals. During the current plant operating cycle, other site preparations will be performed, such as clearing areas to accept the temporary decontamination support system, including a small room within the plant's primary auxiliary building that will hold the demineralizers.

BENEFITS TO BE GAINED

While benefits will vary depending on plant age, radiation fields, and maintenance and equipment-replacement activities, the broad benefit to the utility industry as a whole is estimated at a direct saving of roughly one billion dollars over the next 10 to 20 years. An estimated 3,500 man-rem could be saved at Indian Point 2 alone over a 5 cycle operating period.

There are many non-quantifiable benefits of the demonstration too. Full primary system chemical decontamination will provide the nuclear industry with a means to substantially reduce the collective radiation exposure of its workers in the future. As a direct result of lower radiation fields, worker productivity will improve thereby making nuclear energy more competitive. In the long term, full system decontamination should become a regularly applied technology. It will facilitate life extension of current plants because it will provide a way to maintain equipment without incurring large radiation exposure doses. For the next generation of light water reactors, routine full system decontaminations could help provide for sustainable lower maintenance costs. Finally, reducing the collective radiation exposure should improve the image of the nuclear industry at a time when public perception is an important factor affecting a utility's ability to extend the life of a nuclear power plant past the expiration of its original license.

In summary, full system decontamination should become a vital part of life extension and the future of nuclear energy in this country.

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Author Biography

Stephen Trovato is a Principal Mechanical Engineer with Con Edison's Research and Development Department and is responsible for the management of its Nuclear Research Program. Prior to joining Con Edison in 1984, Mr. Trovato worked as an Engineer with the Corporate Planning Department of the Long Island Lighting Company. Prior to that, he was with the Mechanical Engineering Deivision of the American Electric Power Service Corporation. Mr. Trovato has received his Masters Degrees in Mechanical Engineering from Cooper Union and Business Administration from St. John's University. He has been involved in chemical decontamination research for the last 8 years.

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RESULTS FROM THE DECONTAMINATION OF AND THE SHIELDING ARRANGEMENTS IN THE REACTOR PRESSURE VESSEL IN OSKARSHAMN 1 – 1994

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ABSTRACT

In September 1992 Oskarshamn 1 was shut down in order to carry out measures to correct discovered deficiencies in the emergency cooling systems. Due to the results of a comprehensive non destructive test programme it was decided to perform a major replacement of pipes in the primary systems including a full system decontamination using the Siemens´CORD process. The paper briefly presents the satisfying result of the decontamination performed in May-June 1993. When in late June 1993 cracks also were detected in the feed-water pipes situated inside the reactor pressure vessel (RPV) the plans were reconsidered and a large project was formed with the aim, in a first phase, to verify the integrity of the RPV. In order to make it possible to perform work manually inside the RPV special radiation protection measures had to be carried out. In January 1994 the lower region of the RPV was decontaminated, again using the CORD-process, followed by the installation of a special shielding construction in the RPV. The surprisingly good results of these efforts are also briefly described in the paper.

INTRODUCTION

Oskarshamn 1 is an ABB Atom BWR, 462 MW, in commercial operation since 1972. In September 1992 Oskarshamn 1, together with four other Swedish BWR´s of the same design, was shut down in order to carry out measures to correct discovered deficiencies in the emergency cooling systems. One of the measures was to replace most of the insulation in the containment with reflective metallic insulation. Taking the opportunity of having a great number of pipes free from insulation a comprehensive programme of non destructive testing was executed. As a result of these tests, cracks were detected in cold-bended pipes in the shut-down cooling systems. After an inventory of cold-bended pipes in the primary systems and further tests, additionally cracks were found. In April 1993 it was consequently decided to perform a major pipe replacement. In order to save dose to the personnel involved, a decontamination of the affected systems was performed during 4 days in May-June 1993. Siemens´ CORD-process, a Chemical Oxidation Reduction Decontamination using permanganic acid and oxalic acid, was used in 3 cycles with an excellent result. A total of 7 kg of metals (70 % Fe) with a total activity of 350 GBq (73 % Co60) were removed and an averaged decontamination factor of 17 was achieved. The collective dose saved has been estimated to 3.1 manSv. However, when in late June 1993, cracks also were found in four of the six feed-water pipes situated inside the reactor pressure vessel (RPV) the situation called for a reconsideration. A large project with the aim of verifying and upgrading Oskarshamn 1 was initiated. In Phase 1 of this project the main task was to test and verify the integrity of the RPV. Most of the tests in the RPV could be performed on distance under water. But the tests and the repair work in the lower parts had to be performed on dry surfaces. As it obviously, to some extent, was a need for manual work in the lower region of the RPV, special radiation protection measures had to be considered. A plan including decontamination- and shielding procedures was presented. Calculations showed that it might be possible to perform at least certain jobs on the spot.

DECONTAMINATION OF THE RPV

After studies and investigations it was decided - with approval from the authorities - to perform a decontamination using - again - the CORD-process. Though the very short time available for the preparing of the decontamination it was carried out on the scheduled time in January 1994. The fuel and all the RPV-internals were removed from the RPV. To be able to store the RPV-internals under water in the reactor pool an extension was connected to the RPV.

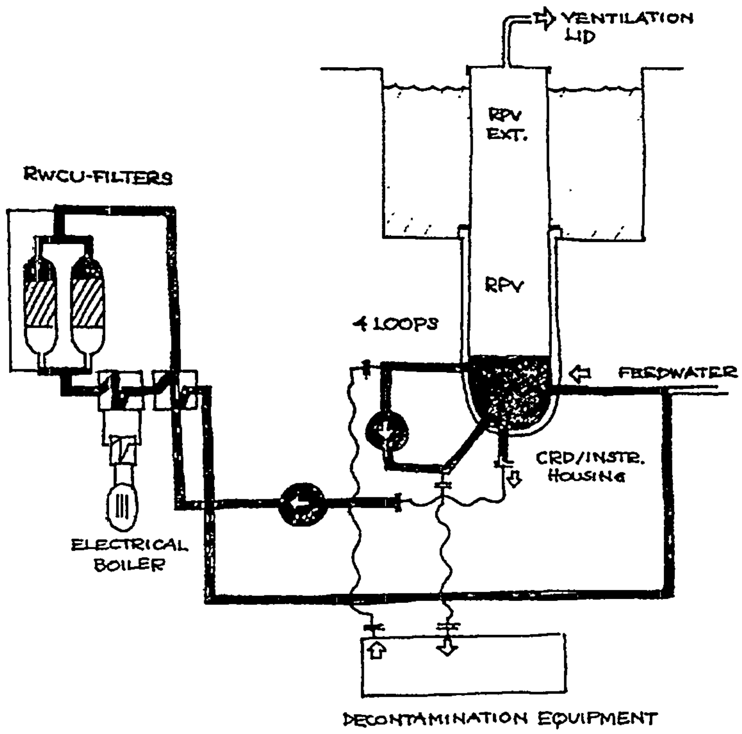

Fig 1. Schematic view of the decontaminated systems (in black)

The RPV was filled with decontamination liquid up to a level 1 m up in the core region. The RPV was covered by a ventilated lid. The main recirculation pumps were used to circulate the decontamination liquids in the RPV and they were heated by means of an existing 5 MW electrical boiler. The ion exchange columns in the reactor water clean-up system were used for taking care of the dissolved cations and activity. A total area of 1 350 m^2 was treated of which approximately 900 m^2 already had been decontaminated in June 1993. The total volume of the affected systems was 156 m^3. The decontamination was performed in four cycles, approximately 30 hours each. In all the operation took 8 days. The result of the decontamination was excellent: 27 kg metals (62 % Fe) and 2,3 TBq (93 % Co60) were removed. According to calculations it corresponded to a removal of 99,88 % of the activity. The waste volume produced was 3,7 m^3 ion exchange resins. After the chemical decontamination the RPV was finally flushed with water at high pressure (< 500 bar).

THE SHIELDING ARRANGEMENTS IN THE RPV

The shielding equipment was then installed. This equipment consists of 5-10 cm thick circular iron walls shielding the inside of the RPV from the top to 2,5 m above the RPV-bottom and a 5 cm thick iron floor placed just above the control rod housing.

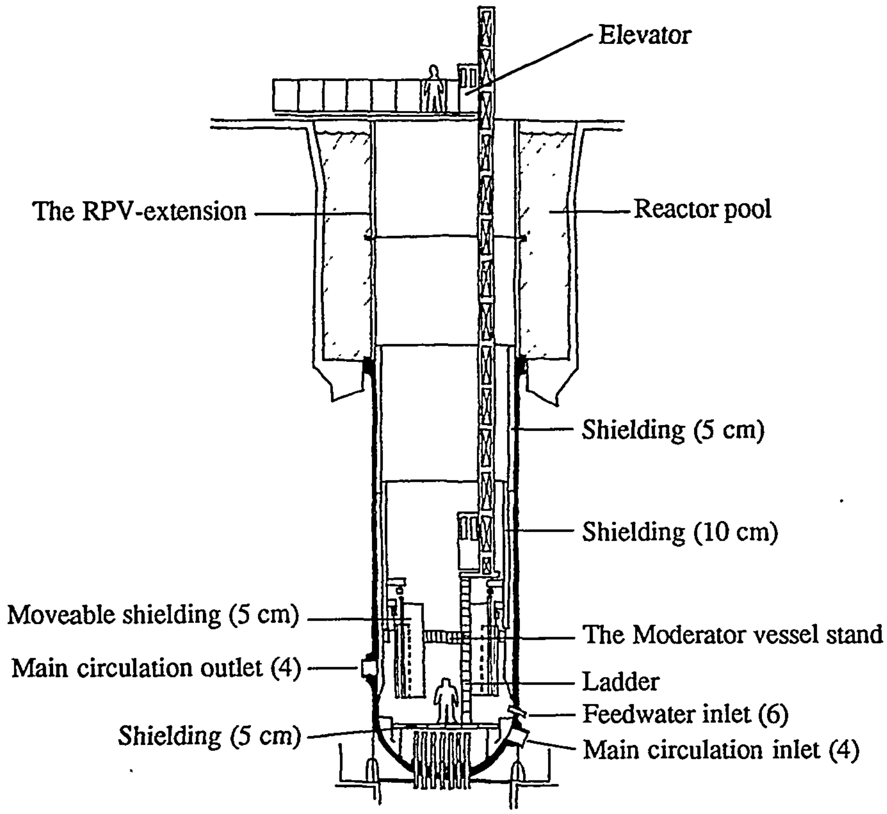

Fig 2. RPV-shielding arrangements

The surface doserate in the core region without shielding was 120 mSv/h and with the shielding 5 mSv/h. The doserate measured on the surface of the RPV underneath the shielding floor was 15-20 µSv/h which indicates a decontamination factor of approximately 200. The lower part of the shielding walls consists of moveable parts for the purpose of not having to remove more shielding than needed for different jobs. The average doserate at the centre of the working area with a 60 ° gap in the moveable shielding walls was measured to be 0,3 mSv/h and at a distance of 0,5 m in front of the gap 0,6 mSv/h. The moderator vessel stand is the main source of radiation in the working area and was therefore specially shielded. The surface doserate on that shielding was 3,5 mSv/h. But we think we are able to take additional measures in order to reduce the doserate further in certain areas if needed.

CONCLUSION

In conclusion we are quite satisfied with the results of our efforts to make it possible to perform the intended work in the RPV. We have also gained valuable experiences for the future. In order to, as far as possible, maintain the low doserates in the primary systems during the future operation we are considering possible methods to minimize recontamination. The verdict upon the RPV will be given in April-May 1994 and hopefully Oskarshamn 1 is back in operation late 1994.

Author Biography

Bengt Löwendahl is a Senior Radiation Protection Officer, acting as a corporate radiological advisor and controller at OKG AB, owner and operator of Oskarshamn NPP in Sweden. OKG AB operates three BWR´s and the Swedish Central Interim Storage Facility for Spent Nuclear Fuel (CLAB). During his time with OKG AB he has held positions as Head of Radiation Protection Section and Head of a Health and Safety Department (Radiation Protection, Emergency Planning, Industrial Safety, Fire Protection and Security). Before joining the OKG AB he worked for the Swedish State Power Board as Head of a Chemistry Section at the Ågesta Nuclear Central Heating and Power Station, a PHWR (now decommissioned) situated in the vicinity of Stockholm.

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PAPER 10-5
DISCUSSION

Wood: What do you do with the radioactive waste from the decontamination?

Löwendahl: As we used the normal clean-up system, we handled the resin as normal plant resin is handled.

Wood: You put it with your regular resins?

Löwendahl: Yes.

Wood: Did you use the ultraviolet process for destroying the decontamination chemicals?

Löwendahl: Yes, the ultraviolet-light process was used for the decomposition of the remaining oxalic acid together with a stoichiometric amount of hydrogen perioxide.

CO_2 PELLET DECONTAMINATION TECHNOLOGY AT WESTINGHOUSE HANFORD

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INTRODUCTION

Experimentation and testing with CO_2 pellet decontamination technology is being conducted at Westinghouse Hanford Company (WHC), Richland, Washington. There are 1,100 known existing waste sites at Hanford. The sites specified by federal and state agencies are currently being studied to determine the appropriate cleanup methods best for each site. These sites are contaminated and work on them is in compliance with the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA).[1]

There are also 63 treatment, storage, and disposal units, for example: groups of waste tanks or drums. In 1992, there were 100 planned activities scheduled to bring these units into the Resource Conservation and Recovery Act (RCRA) compliance or close them after waste removal. Ninety-six of these were completed. The remaining four were delayed or are being negotiated with regulatory agencies.[1] As a result of past defense program activities at Hanford a tremendous volume of materials and equipment have accumulated and require remediation.

BACKGROUND

To support WHC's mission of environmental remediation and restoration, the WHC ALARA (as low as reasonably achievable) Engineering group began evaluating a patented CO_2 pellet cleaning technology in January 1993. The CO_2 process is a unique dry process that uses dry ice as the exclusive decon medium, and does not use any hazardous chemicals, water, solid grit or aggregate materials. This process does not create costly secondary wastes and is a non-destructive surface cleaner.

A decision was made to test the technology on a variety of items at Hanford to gain first hand information about the efficacy of the CO_2 pellet decontamination methodology. Non-Destructive Cleaning, Inc. (NDC), is the vendor chosen to provide the demonstration of services contract at Hanford. The NDC mobile CO_2 decontamination unit is a stand alone, transportable, steel enclosure. No special mounting requirements are necessary and the unit can be placed on any firm flat surface, such as a paved lot or crushed stone. The unit is designed for cleaning items ranging in size from small hand tools to items up to twenty feet long with no weight limit.

NDC was chosen through a sole-source justification based on the perfection of a patented process including ventilation control and containment. A number of vendors can generate and deliver CO_2 pellets, but the key to an effective and safe CO_2 decontamination practice is proper ventilation and containment during the actual work. NDC, Inc., has demonstrated technical expertise in the diverse applications of the process to a variety of materials within the commercial nuclear industry.

CO_2 PELLET PROCESS

The NDC patented process/facility[2] uses small, solid carbon dioxide particles propelled by dry compressed air. The CO_2 particles shatter upon impact with the surface of the material to be cleaned and flash into dry CO_2 gas. The cleaning is accomplished by the rapidly expanding CO_2 gas lifting and flushing the foreign materials out. Microscopic sized particles are captured on high efficiency particulate air (HEPA) filters and larger materials are retrieved using HEPA-filtered vacuum cleaners. Therefore, the NDC mobile unit requires no drip pans or dikes or double walled leak protection designed to prevent leakage of radioactive liquids. Since no solid grit or aggregate is used, the CO_2 unit requires no bulk radioactive solid waste handling equipment. In addition, since no chemicals are required for the process no radioactive chemical processing or mixed waste handling facilities are needed. CO_2 levels have been demonstrated to remain below OSHA requirements and a CO_2 monitor verifies the levels during operation.

NDC had operated within the commercial nuclear industry for 9 equivalent operating years. Some examples of the items successfully decontaminated include: hand tools, power tools, pumps, tanks, glass, pipes, computer components and circuitry, manipulators, and lead shielding. The CO_2 pellets will clean metal objects, as well as softer objects like wood, plastics and rubbery materials without causing damage.

External Design of the CO_2 Unit

The main features of the CO_2 unit are designed to accommodate easy setup and tear down. Each facility arrives on site in three pieces that are simply joined together in place. The three pieces are two stand-alone, transportable, steel enclosure boxes and a flat bed containing the CO_2 tank and air compressor (Figure 1). All electrical interconnections are managed by a central power cable which is connected to a power control and distribution panel located within the mobile unit.

Internal Design of the CO_2 Unit

The CO_2 decontamination unit is designed with separate rooms: a machinery and electrical room, a large decontamination room, a decontamination cell room, and a count room where cleaned items are surveyed after cleaning. The unit has been designed with a complete HVAC system for "out of doors" operation in any environment. All floor and wall covering materials in the decontamination unit are specially selected and installed for ease of decontamination.

The decontamination room is completely lined with stainless steel, and includes a large entry door and an internal hoist that can handle up to two tons. The floor loading capacity is unlimited. The decon room ventilation system includes two pre-filters and a HEPA filter system. The decontamination room is pre-piped for the use of supplied breathing air for worker safety. A special rolling lift table equipped with an air driven vise to hold items for cleaning has also been built in to the unit.

The decontamination cell room houses a unique cleaning cell equipped with CO_2. The design includes high capacity sweep air flow, a pull-out drawer in the decon cell and a foot pedal operated air vise inside the decon cell to hold items being cleaned. The delivery of the CO_2 pellets is also triggered by a foot pedal. The decon cell has been ergonomically engineered for ease of operation. The CO_2 pellets are produced by the pelletizer, which converts CO_2 gas to rice-sized pellets. The pelletizer is located in the machinery and electrical room.

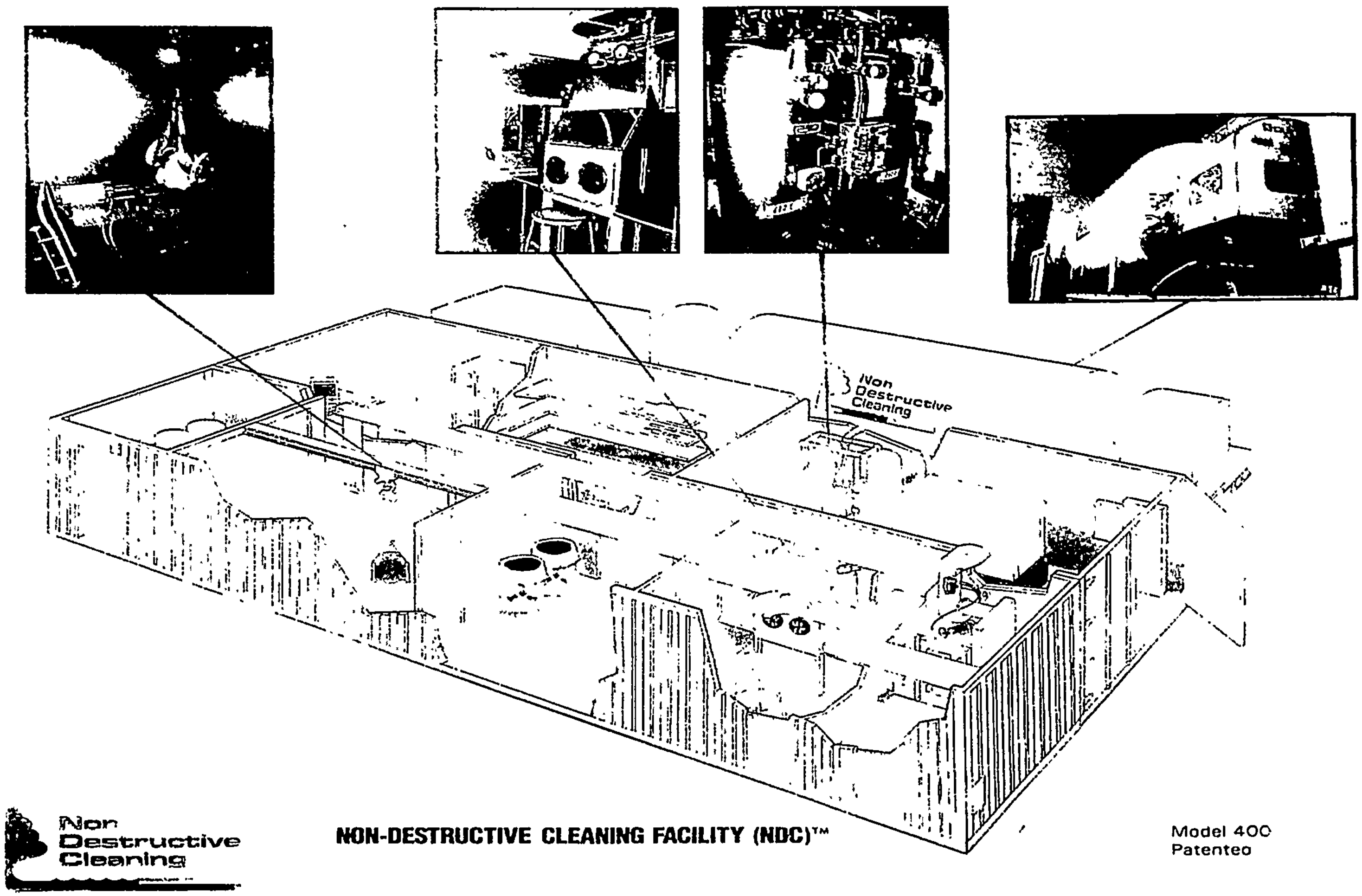

Figure 1. Facility Layout.

DEMONSTRATION OF SERVICES

NDC has been operating their patented process and facility for the last 5 months. The intent of the WHC demonstration of services contract was to test the effectiveness of the CO_2 process on a variety of items within the Department of Energy complex. The CO_2 pellet decontamination unit has operated at Hanford's B-Plant, the 222S laboratory, the Hanford Central Waste Complex (CWC), and will support the mission to clean up equipment located in the 300 Area Nuclear Fuels Fabrication building.

CO_2 SUCCESSES

The CO_2 pellet decontamination activities being conducted and tested at WHC, have resulted in unprecedented accomplishments. The real success has been attributed to the three phase process used in the CO_2 unit. Phase I is the cleaning, phase II is the ventilation control and containment as described in the process section, and phase III is the exhaust gas filtration. This third phase allows us to stay in compliance with environmental laws regulating the release of hazardous effluents to the environment.

B-Plant Demonstration

Successes at the B-Plant resulting from a three month demonstration of the CO_2 pellet decontamination technology included the free release of ten years accumulated materials and equipment. The efforts resulted in the elimination of thousands of cubic feet of radioactive waste as well as the decon of thousands of pounds of contaminated lead shielding for re-use. Items free released included assorted hand tools, electric drills, squirrel cages, shafts and bearings, shelving, door stop carriers, fan blades, and metal collars. An example of the decontamination log sheet (Table 1) shows the initial smearable contamination in disintegrations/minute (dpm)/100 cm^2, final smearable, approximate cleaning times and the air pressure (psi).

222S Process and Analytical Laboratories Demonstration

The demonstration activities at the 222S laboratories have resulted, so far, in greater than 2700 cubic feet of material decontaminated. Most of the material was free released. Where free release is not achieved two other approaches are used: (1) reduction of dose rates (ALARA), or (2) reducing burial costs by converting high-level waste to low-level waste.

In addition, a number of chemical sampling hoods were decontaminated for reuse or excess using the CO_2 process (Table 2). Hepa filter housings and a variety of duct work were also cleaned to determine the CO_2 decon efficiency.

FUTURE PLANS

Near-Term Plans

Beginning in May, the Hanford Central Waste Complex and the 300 Area Nuclear Fuels Fabrication Building are scheduled for a 10 week and 6 week demonstration of CO_2 decontamination activities, respectively. Items to be tested for the CWC are materials, tools, and equipment scheduled for burial at Hanford. Expectations are to recycle and excess all the items and save 100% of the burial costs. The Fuels Fabrication Building has some specialized equipment and fuel carts contaminated with low level uranium scheduled for CO_2 decontamination testing.

Table 1. B-Plant Non-Destructive Cleaning, Inc. Decontamination Log Sheet.

Completed By: ____________

Date of Test: 12-16-93

NDC DECONTAMINATION LOG SHEET

SECTION #	INITIAL SMEARABLE DPM/100CM²	FINAL SMEARABLE DPM/100CM²	D/F SMEARABLE	INITIAL RAD LEVEL	FINAL RAD LEVEL	D/F RAD LEVEL	CLEANING TIME MINUTES	CLEANING AIR PRESSURE	COMMENTS
SQUIRREL CAGE #1	500 CPM	< D		<1 mR	< D		20	120	FREE RELEASED
SQUIRREL CAGE #2	500 CPM	< D		<1 mR			20	120	RE SHOOT
SHAFT + BEARING #1	500 CPM	< D		<1 mR	<1 mR < D		15	120	FREE RELEASED
SHAFT + BEARING #2	500 CPM	< D		<1 mR	< D		15	120	FREE RELEASED
SHELVING	700 CPM	< D		<1 mR	< D		10	120	FREE RELEASED
DOOR STOP (7) CARRIERS	< D	< D	N/A	< D	< D	N/A	10 MIN EA	120	Cleaned for Technical Inspection
4-2x2 LEAD SHEETS	20,000 DPM	< D		<1 mR	< D		15 MIN EA	120	FREE RELEASED
DOOR STOP 4 CARRIERS	< D	< D	N/A	< D	< D	N/A	10 MIN EA	120	CLEANED FOR TECHNICAL INSPECTION
6-2x2 LEAD SHEETS	2K-20K	< D		<1 mR	< D		15 MIN EA	120	FREE RELEASED
FAN BLADES (SCRAP)	2K-5K	< D	N/A	<1 mR	< D	N/A	10 MIN EA	120	FREE RELEASED (10 PIECES)

Table 2. 222-S Laboratory Non-Destructive CO_2 Decontamination Log Sheet.

NON-DESTRUCTIVE CO_2 DECONTAMINATION LOG

DATE	ITEM DESCRIPTION	INITIAL ACTIVITY DPM/100cm²		DOSE RATE	FINAL ACTIVITY DPM/100cm²		CLEANING		COMMENTS RELEASED
		SMEARABLE	FIXED		SMEARABLE	FIXED	TIME (MIN)	AIR PRES PSIG	
3/14/94	6 @ 10" Ø DUCT 3'-4' Long	1K	0-1 K	<.5	<D	<D	90	120	FREE RELEASED
3/14/94	LEAD BRICKS 16	10K	10K→200K	<.5	<1K	<1K	90	120	FREE RELEASED
3/14/94	40 of ↑ LEAD BRICKS	<1K	10K→200K	<.5	<1K	10-200K	10	120	COND RELEASE
3/15/94	Chem Sampling Hood (bottom)	1K	0-1K	<.5	<1K	<1K	30	90-120	FREE RELEASE
3/15/94	HEPA FILTER HOUSING DOOR	1K	0-1K	<.5	<1K	<1K	10	40	FREE RELEASE
3/15/94	HEPA FILTER Housing + ACCS	1-2K	0-1K	<.5	<1K	<1K	15	60-130	Free Release
3/15/94	① SAW'S All Power Tool	<1K	10K	<.5	<1K	2K	15	130	Reshoot
	② Saw's All	<1K	2K	<.5	<1K	<1K	15	130	Free Release
3/15/94	Butter fly Valve	1K	N/A	<.5	<1K	<1K	5	130	Free Release
3/16/94	HEPA FILTER Housing + Accs	1K	N/A	<.5	<1K	<1K	30	130	Free Release
3/16/94	Chem Sample Hood (top)	4K	N/A	<.5	<1K	<1K	30	60-130	Free Release
3/16/94	Chem Sample Hood	1K	N/A	<.5	<1K	<1K	30	60-130	Free Release
3/17/94	Chem Sample Hood	1K	N/A	<.5	<1K	<1K	30	60-130	Free Release
3/17/94	Chem Sample Hood	8.1K	N/A	<.5	<1K	<1K	30	60-130	Free Release
3/17/94	Chem Sample Hood Glass door	10-15 K	8K-25K	<.5	<1K	3 K	30	130	NOT Released.
3/17/94	S.S. Sample Hood	1-2 K	N/A	<.5	<1K	<1K	30	130	Free Released

Long-Term Plans

CO_2 pellet decontamination technology will be used at Hanford in the years to come to assist the U.S. Department of Energy (DOE) contractors in completing their mission of environmental remediation. The long term plans are to, through the governmental competitive bid process, determine the appropriate vendor to continue the work demonstrated by NDC and their CO_2 process. Because of the successes with the demonstration of services with NDC, CO_2 pellet decontamination activities will continue at Hanford.

A workshop was held October 4-5, 1993, in Richland Washington. The workshop was sponsored by WHC Health Physics and supported a DOE Directive to form a CO_2 User's Group, and allowed 5 major DOE sites (7 contractors) who participated to utilize a savings through sharing concept. Eleven of the eighteen invited vendors attended the workshop.

COST SAVINGS

A cost savings of over 5 million dollars has been estimated for the term of the contract. Long term cost savings resulting from the use of CO_2 are estimated at over 50 million dollars.

CONCLUSION

CO_2 pellet decontamination technology will solve many of our decontamination problems. The technology is effective in solving these problems in terms of costs and performance. This state-of-the-art technology is revolutionary in terms of decontamination to ALARA, and in terms of waste management. CO_2 pellet technology will definitely enhance the public's image of Hanford by streamlining the mission of environmental restoration and remediation.

We have definitely learned that CO_2 technology doesn't end at CO_2 pellet delivery. The real technology includes the proper containment technology and the proper ventilation and filtration technologies because, without the entire package, CO_2 technology would be incomplete and unsuccessful.

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Author Biography

Theresa Aldridge is a Principal Health Physicist at Westinghouse Hanford Company (WHC) in the Radiological Protection and ALARA Department. Ms. Aldridge has twenty-five years experience in Health Physics. She has been involved in ALARA activities for several years, including a stint as the ALARA Program Manager for WHC. She also has many years of experience in internal dosimetry. Her early career years were spent as a Senior Technician in internal dosimetry with the Pacific Northwest Laboratory (PNL) Health Physics Department. She directed the Hanford routine internal dosimetry program for PNL Health Physics Department for eight years. Ms. Aldridge was the driving force in bringing the CO_2 decontamination technology demonstration project to Hanford.

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PAPER 10-6
DISCUSSION

Dionne: Do you have a feel for what type of decontamination factors you are getting with this device?

Aldridge: No, I don't. I did not come well prepared to give you that type of information. I can tell you that most of the items that we are bringing into the unit -- probably 95% -- are free released. We are recycling them. There is a lot of lead that we plan to decontaminate. We have 80 tons, most of which is pre-World War II lead. We are really excited about this decontamination and this process. I can get you that information -- I just don't have it today. It is an absolutely amazing process. I'm the sight cognizant engineer, and I have a lot of people who are very skeptical, and they want to come out and see the facility but they are very reluctant to give us their tools and equipment. They feel that this thing can't do a drill that has an aluminum body and a rubber cord and some plastic parts -- you can't clean it. They go away absolutely turned around.

SESSION 11

ROBOTICS AND REMOTE HANDLING

Chair:

Margaret Bennett

MOBILE ROBOTICS APPLICATION IN THE NUCLEAR INDUSTRY

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ABSTRACT

Mobile robots have been developed to perform hazardous operations in place of human workers. Applications include nuclear plant inspection/maintenance, decontamination and decommissioning, police/military explosive ordnance disposal (EOD), hostage/terrorist negotiations and fire fighting. Nuclear facilities have proven that robotic applications can be cost-effective solutions to reducing personnel exposure and plant downtime.

The first applications of mobile robots in the nuclear industry began in the early 1980's, with the first vehicles being one of a kind machines or adaptations of commercial EOD robots. These activities included efforts by numerous commercial companies, the U.S. Nuclear Regulatory Commission, EPRI, and several national laboratories. Some of these efforts were driven by the recovery and cleanup activities at TMI which demonstrated the potential and need for a remote means of performing surveillance and maintenance tasks in nuclear plants. The use of these machines is now becoming commonplace in nuclear facilities throughout the world. The hardware maturity and the confidence of the users has progressed to the point where the applications of mobile robots is no longer considered a novelty.

These machines are being used in applications where the result is to help achieve more agressive goals for personnel radiation exposure and plant availability, perform tasks more efficiently, and allow plant operators to retrieve information from areas previously considered inaccessible. Typical examples include surveillance in high radiation areas (during operation and outage activities), radiation surveys, waste handling, and decontamination evolutions.

This paper will discuss this evolution including specific applications experiences, examples of currently available technology, and the benefits derived from the use of mobile robotic vehicles in commercial nuclear power facilities.

BACKGROUND

The first significant efforts to introduce mobile robotics to the U.S. commercial nuclear industry began in the early 1980's. One of the earliest efforts was to develop a surveillance and sampling vehicle to work in the accident affected areas at Three Mile Island. The radiological conditions in these areas necessitated that an unmanned method be investigated. This work demonstrated that a remote system could be utilized in a commercial nuclear facility to access areas where personnel entry was undesirable or impossible.[1] It was also recognized that there were much broader benefits to be recognized in routine applications of these machines.

Programs sponsored by the Nuclear Regulatory Commission (NRC), the Electric Power Research Institute (EPRI), and the Department of Energy (DOE) along with private efforts led to the development of several systems which are currently used in nuclear facilities to lower exposure to plant workers. The initial programs were focused primarily on surveillance machines such as the SURVEYOR,[2] developed by ARD, Inc. under EPRI sponsorship (Figure 1) and SURBOT®,[3] developed by REMOTEC, Inc. and sponsored by the NRC (Figure 2). Development has continued on vehicles of this type with dozens of these machines now in use with current applications encompassing video and audio surveillance, radiation and contamination measurements, decontamination tasks, pipe inspection, and waste handling. The following paragraphs will give specific examples of some of these tasks and their benefits.

Figure 1. Surveyor.

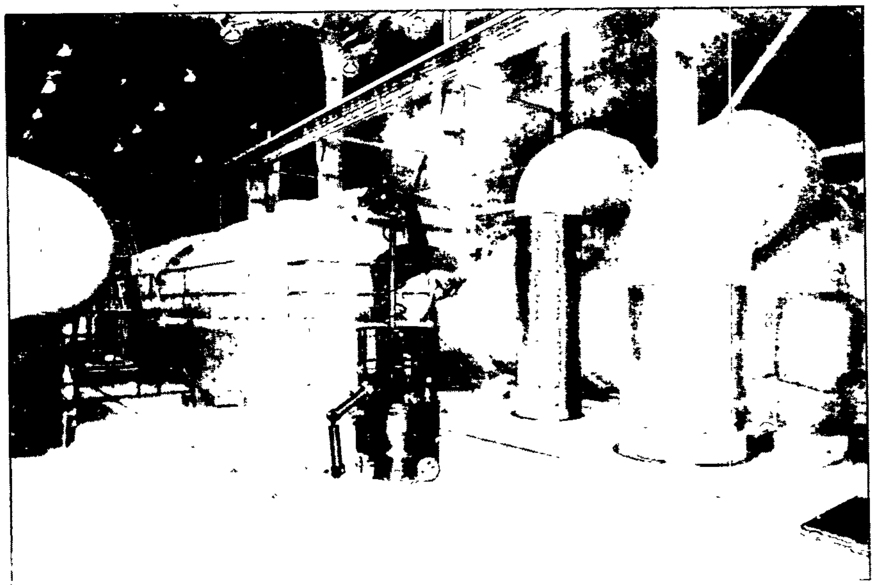

Figure 2. SURBOT® Performs an Inspection During Testing at Brown's Ferry.

RADWASTE TANK FLOW EDUCTOR CLEANOUT

At Pennsylvania Power and Light's Susquehanna Nuclear Plant, a mobile vehicle is used to cleanout plugged flow mixing eductors inside radwaste processing tanks.[4] These tanks include Reactor Water Cleanup (RWCU) resin phase separator and waste sludge phase separator tanks. The vehicle manipulates a flexible hydroblasting nozzle into the nozzle openings thus eliminating the need for human entry. Previous task completion had required human entry into the tank with the accompanying hazards of radiation exposure, confined space entry, and high-pressure fluid use. The completion of the task remotely resulted in significant radiation exposure savings.

The Susquehanna radwaste processing system includes two RWCU phase separator and one waste sludge phase separator tanks. A typical tank configuration is shown in Figure 3. A system of eductor nozzles and associated piping is used to provide mixing in the tanks. The mixture pumped through the nozzles is a dense resin/water slurry and the venturi section of the nozzles tend to plug during processing.

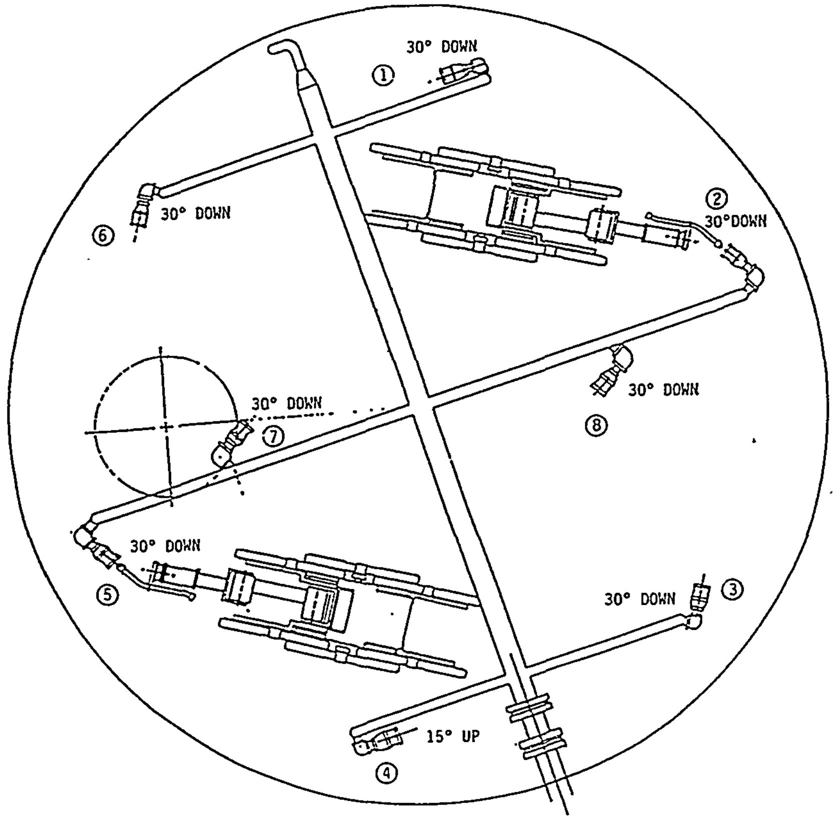

Figure 3. Typical Tank Configuration.

Located in shielded rooms within the plant, the typical radiation levels inside the tanks are 30-50 mSv/hr (3-5 R/hr) with accompanying high contamination levels. Previously, the method for clearing of the nozzles had been for a worker to enter the tank via a manway and manually insert a hydrolaser into each nozzle, one at a time. Water pressure would then be applied to clear the obstruction. Due to high radiation levels in the tank, the worker was limited to only a few minutes of stay time in the tank. This typically resulted in several entries being needed to complete the work. The significant radiation exposure and concern for worker safety in the tank led the utility to investigate alternate means for completing this task. A mobile robot was developed and used to clear the clogged nozzles. This task has been repeated on several occasions and the exposure savings alone have justified the equipment acquisitions and personnel safety has been enhanced by eliminating the confined space entry. An average of 30 mSv (3 Rem) exposure was received during manual cleaning of the tanks. This has been reduced to 4 mSv (400 mrem) on average. Additionally, the robotic vehicle is also used for other operations in high radiation areas of the plant.

ROUTINE SURVEILLANCE IN BWR STEAM CYCLE

The steam cycle operating areas of BWR's have high gamma dose rates due to the presence of N-16. These areas occasionally require entry for tasks such as equipment surveillance, corrective maintenance, and other essential tasks. In addition to exposure concerns, many of these areas typically have elevated temperature and humidity levels further restricting personnel entry.

Several stations, including Brown's Ferry, Hope Creek, LaSalle, Perry, and Pilgrim are utilizing mobile vehicles to perform entries into these areas.[5] Figure 4 shows a machine being used at Hope Creek to perform a typical surveillance task. The typical machine can perform visual and audio surveillance, radiation and contamination measurements, monitor temperature and humidity, and record the information on videotape for record keeping. Additionally, as the operator is located outside the affected area, the tasks can be performed without the time constraints caused by the environmental conditions which can contribute to a more thorough completion of the task. The savings vary according to the plant conditions and the application, with average savings exceeding 100 mSv (10 Rem) annually.[6]

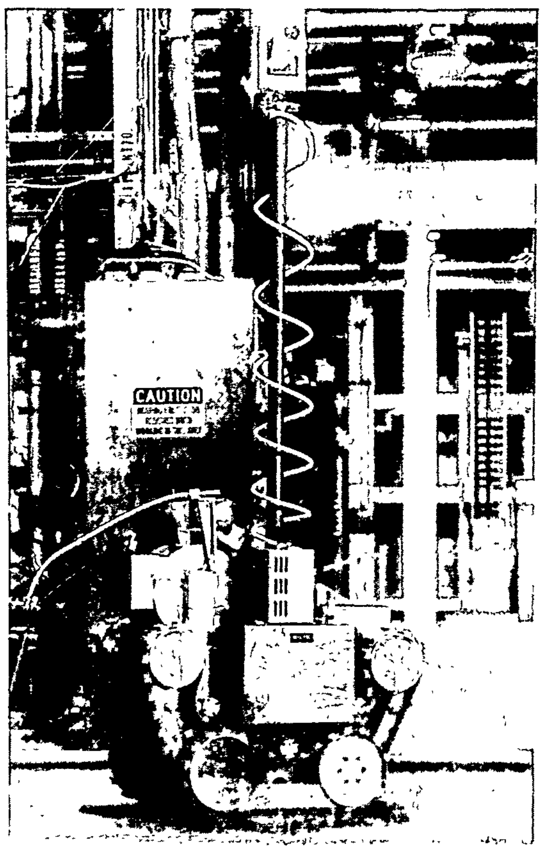

Figure 4. A Mobile Vehicle in Use at Hope Creek.

CONTAINMENT ENTRY AND SURVEILLANCE IN OPERATING PWR'S

This design of a PWR typically utilizes a large concrete structure called the containment building to house the reactor and supporting components such as the steam generators, reactor coolant pumps, and coolant piping. Inside the containment, these components are further segregated inside concrete rooms or partitions. This allows personnel to access the containment building when the reactor is operating, but access into most of the rooms is limited or not allowed due to high radiation levels.

Occasionally during plant operations, access to these areas is needed to monitor plant components. Many times this has required that the plant reduce power or shutdown to lower radiation levels and allow entry. This results in a loss of plant availability to generate electricity and loss of revenue. A method to allow entry into these areas without power reduction will save personnel exposure, improve access to plant information, and improve efficiency. Remote vehicles are being used to perform these and other tasks at several PWR stations including Comanche Peak, Indian Point 2, Palo Verde, South Texas Project, and Vogtle.

DECONTAMINATION TASKS

Several utilities have also used their machines to perform decontamination tasks throughout the plants. The type of machines used vary from specialty underwater vacuuming robots to general purpose machines equipped with task specific end effectors. The typical tasks the machines are used for are not routine plant decontamination work, but are instead located in areas with high dose rates and contamination levels.

In PWR's, remote vehicles are routinely used to vacuum the reactor cavity areas during refueling operations. The machine operates submerged in conjunction with an underwater vacuum and filters to remove activity from the floor prior to draindown for reactor vessel reassembly. This allows for a more thorough cleaning of the floor and is much less labor intensive than manual methods.

Recently, some plants have begun to use these vehicles to enter and decontaminate areas where entries have not been made for several years due to very high dose rates and contamination levels. In one case, a utility used a mobile robot to enter a resin storage tank room where dose rates ranged up to 600 mSv/hr (60 R/hr) with spent resin up to 18 cm. (6 in.) deep on the floor. The machine was used to vacuum the material up to a collection device and perform all post decontamination radiological surveys of the room. The application of a strippable coating with the machine had been tested in mockup, but was deemed unnecessary after vacuuming resulted in acceptable levels of contamination. The machine also performed a complete visual surveillance of the room including the tops of the tanks.

SUMMARY

The use of mobile vehicles in nuclear power plants for exposure reduction and plant availability improvement is commonplace. As the operators gain experience, the applications and savings continue to grow and provide additional benefits to the users. This widespread use has enhanced the maturity of the technology and provided the economies of scale to assist the ALARA professional in the justification process. The growth of this industry outside the commercial nuclear environment also is providing technological advances and product improvements which will have a dramatic effect on the potential savings associated with this technology.

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Authors' Biographies

John R. White is the President and founder of REMOTEC, Inc., a world leader in remote technology and robotic vehicles. He has over 30 years of experience in remote operations in hot cells, fuel reprocessing, power reactors, and other hazardous environments. The holder of numerous patents and trademarks, Mr. White is recognized around the world as one of the leading experts in the field of robotics and remote systems.

Sammy L. Jones is presently Field Projects Manager for REMOTEC, Inc. of Oak Ridge, Tennessee. A former radiological engineer in a commercial nuclear station, Mr. Jones is responsible for the applications and design of remote systems to expand their use in nuclear and other hazardous areas. Mr. Jones's background includes positions in both commercial PWR and BWR facilities in a variety of areas including ALARA, outage planning, and field operations.

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PAPER 11-1
DISCUSSION

Khan: At the time of Chernobyl, a lot of the robotics that went there got stuck in high-radiation areas. Their batteries ran out, etc. Have there been any significant developments since that period? Have robotics come a long way since then?

Jones: The primary problems they had at Chernobyl were terrain problems in reaching the accident scene and problems with the control systems of the machines which were microprocessor controlled, machines similar to ours. That is one of the reasons that DOE is sponsoring some radiation hardening research for mobile vehicles. This research is a big emphasis now, to rad harden the electronics. The machines are either electric or hydraulic, so these components are fairly rad tolerant, and the technology is very easily adaptable. There is a lot of work being performed in the microprocessor area. Some of it is a spinoff from work on space-based systems for NASA. JPL, Sandia, and Lawrence Livermore have done a lot of work in this area.

BOILING WATER REACTOR RADIATION SHIELDED CONTROL ROD DRIVE HOUSING SUPPORTS

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ABSTRACT

The Control Rod Drive (CRD) mechanisms are located in the area below the reactor vessel in a Boiling Water Reactor (BWR). Specifically, these CRDs are located between the bottom of the reactor vessel and above an interlocking structure of steel bars and rods, herein identified as CRD Housing Supports. The CRD Housing Supports are designed to limit the travel of a Control Rod and Control Rod Drive in the event that the CRD vessel attachment were to fail, allowing the CRD to be ejected from the vessel. By limiting the travel of the ejected CRD, the supports prevent a nuclear overpower excursion that could occur as a result of the ejected CRD.

The Housing Support structure must be disassembled in order to remove CRDs for replacement or maintenance. The disassembly task can require a significant amount of outage time and personnel radiation exposure dependent on the number and location of the CRDs to be changed out.

This paper presents a way to minimize personal radiation exposure through the re-design of the Housing Support structure. The following paragraphs also delineate a method of avoiding the awkward, manual, handling of the structure under the reactor vessel during a CRD change out.

DISCUSSION

Existing Housing Support Structure

The CRD Housing Support, Figure 1, structure consists of individual Hanger Rods that are attached to a beam structure located immediately below the reactor vessel. The Hanger Rods extend to an elevation below the CRDs. Support Bars span the distance between each Hanger Rod in the area below the CRDs and are held to each Hanger Rod by means of a bolted connection. A pair of Grid Bars are installed on top of the Support Bars which interlock with adjacent Grid Bars to form an integrated system. The Grid Bars transfer the load of a ruptured CRD housing to the support bars and additionally limit any CRD travel should an ejection occur. The Grid Bars are held in place with two Grid Clamps and a Grid Clamp Bolt. In addition, each Hanger Rod has a nut which must be adjusted to provide the desired Grid Bar/CRD housing clearance.

To change out a CRD, the Grid Bars must first be removed. The quantity of Grid Bars removed is dependent upon the number and location of the CRDs selected for replacement. To remove the Grid Bars for each CRD, the operator must manually unscrew the Grid Clamp Bolt, remove the two Grid Clamps, and then remove the two Grid Bars. Each weighs approximately 40 pounds, and is awkward to handle in the confined space under the vessel. Since the Grid Bars are interlocking, all Grid Bars have to be removed starting from the peripheral row and working inwards to the specific CRD scheduled for removal. The CRD is then

Control Rod Drive Housing Support Bars and Radiation Shields Situation before Modification

ASEA BROWN BOVERI

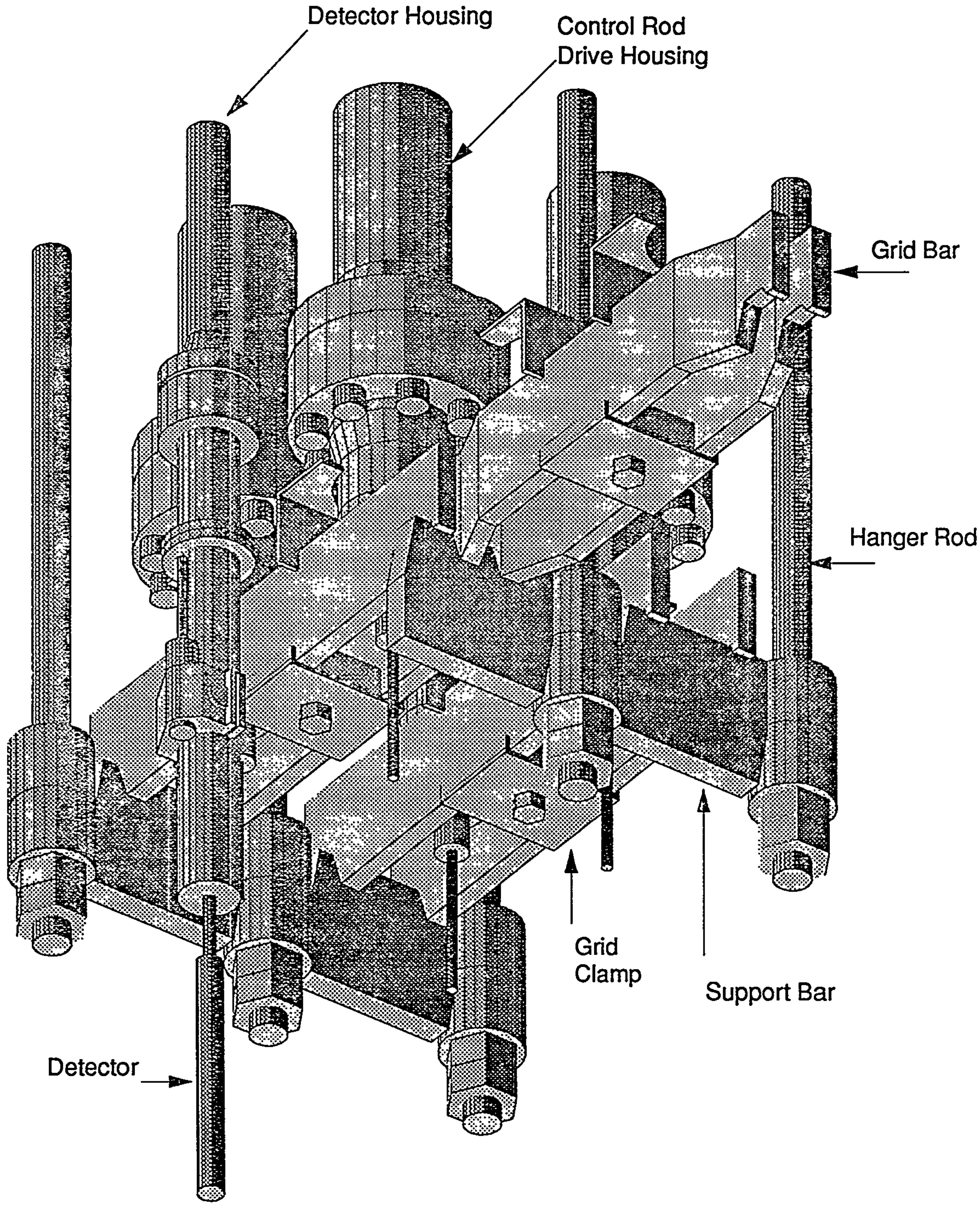


Figure 1

replaced. Once the change out is complete, all of the removed Grid Bars must be re-installed, in the reverse order.

As the Grid Bars are heavy and awkward to handle, especially in the confined space under the reactor vessel, a dropped section could result in a serious injury. Additionally, removal and re-installation of the Grid Bars is a time consuming outage process. Because of the high radiation fields directly under the reactor vessel, manual handling of the Grid Bars results in a substantial radiation dose to personnel during each outage. Moving heavy pieces of metal in this area also carries the risk of snagging nearby electrical cables.

Proposed Solution

Description

The ABB solution (patent pending) shown in Figures 2 and 3, is to replace the existing Support Bars with a new Support Bar system that uses the same Hanger Rods but which run perpendicular to those in the existing system. The Grid Bars in the original system are replaced with a cylindrical Radiation Shield at each CRD. The Radiation Shield provides the same function as the original Grid Bars while doubling as efficient radiation shields for personnel working under the vessel. Each Radiation Shield interlocks with the adjacent Radiation Shields to ensure that it remains in its desired location. This interlock, however, has been configured such that it does not require disassembly of any adjacent CRD supports and Support Bars.

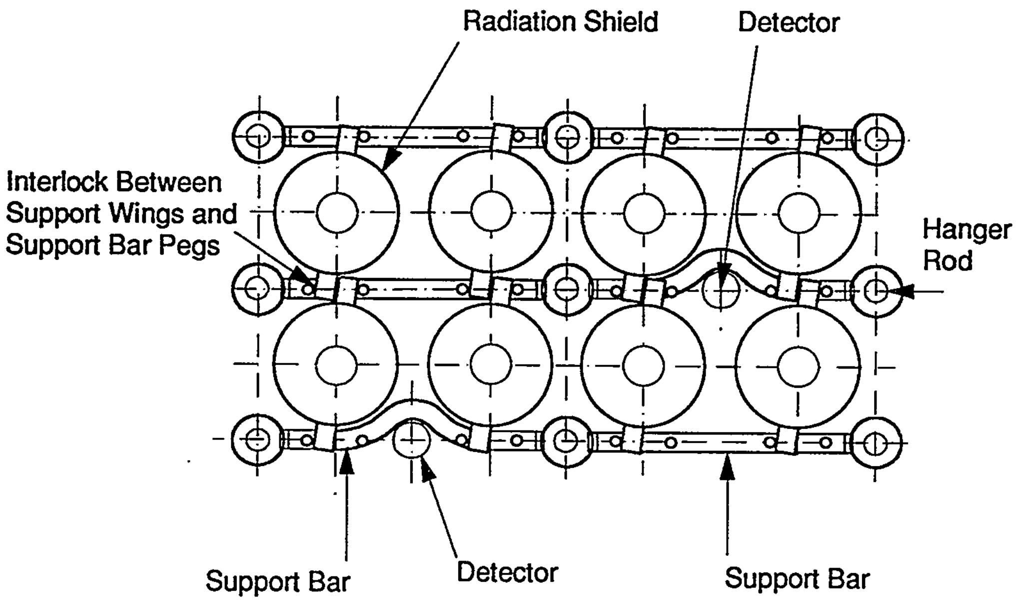
In addition to the CRDs exiting the bottom of the reactor vessel, detectors (Low Power Range Monitors (LPRMs) and Source Range Monitors (SRMs)) are located at various positions between CRDs. To avoid an interference in the areas where the detectors are located, the Support Bars are configured with a built in offset to bypass detector positions. In this manner, maintenance on the detectors and drive units can be performed without removing the Support Bars.

Operations Description

During refueling operations, specific CRDs are selected for change out. With the ABB Support Bar system, in conjunction with a remotely operated CRD Handling Machine, the Radiation Shields can be remotely removed and installed individually using a CRD Handling Machine without removing any of the Support Bars. This allows the remaining Support Bars and Radiation Shields to remain in place to provide shielding during the change outs. The removal process becomes a simple procedure, whether done manually or remotely. To remove a Radiation Shield, the shield is lifted slightly off the Support Bar network to clear its interlock with adjacent Radiation Shields, rotated 90^0 so that the interlocks now clear the Support Bars, then lowered out of the Support Bar network and stored out of the way. The Radiation Shield is re-installed after the CRD replacement is complete. This leaves the overall radiation shield intact and ready for the next CRD change out. By reducing the process to the removal of only one Radiation Shield per changed CRD, the time required for the change out is significantly reduced. Any CRD can be changed as fast as any other CRD with the same low dose. These unique features of the ABB system will result in considerable reductions of personnel exposure and time over the current Grid Bar configuration.

The new Support Bar and Radiation Shield arrangement would reduce the general area dose under the reactor vessel to approximately half of the radiation field that is presently seen. This corresponds to the dose rate reduction data recorded when temporary radiation shields are used. Therefore, an area which normally has a dose rate of 300 millirem per hour would have a reduced radiation field of approximately 150 millirem per hour after installation of the new system.

A typical accumulated dose accrued during the change out of 20 CRDs (exclusive of teh shoot out steel remaoval and installation) without radiation shielding has been recorded as approximately 10 man-rem. A


Radiation Shield
Detector
Interlock Between
Support Wings and
Support Bar Pegs
Hanger
Rod
Support Bar
Detector
Support Bar

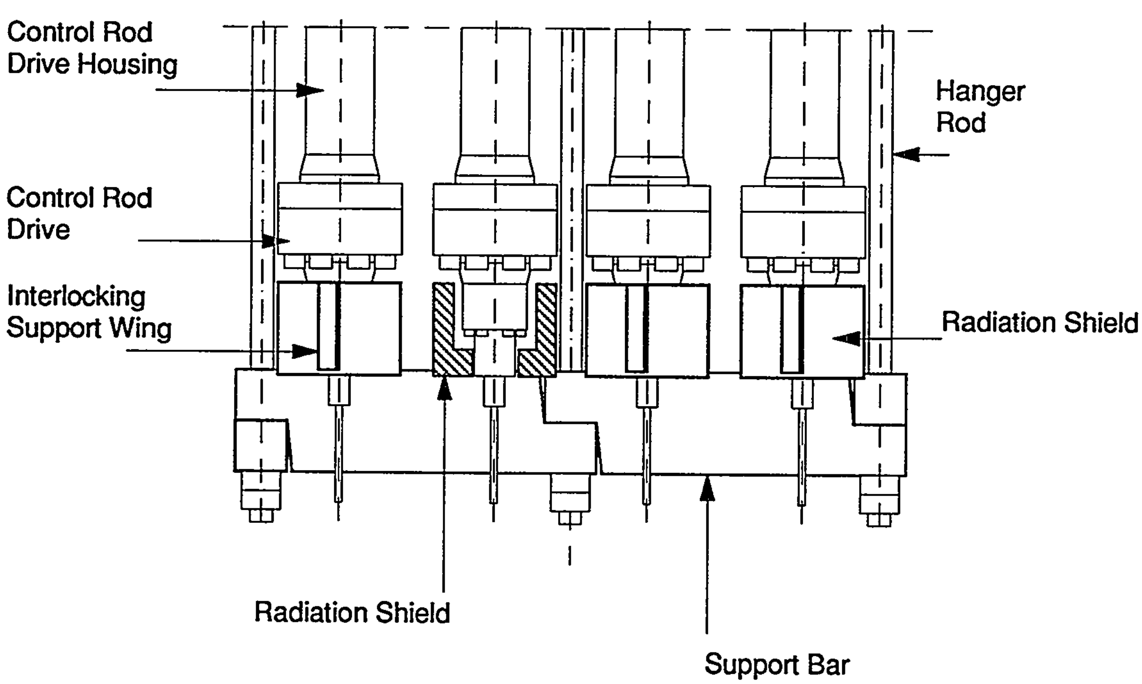

Control Rod
Drive Housing
Hanger
Rod
Control Rod
Drive
Interlocking
Support Wing
Radiation Shield
Radiation Shield
Support Bar

Figure 2

ABB
ASEA BROWN BOVERI

Control Rod Drive Housing Support Bars and Radiation Shields

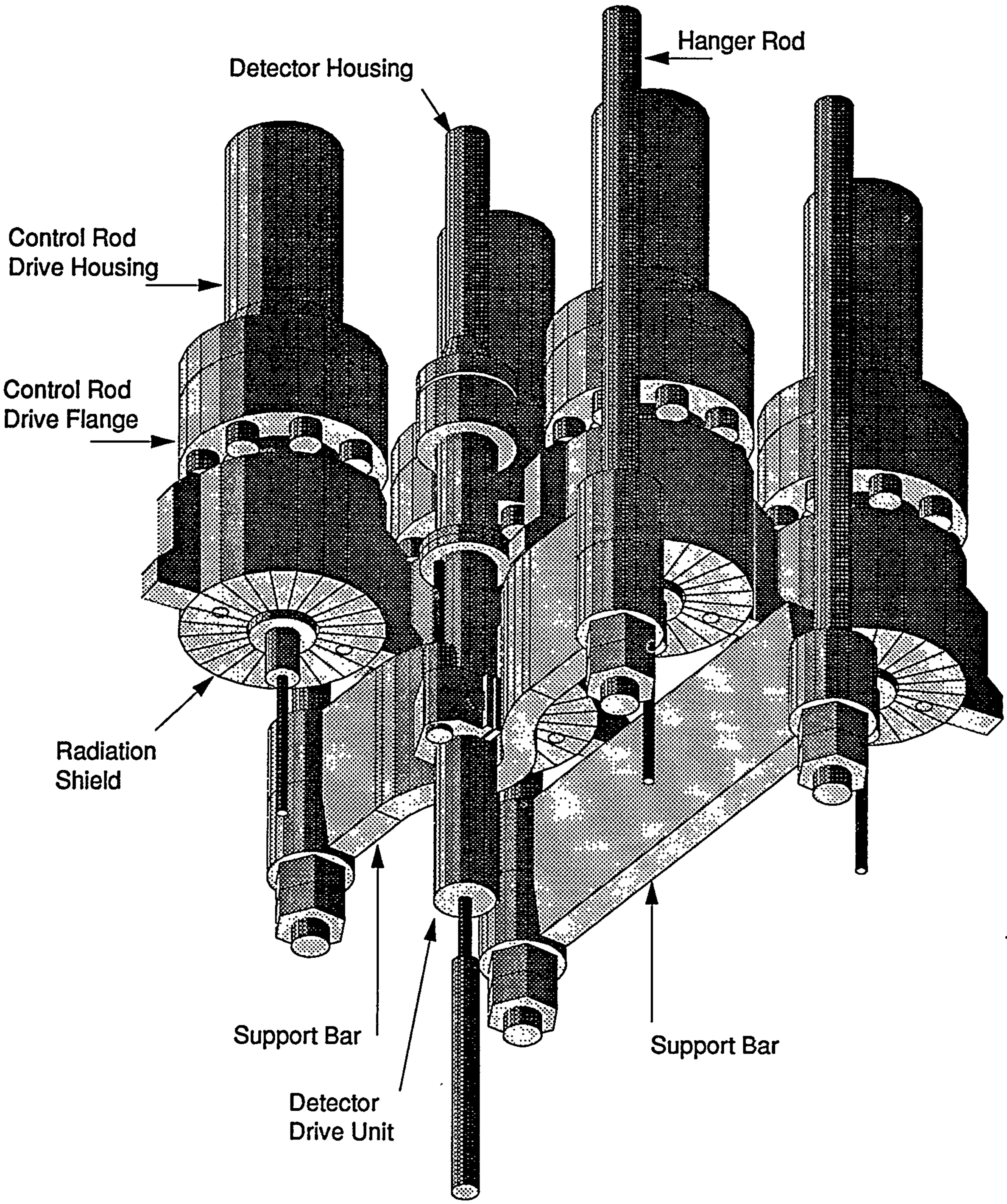


Figure 3

savings of approximately 5 man-rem is expected to be realized through the use of ABB's individual Radiation Shields and Support Bar system.

Considerable dose savings would also be achieved since, with the ABB system, the steel grid does not have to be removed or re-installed for a CRD change out. The removal and re-installation activities typically account for approximately 2 man-rem during each outage. Hence, the total estimated savings for typical CRD replacement activities, using the new ABB system, is estimated to be approximately 7 man-rem (5 man-rem for CRD change out and 2 man-rem for removal and re-installation of the Radiation Shields) or 60% of that now seen.

The following table shows accumulated doses and man-hours for the replacement of 40 CRDs at a US BWR:

Task Description	**Existing System (man-hours)**	**Existing System (Rem)**
Remove Shoot Out Steel	40	2.5
- PIP Work	100	1.2
- Prep CRD Tools and Equipment	16	0.16
- Decontaminate CRD	15	0.15
- Rad Protection	75	0.15
Remove/Install CRDs	80	2.5
- Rad Protection	20	0.7
- QC Support	10	0.2
- Under vessel Support	15	0.53
CRD Flush and Disassembly	344	2.65
Transport CRDs and Filters	31	0.6
Repair CRDs	65	2.2
Test Tip Tubing	16	0.2
Install Shoot Out Steel	50	0.8
- Rad Protection	10	0.15
- Under vessel Support	15	0.3
TOTAL	**902.00**	**14.99**

Authors' Biographies

Bengt Baversten is a BWR Consultant on loan to ABB Combustion Engineering Nuclear Operations (ABB CENO) from ABB Atom, Vasteras, Sweden. At ABB Atom, he is a Service Manager of BWR Service Technology. He is presently responsible for the transfer of BWR service technology to the US BWR utilities through ABB CENO. Previously, he was responsible for the field service related activities at ABB Atom which entailed not only field service activities, but the development of new tools and service methods. He has a B.S. in Mechanical Engineering from Eskilstuna Hogre Tekniska Laroverk, Sweden.

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PAPER 11-2
DISCUSSION

Borst: Is the shielding material some sort of encapsulated lead?

Baversten: We think it would be steel only since we have about up to an inch and a half available of wall thickness in these buckets and also in the bottom of it. Steel would be enough. Each of these buckets would weigh about 40 to 80 pounds, meaning the same weight per steel grid. I mentioned 40 pounds per piece, that would be about 80 pounds (for the steel grid). These buckets could be twice as heavy, twice the amount of steel.

Borst: Is my Engineering Department going to have strong concerns when I say that I want to get rid of some their shoot-out steel?

Baversten: Actually, this is the project going on now at the Susquehanna plant. If you want to save dose you have to replace what you have today.

Borst: Is that actually installed at Susquehanna now?

Baversten: No, they are now planning for it. The design is going to be fine tuned. That is why, during this presentation, there are differences in many of the sketches that I showed. They have developed over the last few weeks. Of course, in order to replace the existing bars and install the new bars, you will take some dose. Actually, a fair amount of dose. I assume that the time to do that replacement would be twice as long as you do today to remove it and install existing shoot-out steel. But that's a one time job and after that you just have savings.

Giordano: The concept, as I understand it, is that you would leave the shielding installed undervessel during operation?

Baversten: Yes, that's right.

Giordano: Is that also going through the design considerations from seismics and other things at Susquehanna?

Baversten: Yes, to the same extent that you have for existing steel.

Giordano: It appeared to me that in the removal and installation of the CRD, there is one added step of taking that lead shield out on the pogo stick and then going up and getting the drive and removing that, or does it just sit there on the rail?

Baversten: Actually, there is an extra step. You go up with your mast until you touch the steel piece, turn it 90°, lower it down, remove the steel, and then you have access to your control rod drive.

Giordano: Then, when I am removing the drive, the plugs are there on the platform as I am doing the work, but off to a different side. Does a man have to move that out of the way?

Baversten: You mean the radiation shield?

Giordano: Yes.

Baversten: Just place it on the platform. You have a fixed position there to place it. You have only one of them out at a time. You don't need to store anything. You don't need to move shoot-out steel out of there to another area.

AUTOMATION OF STEAM GENERATOR SERVICES AT PUBLIC SERVICE ELECTRIC & GAS

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ABSTRACT

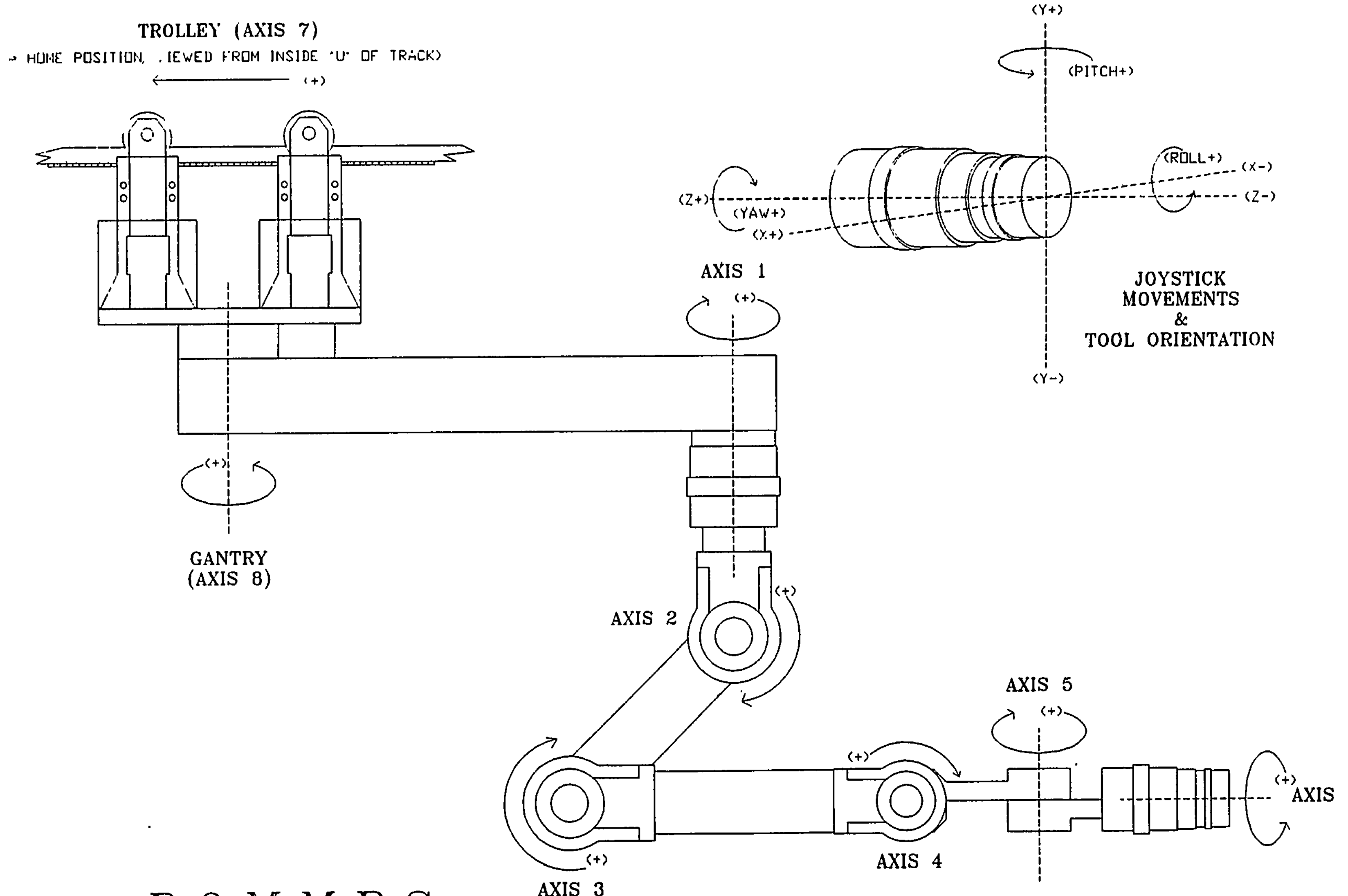
Public Service Electric & Gas takes an aggressive approach to pursuing new exposure reduction techniques. Evaluation of historic outage exposure shows that over the last eight refueling outages, primary steam generator work has averaged sixty-six (66) person-rem, or, approximately twenty-five percent (25%) of the general outage exposure at Salem Station. This maintenance evolution represents the largest percentage of exposure for any single activity. Because of this, primary steam generator work represents an excellent opportunity for the development of significant exposure reduction techniques.

A study of primary steam generator maintenance activities demonstrated that seventy-five percent (75%) of radiation exposure was due to work activities of the primary steam generator platform, and that development of automated methods for performing these activities was worth pursuing. Existing robotics systems were examined and it was found that a new approach would have to be developed. This resulted in a joint research and development project between Westinghouse and Public Service Electric & Gas to develop an automated system of accomplishing the Health Physics functions on the primary steam generator platform. R.O.M.M.R.S. (Remotely Operated Managed Maintenance Robotics System) was the result of this venture.

R.O.M.M.R.S. is a fully integrated robotic arm, delivered on an overhead track and trolley system and using multiple end effectors (an "end effector" is basically an automated tool at the end of a robotic arm that, when activated, may accomplish a given task.) R.O.M.M.R.S. has successfully completed mock-up testing and will be implemented in Salem's fall Unit Two Outage for a complete field test and evaluation. R.O.M.M.R.S. will perform the Health Physics functions normally provided by radiation protection technicians (e.g., beta and gamma radiation surveys, air sampling, decontamination and job observation). Additionally, R.O.M.M.R.S. will attempt to do bolt hole cleaning, one of the many platform worker activities executed by maintenance technicians.

INTRODUCTION

With the increasing competition and rising O&M costs in our industry, the option of replacing steam generators due to tube degradation or in support of plant life extension, is becoming less and less an economic option. for this reason utilities will be striving to ensure that the steam generators they currently own will last the life of the plant. This implies that the scope of work and associated radiation exposure for primary steam generator maintenance will be increasing as plants step up NDE exams and other maintenance activities to repair degraded tubes. In an effort to address this concern, R.O.M.M.R.S. was developed with the intention of reducing labor requirements, increasing overall job efficiency, and decreasing radiation exposure associated with these activities.



R.O.M.M.R.S.
SIMPLIFIED DRAWING

NOTE: A YAW+ ROTATES AXIS 6 IN A NEGATIVE DIRECTION.

Figure 1

R.O.M.M.R.S. System Overview

R.O.M.M.R.S. consists of a two degree of freedom ("degree of freedom" refers to any of a limited number of ways in which a robot may move) mobile base, delivering a six degree of freedom robotic arm (Figure 1). The mobile base is supported by a modular track system that can be configured to allow the robot access to a given area of interest. The robot is animated using a highly accurate three-dimensional computer simulation of the robot and its environment (i.e., the primary steam generator platform, etc.) on a Silicon Graphics Indigo 2® work station. The Indigo 2® was the computer used to create the dinosaur animation sequences in the film "Jurassic Park." This computer (located outside the Radiologically Controlled Area) is networked to a Robot Operating Computer so that manipulation of the robot model induces the actual robot to move in kind.

The 3D simulation is accomplished using a modified version of the ROBCAD® software. ROBCAD®, originally a robot design tool, provides the R.O.M.M.R.S. operator with the ability to initiate and view movements of the robot within its environment. This the first time that a 3D simulation has been used to control an industrial robot. Another breakthrough accomplished by the R.O.M.M.R.S. design team is the development of an software algorithm which can incorporate all eight (8) degrees of freedom of the R.O.M.M.R.S. mechanism. Recognizing that there is often a difference in dimensions between what is shown in plant drawings and what is actually in the field, the ROBCAD® system allows for easy re-calibration of the 3D model. By selecting several predetermined calibration points in the field, the robot can communicate actual locations of these points to the work station so that ROBCAD® can make appropriate adjustments in the model to correct the simulation to actual field dimensions.

ROBCAD® also provides collision detection for R.O.M.M.R.S., preventing damage to plant equipment, or to the robot itself. As additional field equipment is added to the environment, these obstacles can be accurately modeled into the simulation along with appropriated collision detection safeguards.

R.O.M.M.R.S. is able to perform multiple tasks by choosing from various custom designed end effectors. In order to perform health physics tasks, six (6) different end effectors were designed:

- General Purpose Gripper - three fingered claw used to grab objects as well as change air sample heads.
- Swiping Tool - obtains loose contamination samples.
- Beta Scanning Tool - surveys environment surfaces for beta radiation
- Telescopic Steam Generator Bowl Survey Tool - performs gamma radiation surveys of the steam generator bowl area.
- Decontamination Tool - performs decontamination of platform surfaces.
- Vacuum Tool - provides loose debris removal capability.

These end effectors are harnessless (another robotics breakthrough) to allow remote coupling and uncoupling.

Gamma detection is continuously accomplished by four integrated gamma detectors (SAIC® Commercial Products PDE-4™) located at the tool coupling. Radiological surveillance is performed by dividing an area into a grid of eight inch squares (this information is then recorded in the database of the 3D simulation) and moving the robot to each section of the grid, at planes of eighteen (18), thirty-six (36) and seventy (70) inches above the surface (i.e., knee, waist and head readings). The readings are displayed and recorded at the work station.

Cost/Benefit

Public Service Electric & Gas and Westinghouse have been developing R.O.M.M.R.S. since 1991. By utilizing existing technologies, equipment and in-house expertise, the initial financial obligations for both parties were kept to a minimum. The total cost to Public Service has been a little less than two million dollars.

Salem station expects to get back far more than our initial investment through outage contractor staffing reductions and collective exposure savings. When fully operational, we expect to reduce our primary steam generator health physics coverage by at least three (3) ANSI qualified senior technicians per platform per outage. This equates to at least $100,000 savings per outage.

The collective radiation exposure savings will also be significant. Based on historical outage data, we expect to save five (5) to twenty (20) person-rem from our outage total. This indicates an additional savings of $50,000 to $200,000 based on $10,000 per person-rem.

Future Applications

Although R. O.M.M.R. S. is a stand alone system, its future allows for integration into other control systems. For example, the operator that is controlling the robot that gathers eddy current data inside the steam generator bowl, will have the ability to control R.O.M.M.R.S. from the same work station. This will allow control of both systems by one operator in a completely integrated system. The benefits of this integration are reduction of support personnel, and increased overall job efficiency by using R.O.M.M.R.S. to support the steam generator bowl robot (i.e., tool change out, etc.).

It is important to remember that industrial robotic applications have proven to be economically beneficial when ever the task at hand was either dangerous, dirty or dull and repetitive. These are the three "D's" of industrial robotics. The basic idea behind R.O.M.M.R.S., a modular track system which delivers a remotely controlled robot, is very flexible. The track system can be configured to allow the robot to go around corners and also move vertically. By adding track sections, the robot can travel as far as the applications requires. The track and trolley could be reconfigured to deliver any type of instrument, tool or monitoring device to an otherwise inaccessible area. Adding "bus bars" to supply power to the robot from the track would be simple to accomplish. This would allow the system to be completely free of wire and cables, thus providing maximum system flexibility.

A list of potential applications for R.O.M.M.R.S. is virtually endless. With the modular track design, a reliable field tested robotic arm capable of delivering any type of end effector, and the computer control system allowing easy performance of both simple and complex tasks with minimum operator experience, makes R.O.M.M.R.S. an extremely adaptive system. This system has application potential in present PWR and BWR reactor environments, advanced reactor designs, underwater applications, as well as any other hazardous environment.

Author Biography

John Wray is the Radiation Protection Manager at the Salem Nuclear Generating Station responsible for managing the Operational Health Physics, ALARA, and Radioactive Waste programs at this two unit PWR site. He has 21 years of varied nuclear power industry experience including assignments with a major architect engineering firm, the Nuclear Regulatory Commission and Southern California Edison's corporate office. John is a Registered Professional Nuclear Engineer and a Certified Power Reactor Health Physicist. Mr. Wray holds a B. S. degree in Nuclear Engineering from Rensselaer Polytechnic Institute and an M.S. degree in Environmental Health Engineering from Northeastern University.

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COMPUTER-CONTROLLED WALL SERVICING ROBOT

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After four years of cooperative research, Pentek has unveiled a new robot with the capability to automatically deliver a variety of cleaning, painting, inspection, and surveillance devices to large vertical surfaces (Figure 1).

The completely computer-controlled robot can position a working tool on a 50-foot tall by 50-foot wide vertical surface with a repeatability of 1/16 inch. The working end can literally "fly" across the face of a wall at speed of 60 feet per minute, and can handle working loads of 350 pounds.

The robot was originally developed to decontaminate the walls of reactor fueling cavities at commercial nuclear power plants during fuel outages. If these cavities are left to dry after reactor refueling, contamination present in the residue could later become airborne and move throughout the containment building. Decontaminating the cavity during the refueling outage reduces the need for restrictive personal protective equipment during plant operations to limit the dose rates.

The initial design considerations for the new cavity decontamination robot include:

1. Application to all reactor cavities of differing configurations at both BWRs and PWRs.

2. Time reduction in reactor cavity in decontamination in an effort to reduce outage time.

3. Eliminate use of the reactor building crane.

4. Reduce occupational risks associated with reactor cavity decontamination.

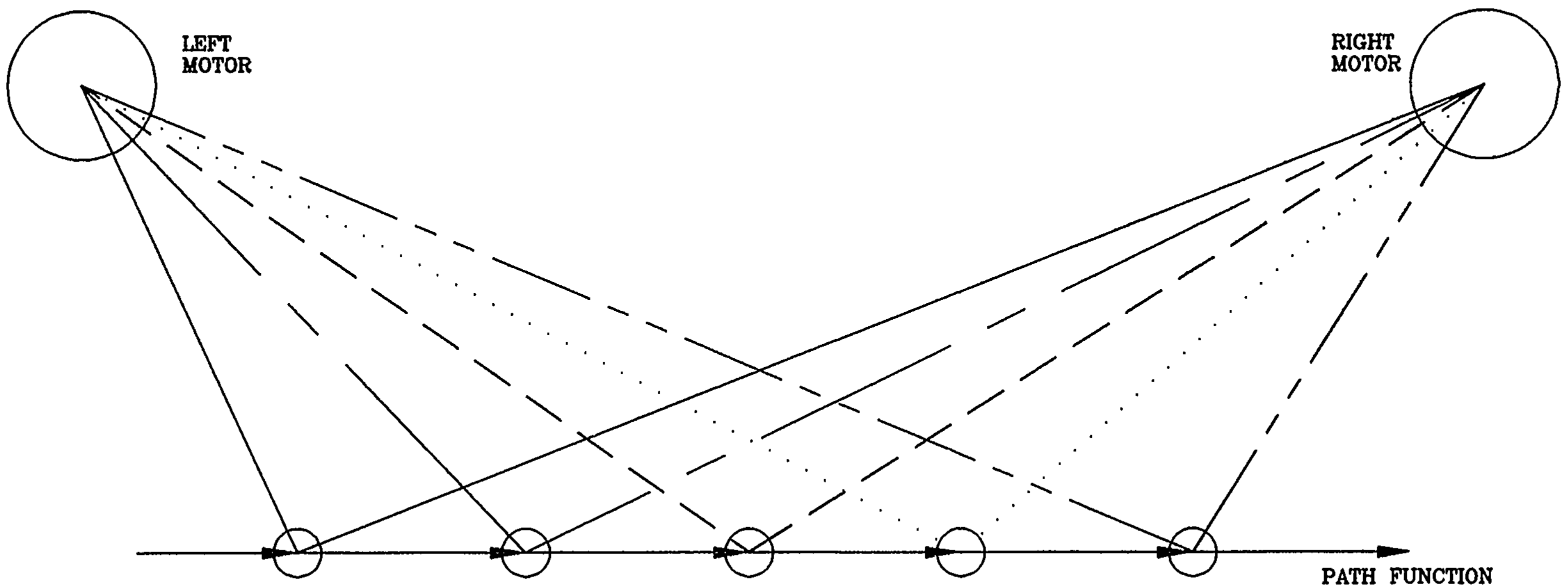
Cavity decontamination is typically completed just after the reactor has been refueling and the reactor cavity water is simultaneously being drained. Access to these 25' to 40' high walls is typically limited to the top of the cavity. This restricted access makes manual decontamination a time-consuming and risky exercise. By employing a vehicle that only operates on the wall surface, the unnecessary time and risk can be virtually eliminated.

The WALLWALKER can be mounted with scrubbers, high-pressure water jetting, CO2 blast, or even scabbling techniques with vacuum attachments for concrete surface decontamination. Complete system operation is implemented by an off-board computer with sensory input and control output provided via tether.

This presentation, complemented by a video demonstration, will discuss the design criteria of the WALLWALKER, its demonstration successes, and the various potential applications of this new robot.

Speaker Biography

Sheldon Lefkowitz is President of Pentek, Inc., located in Corapolis, Pennsylvania. Originally from Queens, New York, Mr. Lefkowitz earned a Bachelor's Degree in Mechanical Engineering from the Cooper Union in 1973, and a Master's Degree in Mechanical Engineering from M.I.T. in 1974. He has worked as a heat transfer design engineer with General Electric, a nuclear systems safety analyst for the California Energy Commission, and a consulting engineer engaged in nuclear power plant safety and licensing projects. Mr.



Pentek's new wall servicing robot, depicted above, functions by precisely controlling the position of a work point through the use of simple triangulation calculations. As can be seen, two (2) computer controlled motors attached to the upper corners of the pool walls direct their respective reeving systems to extend or retract the two (2) working cables; hence, the lengths of these cables are always known precisely (to within 25ppm). The distance between the motors is fixed, and is known as precisely as the positions of the motors can be measured in the field. Therefore, the three (3) sides of the triangles which are used to define position over a specified path function are known to great precision; certainly within the tolerances required for the conduct of remote decontamination operations.

Lefkowitz started Pentek in 1981, shortly after the accident at Three Mile Island, and has been engaged in the decontamination and handling of hazardous materials for the past 13 years. Pentek is engaged in the manufacture of lead paint abatement systems, and performs hazardous material abatement services in all regions of the country and around the world.

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PAPER 11-4
DISCUSSION

Mei: The picture that you showed us is more like a very flat wall. Can the robot also handle curved wall surfaces or if the wall is in and out?

Lefkowitz: Yes. A curved wall is treated as a series of arc segments. The work point is moved across the face of the wall along the plane of the chord or tangent line. Within limits, the work point is held at a small but varying distance from the surface. If the surface of the wall is in and out (i.e., the wall is not flat), then the load path is established at a maximum distance from the wall, but sufficient to travel over the closest obstruction.

Mei: Can you keep a constant distance from the wall?

Lefkowitz: Absolutely. If greater control of standoff distance is required, the control of a third dimension (into and away from the wall) can be accommodated by a local positioning mechanism attached directly to the work point.

THE INTEGRATED RADIATION MAPPER ASSISTANT

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ABSTRACT

The Integrated Radiation Mapper Assistant (IRMA) system combines state-of-the-art radiation sensors and microprocessor based analysis techniques to perform radiation surveys. Control of the survey function is from a control station located outside the radiation thus reducing time spent in radiation areas performing radiation surveys. The system consists of a directional radiation sensor, a laser range finder, two area radiation sensors, and a video camera mounted on a pan and tilt platform. This sensor package is deployable on a remotely operated vehicle. The outputs of the system are radiation intensity maps identifying both radiation source intensities and radiation levels throughout the room being surveyed. After completion of the survey, the data can be removed from the control station computer for further analysis or archiving.

INTRODUCTION

A significant advancement in remote radiation surveying is in the testing and evaluation phase of the development effort. At the conclusion of the this phase, the findings from the testing will be evaluated and the commercial product developed. The **IRMA** System, being developed by Odetics, Inc. and three utility companies is an attempt to reduce radiation exposure to plant personnel by performing radiation exposures remotely, while providing a more comprehensive set of data for use in the planning and execution of plant activities.

This testing focuses on determining the accuracy of the new sensor package, the versatility of remote controls designed into the system, and the flexibility of the data acquisition and display subsystem.

SYSTEM OVERVIEW

The **IRMA** system concept uses a lightweight sensor package mounted on a mobile platform that is remotely maneuvered for mapping radiation sources and intensities in enclosed areas.

Major elements of the system include:

1. Sensor package
2. On-board electronics
3. Pan and tilt platform
4. Fiber optic data link and tethered control line
5. Control workstation
6. Robotic vehicle

Two enclosures - one for sensors and one for electronic controls - are mounted on a pan and tilt platform, which are mounted on a robotic vehicle. The vehicle is tethered to a personal computer-based workstation, which controls vehicle movement and pointing of the radiation sensors, while processing and displaying sensor acquired data.

The primary technical breakthroughs of IRMA are the reduction in weight and size of individual components, the integration of data acquisition, robotics controls, and data display systems.

SYSTEM COMPONENTS

Sensor Package

The sensor package contains all of the sensors used by the **IRMA** system. Two types of radiation sensor are - a directional sensor with a narrow field of view for pinpointing radiation sources and two wide area sensors for measuring the intensity of the local gamma radiation field. A bismuth germinate (BGO) crystal, measuring 1.27 cm in diameter and length, is the scintillator used for directional radiation sensing. It is optically bonded to a photomultiplier tube (PMT) encased in a specially designed tungsten collimator. The wide-area sensor assembly consists of an identical BGO crystal and a plastic scintillator of the same dimensions, each bonded to a PMT. Each type of sensor responds differently to incident gamma ray energy, and a comparison of their outputs yields information about the gamma field energy.

Two lasers are incorporated in the package. One is a highly-accurate range finder that feeds distance information from the location of the radiation source to the **IRMA** sensor package. The distance between the sensor and a radiation source is essential input to determining source strength. A second low-power laser points a red pointer beam in the same direction as the radiation and range finding sensors so the operator can visually identify the precise location of the readings taken by the **IRMA** system.

Viewing of the pointing laser and other visual feedback are provided by a color video camera on the sensor platform. A motorized zoom and focus lens allows the operator to closely examine specific objects in the room and to assess the general physical conditions of the area being surveyed.

On-board Electronics

A computer in the sensor platform controls the use of each sensor. This processor is a NEC V53 operating at 16MHz on a STD bus. This processor that initiates data acquisition, controls the pan and tilt platform motors, and interfaces with a fiber optic multiplexer to transmit both video and data signals to the control station. In addition to the control computer, the electronics enclosure contains a counter board, power supplies, motor servo boards and the fiber optic multiplexer.

The power supply and detector amplifier and digitizer boards are located on the back of the radiation sensor enclosure. Detector gain and power supply voltage are controlled by adjusting potentiometers inside the electronics box. Cabling from the detector electronics to the control computer transmits the detector output signals. Sensor and electronics enclosures are stainless steel to aid in the decontamination of exterior surfaces.

Pan and Tilt Platform

The pan and tilt platform, with an exceptionally high payload-to-weight ratio, was developed for the **IRMA** system. The platform weight 14 pounds, yet it can position the 66-pounds sensor payload to within 0.1 degrees repeatedly.

Constructed of honeycomb aluminum encased in a thin steel skin, the platform's range of motion is +/180 degree azimuth and -60 to +90 degrees in elevation. The drive mechanism is embedded in the platform's "L" shaped form. It is designed as a self-sufficient module that may be removed from the vehicle for use on another vehicle or installed in an area on a long term basis.

Fiber Optic Data Link and Tethered Control Line

The reinforced, 62 meters long fiber optic link provides a high bandwidth bidirectional link over one optic fiber. An electro-optical converter at the control station decodes up to three bidirectional data channels, one unidirectional video channel and one bidirectional audio channel.

The fiber optic cable is deployed and recovered from a continuously-turning reel on the robot. The direction of the vehicle determines the direction of rotation for the fiber optic reel. A strain guage on the tether senses the direction of tether demand and directs the tether handler motor to turn in a direction to either pay out or take up the tether cable. Tether management is automatic and designed to minimize cable contamination by reducing the amount of cable dragging on the floor. Two electrical conductors enclosed in the same cable provide power to the sensor package, tether handler and trickle charge of the robotic vehicle batteries

Control Station

The control station is based on a 33-MHz 80386 processor with 8 megabytes of RAM, 128K cache memory, a 200 MB hard drive and two floppy drives. A 1024 x 768 VGA monitor with touchscreen capabilities serves as the video and data display terminal at the control station. A video adaptor allows real-time digital video processing with freeze frame capability to capture images from the platform camera.

The DOS system uses Microsoft Windows™ as the user interface. While users may define individual display configurations, the general intent is to provide a video window, status windows, control windows, and the ability to monitor certain types of data presentations during operation. Control actions are entered by touchscreen, and input parameters are entered by the keyboard feature. The primary purpose of the control station is to acquire data for later processing.

Mobile Vehicle

The robotic vehicle used in the prototype system uses an innovative means of locomotion. Four clusters of three wheels each are in a circular pattern about a common axis. All 12 wheels are continuously driven. Each cluster has two lower wheels on the ground, which propels the vehicle. When the vehicle encounters an obstacle, the top wheel is brought into use to engage the obstacle and lift the vehicle over. Limited movement over berms and pipes is possible with limited operator intervention.

The front and rear wheel clusters on each side of the vehicle connect by a drive chain that connects to a drive motor through a reduction gear. Each side of the vehicle is independently driven and controlled to allow skid steering. The vehicle's speed, steering, and direction if controlled by a joystick at the control station. Collision avoidance sensors alert the control station operator to impending collisions.

SYSTEM USAGE

Scanning Process and Data Display

Once the **IRMA** system is positioned in a room, a series of visual scans are necessary to orient the location of the **IRMA** sensor relative to the boundaries of the room. Using the cursor, the operator can select a specific area in the field of view of the camera up to and including the entire field of view. **IRMA** then automatically calculates scanning steps, begins the data acquisition process, and captures a mosaic of the area in video images. Any number of scans may be ordered by the operator, and they are identified and stored in sequential order.

The operator can select a data display and IRMA will calculate the intensity of the radiation source for each data point for which valid data is obtained. This display is in the form of a translucent overlay on the visual image of the camera's field of view when the survey was taken. The operator can vary the intensity of the color overly from invisible to almost opaque depending on his preference at the time of viewing. Software is provided to capture these overlaid images and print them out on hard copy with a color printer.

In addition to the visual radiation image that **IRMA** generates, the processed radiation intensity data is stored in an ASCII format. In this format, the data is accessible by many standard spreadsheet and data base programs for further analysis. Spreadsheet macros are being developed to provide radiation contour maps and other types of data displays.

Information Generated

The information generated by the **IRMA** system is different from traditional health physics information. **IRMA** measures the gamma surface flux from a source within the field of view of the collimated detector. This measurement is repeated for all such source areas within the scanned field of view. In some instances of localized sources within a small area, this gamma surface flux value can be interpreted as the localized contact radiation level. In most instances, this gamma surface flux value as determined by **IRMA** cannot be directly correlated to the localized contact radiation level due to the contribution to the localized contact radiation level from other radiation sources within or outside of the field of view of the collimated detector. In essence, the **IRMA** output is a mapping of radiation source intensities by specific location relative to the sensor that can, in turn, be processed outside of the **IRMA** data collection system to assist in the prediction and analysis of radiation levels in plant areas mapped.

APPLICATION ENVIRONMENT

The **IRMA** development was initiated in 1990 as a cooperative venture of Odetics, Public Service Electric and Gas, New York Power Authority, and Southern California Edison. The first application of the **IRMA** system is to assist health physics personnel within the nuclear generating facility. Additional applications will be developed in other areas of the nuclear industry as needs arise.

Radiological surveys are performed to provide precise information about work areas to ensure that any worker entering the area will not be exposed to levels beyond the occupational radiation exposure limit. One of benefits of **IRMA** is to reduce the exposure to health physics technicians who conduct those surveys. Historically, the health physics technicians receive higher radiation exposures than most radiation worker groups[1]. Methods are continuously being sought to reduce their exposure.

Reduced exposures attributable to the survey function is not the only benefit that can be derived from an IRMA type of system. Other potential benefits include:

1. Accurate trending of radiation levels throughout the plant including the trending of the development of "hot spots".

2. More accurate determination of the effectiveness of radiation level reduction techniques such as flushing or chemical decontamination.

3. Prevention of the increase in radiation fields through the early accurate identification of such increases and the specific sources contributing to them.

4. Better planning of individual work tasks and task sequencing to reduce the time spent near radiation sources.

5. Effective use of Health Physics technicians, freeing them up for tasks other than manual surveying of plant areas.

6. More precise locatiou of "cool" spots in the work area to retire to during period of work inactivity.

7. Better identification of the location of radiation sources to allow for more accurate placement of temporary shielding and reduction in the amount of unnecessary shielding being used due to lack of accurate location of the radiation sources.

SYSTEM PERFORMANCE

Part of the effort of the prototype testing is to evaluate the performance of the overall IRMA system. In actual practice, the performance of the IRMA system will vary with the background radiation, the source strength and the distance from a specific source intensity. The performance of an IRMA type of system is a series of tradeoffs of size, weight and sensitivity, the selection of the parameters for the prototype unit was made to minimize the size and weight of the unit while achieving a level of sensitivity believed necessary for a nuclear power facility. Table 1 provides a summary of the pertinent data and observed performance of the prototype IRMA system.

Table 1

IRMA Data and Performance Summary

Collimated Radiation Detector	
Field of View	8 degrees nominal
Sensing Element	Bismuth Germanate(BGO)
Photomultiplier Tube	Hammatsu 647-04
Collimator Material	Tungsten Alloy HD-17
Environmental Enclosure	Type 304 SS.
Electronics	Custom
Size	11"l x 5"w x 5.25"h 28 cm x 12.7 cm x 5.7 cm
Weight	65 lbs. (29.5 kg.)
Dynamic Range	1 mrad/hr to 100 rad/hr 10 μG/hr to 1 G/hr
Lower Limit of Detection(equivalent point source)	1 mCi @ 20 feet 1 mr/hr Cs-137 background; 5 mCi @ 5 feet, 100 mr/hr background.
Area Radiation Sensors	
Sensing Element -- Low Range	Bismuth Germanate(BGO)
Sensing Element -- High Range	Plastic -- Bicron BC 400
Photomultiplier Tubes	Hammatsu 647-04
Dynamic Range	0.5 mrad/hr to 100 rads/hr 5 μG/hr to 1 G/hr
Pan and Tilt Platform	
Size	15"l x 18"w x 20"h 38 cm. x 46 cm. x 61 cm.
Weight	25 lbs. (11.4kg.)
Payload	70 lbs. (31.8 kg.)
Speed	10 deg. per sec.
Accuracy	0.1 deg.
Materials of Construction	Nickel plated Al. Alloy & coated carbon steel

Table 1 (Continued)

IRMA Data and Performance Summary

Laser Range Finder

Size	7"l x 4.5"w x 2.8"h 17.8 cm x 11 cm x 7.1 cm
Weight	2.6 lbs. (1.2 kg.)
Range	0.65 feet to >330 feet 0.2 m. to >100 m.
Accuracy	± 2" (±5 cm.)
Resolution	1 cm.
Power Consumption	0.75 A @ 12 V. approx.

Video Camera

Color CCD	
Remote Focus and Zoom	6:1 Zoom
Camera	
Size	6.7"l x 2.5"w x 2.6"h 17 cm x 6.3 cm x 6.5 cm
Weight	1.34 lbs. (0.61 kg.)
Power	5 W @ 12 V
Lens	
Size	3.5"l x 3.7"w x 3.4"h 8.9 cm x 9.5 cm x 8.6 cm
Weight	1.36 lbs. (0.62 kg.)

REFERENCE

1. L.T. Davis, et al., "Job Analysis of Nuclear Power Reactor Health Physics Technicians," NUREG/CR-3750, BNL-NUREG-51769, Brookhaven National Laboratory, Upton, N.Y., August, 1984.

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STRATEGY PROPOSED BY ELECTRICITÉ DE FRANCE IN THE DEVELOPMENT OF AUTOMATIC TOOLS

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The strategy proposed by EDF in the development of a means to limit personal and collective dosimetry is recent. It follows in the steps of a policy that consisted of developing remote operation means for those activities of inspection and maintenance on the reactor, pools bottom, steam generators (SGs), also reactor building valves; target activities because of their high dosimetric cost.

One of the main duties of the UTO (Technical Support Department), within the EDF, is the maintenance of Pressurized Water Reactors in French Nuclear Power Plant Operations (consisting of 54 units) and the development and monitoring of specialized tools. To achieve this, the UTO has started a national think-tank on the implementation of the ALARA process in its field of activity and created an ALARA Committee responsible for running and monitoring it, as well as a policy for developing tools. This point will be illustrated in the second section on reactor vessel heads.

EDFs CURRENT ALARA STRATEGY

The objective of EDF's ALARA strategy is the adequacy between the reduction in collective dosimetry and its cost, with the priority of reducing exposure of personnel with the highest personal dosimetry.

The ALARA strategy breaks down into two distinct cases, depending on the type of maintenance operation.

FIRST CASE

In the case of a maintenance operation of a repetitive nature, therefore that could be contracted out, EDF draws up the Technical Specifications of the operation and a compulsory dosimetric results clause. The contractor(s) develop(s) the equipment and procedures required to accomplish this objective. the main aim of this process is to make the contractor responsible for achieving the best personal and collective dosimetric results possible, and a financial encouragement would be quite feasible on achieving certain objectives. The main obstacle for EDF remains the most precise evaluation of forecasted dosimetry that needs on-site mapping of irradiation rates in the various worksites, and analysis of forecasted operator dosimetry. To do this, experience feedback and worksite monitoring are necessary. Processes of this type are underway for Vessel head opening/closing operations, fuel pit decontamination, cleaning of the secondary side of SGs and primary valve maintenance.

SECOND CASE

In case of an exceptional maintenance operation, the investment required to develop the adapted means may be beyond the contractor's financial means. The UTO assumes the costs of developing the tool and makes it available to the contractor, who shall ensure that it is used correctly. The ALARA attitude of the UTO is the same as in the first case and a partnership with the user enterprise optimizes the tool as regards dosimetry reduction.

Examples of several developments underway:

- Tool for inspecting and aiding transfer device alignment
- Removal of foreign bodies from the SG secondary side
- New generation SG nozzle dams
- Decontamination by soft chemistry.

APPLICATION TO OPERATIONS ON VESSEL HEADS

After the discovery of a penetration defect in the vessel head of Bugey 3 in 1991, EDF, backed by nuclear industrialists, investigated to find the origin and extent of the problem throughout all nuclear plants in France.

Approved processes were employed for the inspections and repairs performed in 1992. They were in part implemented with semi-automatic means developed beforehand for exceptional operations of an assurance nature (dismounting CRDM), but also and often with means requiring the presence of operators under the vessel head, consequently, exposed to high doserates (approximately 50 man Sv/h).

As a result of means generally ill-adapted to operating conditions, a lot of time was spent in 1992 on vessel head worksites that accounted for about 4% of non-availability in the EPN, and substantial collective dosimetry (9 man Sv) during these worksites.

The vessel head phenomenon proved to be a generic affair in 1992. A specific ALARA organization was created for this affair as a result of the massive collective doses in the first few months:

September 91 to December 91: 2 man Sv

An inspection program requested by the Safety Authorities was made possible by the development of technically adapted and robotized means to reasonably reduce operator doserates.

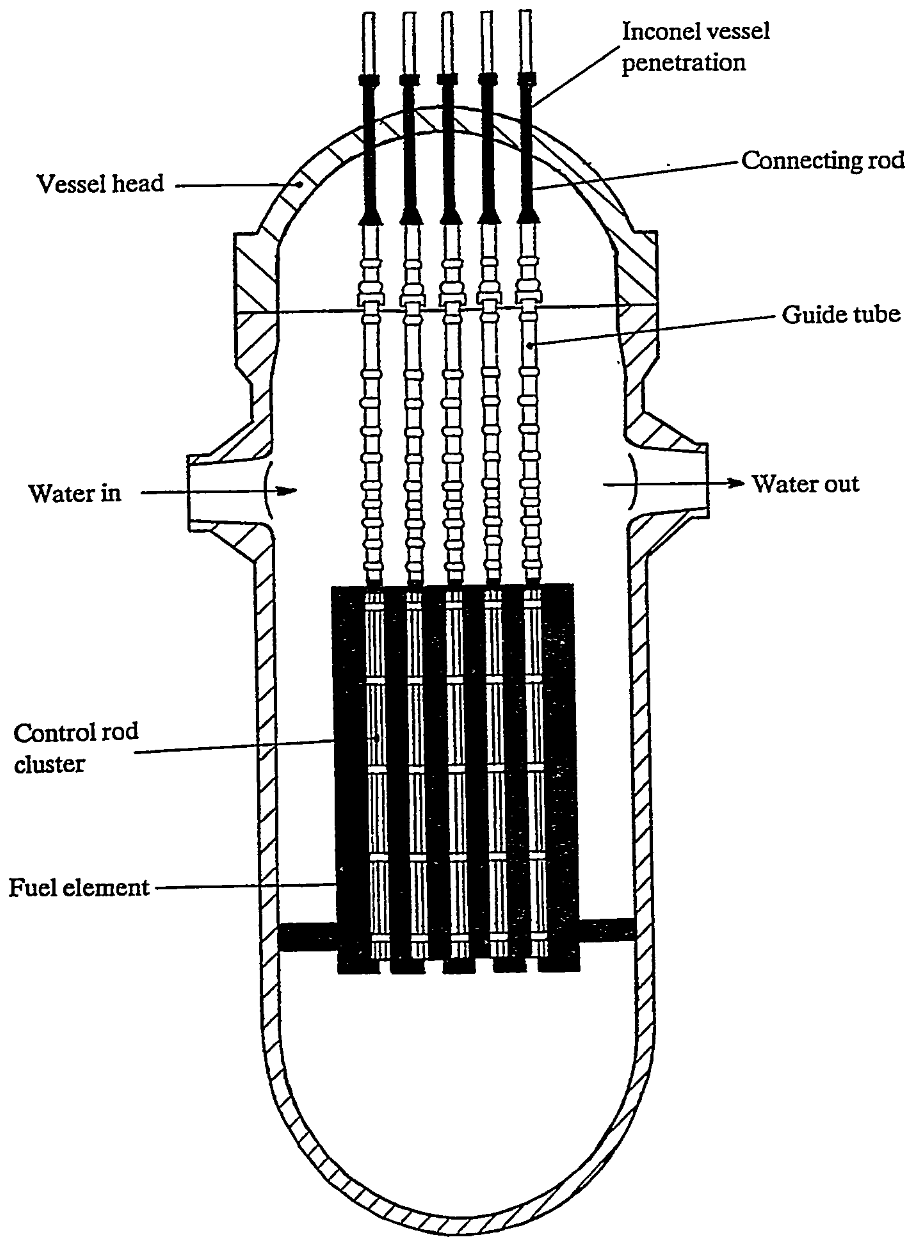
This would explain why EDF, the constructor FRAMATOME and French (ACB, COMEX, INTERCONTROLE, etc.) and foreign (ABB, etc.) contractors made substantial efforts in 1992 to rapidly and urgently develop and research the best means and procedures to perform both inspections and repairs in theoretically subcritical time as regards unit outage, with minimum dosimetry (automated or robotized means keeping human intervention under the vessel head to a minimum).

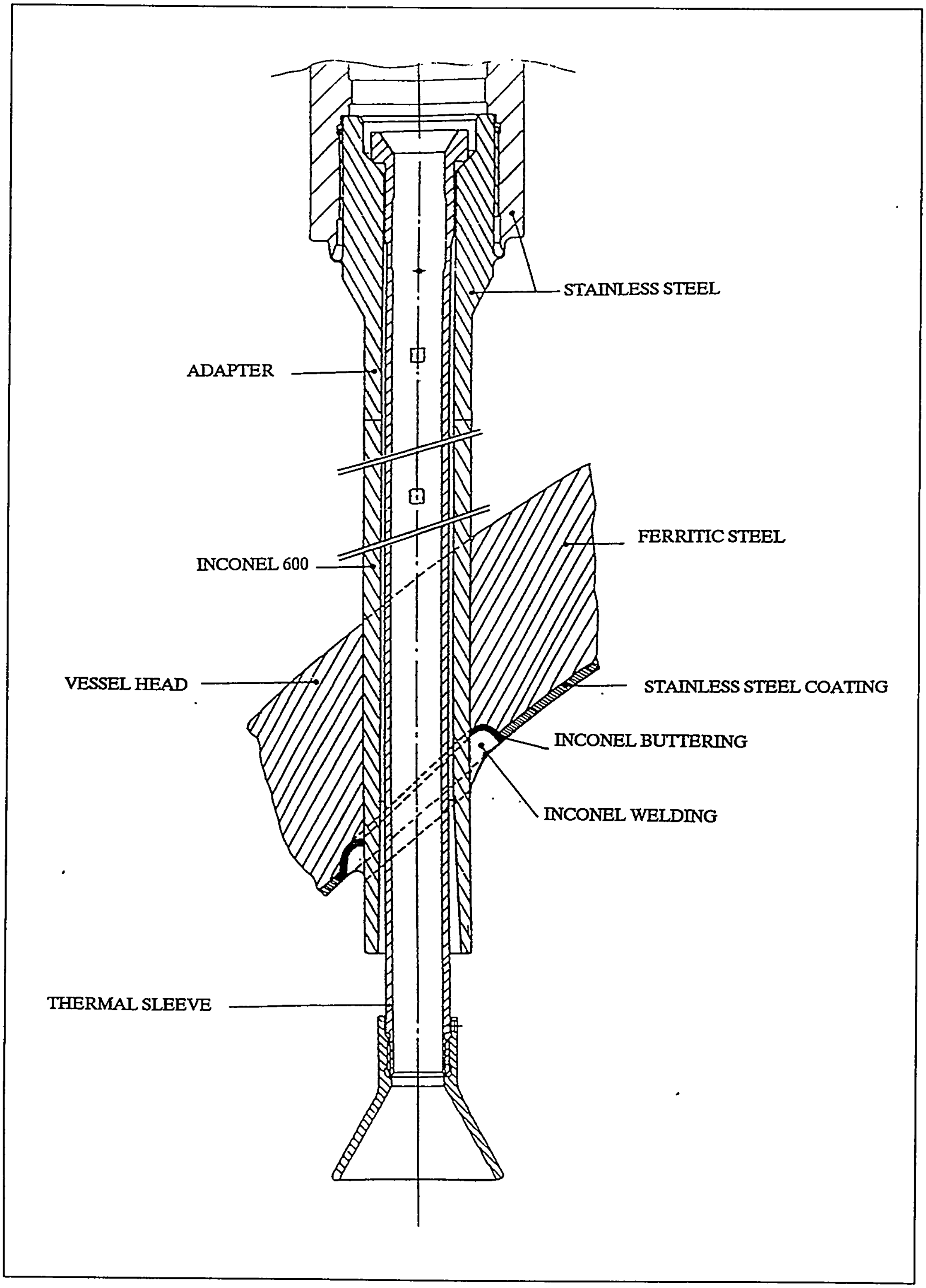
Robotized means were implemented in 1993 throughout all sites in a major inspection program. The intention is to establish a zero point status in all units affected by this problem during 1993 and 1994.

DESCRIPTION OF THE PHENOMENON

Vessel heads in pressurized water reactors have 65 to 78 penetrations for the passage of the CRDM drive shaft tubes and mounting of the motor drive mechanism (Figure 1). In French reactors, this penetration is a tube with an internal diameter of 70 mm, 16 mm thick. The lower section ends in a cylindrical or conical flaring depending on the reactor (Figure 2). Inside, there is a thermal sleeve made of austenitic steel ending in a

FIGURE 1
LAYOUT OF THE CONTROL ROD ADAPTERS ON THE VESSEL HEAD




STAINLESS STEEL
ADAPTER
FERRITIC STEEL
INCONEL 600
VESSEL HEAD
STAINLESS STEEL COATING
INCONEL BUTTERING
INCONEL WELDING
THERMAL SLEEVE

guide in the shape of a trumpet. The space between the sleeve and the adapter is approximately 2 mm. It is locally reduced in certain penetrations by the adapter oval shape. In most reactors throughout the world, this adapter is made of a nickel alloy, known as Inconel 600, which has proved to be vulnerable to stress corrosion cracking in primary water. The cracks develop in proportion to rising temperature under the dome. Initiating stress, result from deformation of the adapter under the effect of welding on the vessel head. This deformation is increased and asymmetric for the adapters on the edge of the head because they are at an incline compared to the surface.

It has been proved today after numerous inspections, that the predominant influence of residual stress has caused longitudinal cracks concentrated along two opposing lines of the tube.

CHANGES IN MEANS AND DOSIMETRY

Given that there is a 2 mm space between the thermal sleeve and the adapter, the control rod mechanism and the sleeve have to be dismounted and cut open.

These long and costly operations, using first generation, often manual tools resulted in high doserates : 0.44 man Sv in the inspection of one 900 MW vessel head.

Cracks are detected by ultrasonic means (depth - position in relation to the weld) and sometimes, repairs by excavation of the cracked zone and then build-up welding. These entirely manual repairs on the first operations required the physical presence of an operator under the head, resulting in high doserates.

These tools and techniques have been used to inspect and repair several heads, it is quite inconceivable, because of the high doserate generated, to use them throughout the EPN plants.

Detection Test Using an Eddy Current Detector

Development of inspection robots by industrial groups like ABB and FRAMATOME, has enabled an eddy current detector to be inserted between the adapter and the thermal sleeve to detect any cracks (Figure 3). This has considerably helped reduce dosimetry by robotizing manual tasks (no personnel under the head) and by getting rid of costly operations, in terms of doserate, on the rod mechanism and thermal sleeves (dismounting and remounting).

In fact, all vessel head adapters are inspected by eddy current and only 3% approximately require ultrasonic testing and eventually repairs.

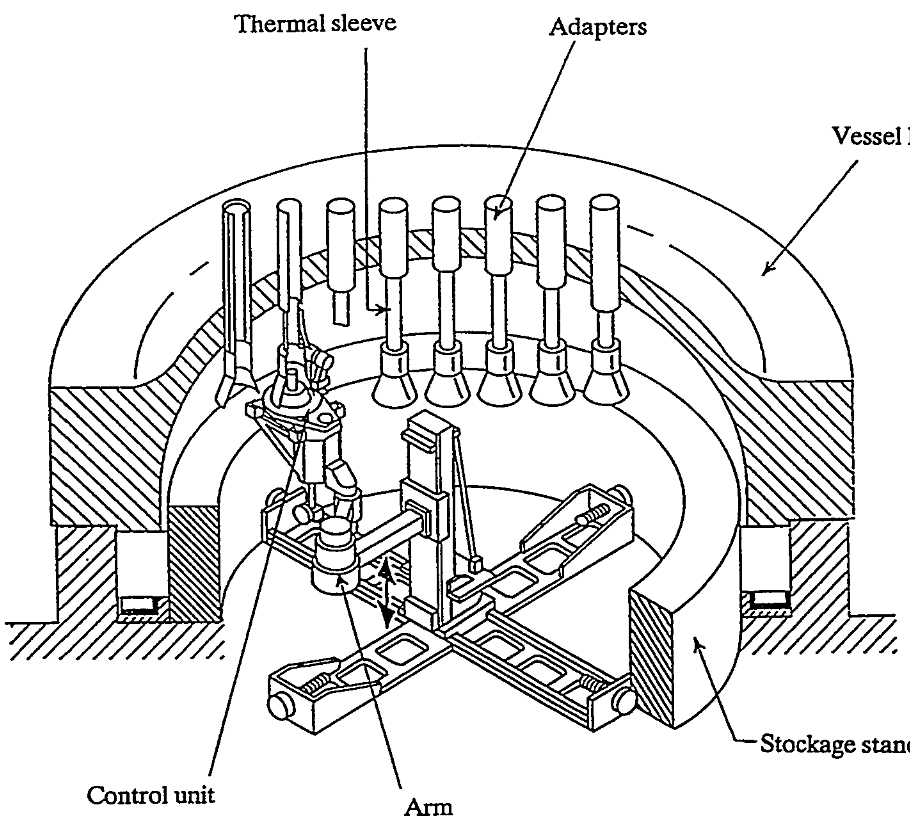
The dosimetry count for eddy current detection of a vessel head has now been improved from 0.44 man Sv to 0.04 man Sv.

Adapter Repair

Adapters were first repaired by excavation and manual welding, performed under the vessel head inside biological protection. FRAMATOME has since developed a series of automatic machines that are manually positioned by an operator under the vessel head, simply to connect the machine to the adapter.

The average dosimetry count for such an operation was improved by 0.2 man Sv or so for the first manual operations to 0.08 man Sv for automatic operations.

ROBOT OF CONTROL UNDER HEAD



Characterization of Adapters with Sabre Ultrasonic Detector

On all adapters controlled by eddy current detectors, about 3% had cracks that needed to be determined (depth of the crack) by ultrasonic means. In 1993, this operation required dismounting and remounting of control rod mechanisms and thermal sleeves. To avoid these high doserate operations, COMEX, ABB and FRAMATOME suggested defining cracks using a sabre type tool carried by a robot or tank capable of inserting an ultrasonic detector into the 2 mm space between the thermal sleeve and the adapter. This operation, totally remote controlled, allows an economy of 80 man mSv for first-of-a-kind mechanism dismounting and 15 man mSv thereafter.

Dismantling and Remounting of the Thermal Sleeve

Dismantling of control rod mechanisms is only justified when removing a thermal sleeve. If non destructive testing does not any more require this operation, repairs do require it, in fact, it would be difficult to design excavation and welding tools that fit into the 2 mm space available.

FRAMATOME and COMEX are currently developing tools that will remove, from under the vessel head, the lower section of the thermal sleeve thereby giving access to the adapter and also reweld this part of the sleeve onto the section that remained in place. This entire operation is remote controlled and no human intervention is required under the vessel head.

This service will be operational by the middle of 1994, it will be the last in a series of developments that aim at reducing operator dosimetry to a minimum by means of automated operations adapted to the environment and extent of the problem to be handled.

SERVICE CONTRACTS AND DOSIMETRY

Reducing dosimetry is a matter of making contractors in charge of the operation aware of the problem. In addition to the legitimate tendency of industrial groups to avoid exposing their personnel, EDF is determined to go even further, by including obligations or incentives to reduce doserates in the service contracts.

Dosimetry Planning

Each contractor with a service contract concerning reactor vessel heads is obliged to propose a detailed exposure forecast prior to the operation.

The "DOSIANA" software enables operations to be broken down into elementary tasks, each task is attributed a theoretical dose. During the operation, the task is monitored. This enables highly penalizing operations to be indicated, where it is necessary to make efforts to reduce doserate, and to detect the differences between real and theoretical doserates so the undefined part of the operation can be quantified and analyzed.

Financial Incentives to Reduce Doserate

Solving the problem of reactor vessel heads sometimes means replacing the head. Such an operation requires dismounting of the entire control rod mechanism and remounting it on the new head. An initial operation with existing tools, developed to dismount and remount several units, therefore not totally adapted to the replacement of a complete vessel head, generated 600 man mSv.

On the assumption of more heads needing to be replaced, EDF negotiated a financial package with the industrial contracting group in exchange for a substantial reduction in recorded doserate. The target doserate, that is now in the contract, is 260 man mSv, if this quantity is overshot, there will be no financial penalty, but it will be considered as nonobservance of a contractual commitment on behalf of the industrial group, and therefore, a reason for EDF to possibly break the contract.

WORKSHOP CLOSING

Bennett (Chair): I would like to take the opportunity to congratulate the Brookhaven staff for providing such an enjoyable, friendly, and well organized conference. Thank you from all of us.

Khan: Thank you, Margaret. On behalf of the Nuclear Regulatory Commission and the Brookhaven National Laboratory ALARA Center, the co-sponsors of this workshop, we thank all speakers for their excellent presentations, we thank all chairpersons for helping us to run this workshop, and we wish all participants a happy and safe journey home. I would like to end with the words of William Shakespeare, "Farewell, farewell dear friends. If we meet again we shall smile indeed. If not, then this parting is well made." The Third International Workshop on ALARA is now closed.

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NRC FORM 335
(2-89)
NRCM 1102,
3201, 3202

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BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC. Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CP-0143
BNL-NUREG-52440

2. TITLE AND SUBTITLE

Proceedings of the Third International Workshop on the Implementation of ALARA at Nuclear Power Plants

Held at Hauppauge, Long Island, New York

3. DATE REPORT PUBLISHED

MONTH	YEAR
March	1995

4. FIN OR GRANT NUMBER

A3259

5. AUTHOR(S)

Compiled by T. A. Khan

6. TYPE OF REPORT

Conference Proceedings

7. PERIOD COVERED *(Inclusive Dates)*

May 8-11, 1994

8. PERFORMING ORGANIZATION – NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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9. SPONSORING ORGANIZATION – NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This report contains the papers presented and the discussions that took place at the Third International Workshop on ALARA Implementation at Nuclear Power Plants, held in Hauppauge, Long Island, New York from May 8-11, 1994. The workshop brought together scientists, engineers, health physicists, regulators, managers and others who are involved with occupational dose control and ALARA issues. One-hundred and seventy five persons from eleven countries attended the workshop. The countries represented were: Canada, Finland, France, Germany, Japan, Korea, Mexico, the Netherlands, Spain, Sweden, the United Kingdom and the United States. The workshop was organized into twelve sessions and three panel discussions. The topics were as follows: Session 1, Controlling Radiation Fields; Session 2, Panel Discussion on Recent Recommendations on Dose Limitation; Session 3, Presentations and Panel Discussion on ALARA in New Reactors; Session 4, Pathways to ALARA; Session 5, Panel Discussion on Economics Versus Excellence; Session 6, Short Presentations on ALARA Implementation; Session 7A, PWR and CANDU Presentations; Session 7B; BWR and Gas-Cooled Presentations 1; Session 8A, PWR and CANDU Presentations; Session 8B, BWR and Gas-Cooled Presentations; Session 9, Decommissioning of Nuclear Power Plants; Session 10, Decontamination of Nuclear Power Plants, and Session 11, Robotics and Remote Handling. The workshop was sponsored jointly by the U.S. Nuclear Regulatory Commission and the Brookhaven National Laboratory's ALARA Center.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

ALARA - MEETINGS, BWR TYPE REACTORS - ALARA, CANDU TYPE REACTORS - ALARA, GAS COOLED TYPE REACTORS - ALARA, PWR TYPE REACTORS - ALARA, NUCLEAR INDUSTRY - OCCUPATIONAL EXPOSURE, BNL, Design, Dosimetry, International Organizations, Leading Abstract, Modifications, Occupational Safety, Radiation Doses, Radiation Protection, Regulations, Water Chemistry

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)
Unclassified

(This Report)
Unclassified

15. NUMBER OF PAGES

16. PRICE