NUREG/CP-0166
Vol. 1

Proceedings of the Twenty-Sixth Water Reactor Safety Information Meeting

Volume 1

- **Plenary Sessions**
- **Pressure Vessel Research**
- **Severe Accident Research, Fission Product Behavior**
- **Nuclear Materials Issues and Health Effects Research**
- **Materials Integrity Issues**

Held at
Bethesda Marriott Hotel
Bethesda, Maryland
October 26–28, 1998

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research

Proceedings prepared by
Brookhaven National Laboratory

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Manuscript Completed: May 1999
Date Published: June 1999

Compiled by: Susan Monteleone

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Office of Nuclear Regulatory Research
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Washington, DC 20555-0001

Proceedings prepared by
Brookhaven National Laboratory

ABSTRACT

This three-volume report contains papers presented at the Twenty-Sixth Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel, Bethesda, Maryland, October 26-28, 1998. The papers are printed in the order of their presentation in each session and describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included papers presented by researchers from France, Germany, Italy, Japan, Norway, Russia, Sweden and Switzerland. The titles of the papers and the names of the authors have been updated and may differ from those that appeared in the final program of the meeting.

PROCEEDINGS OF THE
26TH WATER REACTOR SAFETY INFORMATION MEETING

OCTOBER 26-28, 1998

Published in Three Volumes

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PROCEEDINGS OF THE
26TH WATER REACTOR SAFETY INFORMATION MEETING
OCTOBER 26-28, 1998

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26TH WATER REACTOR SAFETY INFORMATION MEETING

October 27, 1998

Ashok C. Thadani, Director
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission

INTRODUCTION

Good Morning Ladies and Gentlemen.

For those of you who do not know me, my name is Ashok Thadani and I am the Director of the Office of Nuclear Regulatory Research. On behalf of the U.S. Nuclear Regulatory Commission, I am pleased tc welcome you this morning to the 26th Water Reactor Safety Information Meeting. This year we have a three day program which will focus on many of the nuclear safety issues that we are facing today.

All of us here pursue a common goal, that of nuclear safety. And your presence here means that your organization shares a common value that the pursuit of knowledge through reactor safety research and experimentation should continue despite the pressures of budgetary constraints and downsizing. We share this value because we all know that research is essential to assuring safe use of nuclear technology because good science leads to good decisions - both in terms of safety as well as resource allocation. Your attendance here in times of declining budgets and international monetary strife speaks to the importance that your organization places in the belief that developing and sharing technical knowledge and understanding are essential to the safe and efficient use of nuclear technology.

But we cannot take anything for granted. Just because we share the same concerns and have drawn the same conclusions does not mean that others have also done so, even in our own organizations and governments. We are in a changing regulatory environment which is challenging us as researchers to be more pro-active, outcome oriented and cost-effective. We must use risk-informed thinking throughout our programs to effect safety improvements and reduce regulatory burden. This will result in significant changes in the way we conduct business. In a few minutes Chairman Jackson will share her thoughts on transitioning to risk informed and performance based regulation, a major departure from the NRC's past largely prescriptive regulatory philosophy. The nuclear industry is undergoing economic deregulation and restructuring which will increase pressures on them and us to use risk insights and other research to gain efficiencies. Also, there is increased stakeholder interest in research programs, a phenomenon which promotes increased cooperative interactions. Looking inside, we must synchronize our research programs with agency needs for safe and efficient regulation. Looking out, those of us who are regulators must be more mindful of industry and international efforts so that we can use cooperative research programs to achieve mutual goals. Lastly, and in summary, we must be more accountable-continued resource constraints and public scrutiny and legislation require us to focus on outcome measures which will gauge our performance and tell whether our research efforts are "on the mark." I am not daunted by these challenges. In fact, I am energized that you and I can all make a difference, whether it be in extending the safe life of our existing plants, or in licensing a new generation of future plants, or assuring that plants continue to be operated in a safe manner.

I look forward to this year's plenary sessions and the presentations of papers. You will find the papers to be topical and representative of the leading edge of nuclear safety research. Today, our NRC Chairman will speak on "The Transition to Risk-Informed Regulation: the Role of Research," and NRC Commissioner Diaz will speak after lunch today on "Light Water Reactor Safety: The Convergence Process." I am especially pleased to have a separate session on Tuesday afternoon covering the full Halden Program and its Organization for Economic Cooperation and Development Program Director Carlo Vitanza will speak about Halden at the Tuesday luncheon. A new session of papers will be presented this afternoon on nuclear materials issues and health effects research featuring new insights on the "Linear Non-threshold Dose Response Hypothesis." There is also a separate session Wednesday afternoon on cable aging which you will find revealing. Of course we have many of the same sessions as before with updated research results.

I encourage your attendance at Tuesday's panel discussion on the Future of Research as I am certain that you will find it stimulating and lastly, on Wednesday at lunch, Dr. Herb Kouts, of the Defense Nuclear Safety Board, and previously NRC's first Research Director will speak on the history of safety research programs and lessons to be drawn from It. I am sure you will agree that the three day program is indeed focused on issues of interest to all of us.

I would now like to introduce the Chairman of the NRC as our keynote speaker. Dr. Shirley Ann Jackson. Dr. Jackson has been Chairman of the NRC since July 1, 1995. Dr. Jackson earned a Bachelor of Science degree in Physics and a Doctorate in Theoretical Elementary Particle Physics both from the Massachusetts Institute of Technology (MIT) in Cambridge, Massachusetts. Among the many firsts which Dr. Jackson has achieved, she is the first African-American woman to receive a Doctorate from MIT in any subject.

Prior to her becoming Chairman of the NRC, she conducted research at Bell Laboratories and was a Professor of Physics at Rutgers University. Since joining the NRC, she has brought energy and a sense of commitment to the agency. Among her achievements here have been her emphasis on the pursuit of risk-informed, performance-based regulation; her initiative of a strategic assessment and rebaselining effort which has indeed given new direction to the agency. Her recent induction into "The National Women's Hall of Fame" in 1998 reflects very well the many achievements of Dr. Jackson. We are privileged to have her with us today to share her thoughts on risk-informed regulation.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
Office of Public Affairs
Washington, D.C. 20555

Transitioning to Risk-Informed Regulation: The Role of Research

by

Dr. Shirley Ann Jackson, Chairman
U.S. Nuclear Regulatory Commission

26th Annual Water Reactor Safety Meeting
October 26, 1998

Introduction

Good morning, ladies and gentlemen. I am pleased to join you at this opening session of the 26th Annual Water Reactor Safety Meeting. Given the agenda discussion topics, I believe this will be a very informative meeting, and I encourage each of you to contribute your insights and ideas to the dialogue. To begin the dialogue this morning, I would like to discuss with you a significant transition taking place at the Nuclear Regulatory Commission (NRC)–the move toward "risk-informed regulation"–and, in particular, the role of research in achieving this goal.

I believe that the acquisition of valid risk information, and the prudent use of that information in decision-making related to nuclear safety matters, are achievements essential to the continued effectiveness of the NRC and the industries it licenses and regulates. For this reason, I have made the theme of risk-informed regulation central to my tenure as the NRC Chairman. In fact, the Commission is committed to the goal of using risk information and risk analysis as part of a policy framework that applies to all phases of our nuclear regulatory oversight, including rulemaking, licensing, inspection, assessment, and enforcement.

Just as a sound policy framework clearly is the key to making prudent decisions, a vigorous, focused safety research program is fundamental to achieving a robust foundation for risk-informed regulation. Therefore, in my remarks today, I want to answer a series of questions that will place into context the role of research in risk-informed regulation: (1) Why is it so important that the NRC make the transition to risk-informed regulation? (2) Why is research key to the transition? (3) What has been accomplished to date–both by the NRC and by the nuclear industry? (4) What are our areas of current focus? and (5) What initiatives are being planned for the future?

I. Why Is It So Important for the NRC To Make the Transition To Risk-Informed Regulation?

Before answering this question directly, let me set the stage with a brief acknowledgment of the more far-reaching and global changes that are facing the nuclear power industry and the NRC today. The deregulation of the electricity generation market to allow and to encourage competition is expected to lead to new ownership arrangements, and to an increased focus on the control and reduction of facility operating costs. Faced with this changing environment, nuclear power licensees must decide whether to complete the existing terms of their licensed nuclear plant operations, to decommission early, or to apply for a 20-year renewal of their operating licenses. Some already have chosen to decommission prior to the end of the license term. Two licensees, Baltimore Gas and Electric Company and Duke Power, have submitted applications for license renewal--for Calvert Cliffs and Oconee, respectively. For those licensees who choose to continue operation under either the current or a renewed license, the reduction of operating costs clearly will be a primary objective. Licensee efforts to eliminate unnecessary burdens or to achieve greater flexibility will, in many cases, involve interactions with and oversight decisions by the NRC. But these decisions will not be easy to make. As you all are aware, nuclear technology is very complex, not only from the standpoint of the complexity and diversity of plant design features, but also in terms of operational factors such as the human-machine interface, aging effects, and potential accident sequences. The challenge of deciding how, and when, it is appropriate to reduce design margins, to enhance flexibility, and to relieve unnecessary regulatory burden without allowing an undue risk to public health and safety is a significant one.

Given this background, the importance of the NRC transition to risk-informed decision-making in regulatory matters quickly becomes evident. Essentially all Commission and NRC staff decisions can be made more effectively, if they can be based on valid information about the risk importance of the decision. For each rulemaking, regulatory guide, or generic letter we issue, the Commission conducts a regulatory analysis to weigh the costs associated with the action against the risk reduction and safety enhancement to be achieved. For nuclear power reactors, the Commission also has adopted the backfit rule which requires that, with the exception of cases involving compliance or adequate protection, the proposed action must provide substantial additional protection before it will be taken. In a number of instances, a significant volume of risk information and risk analysis has been developed to support these decisions. However, in other cases, we have had only <u>qualitative</u> information available about the risk reduction potential of a rule change, whereas quite specific <u>quantitative</u> information has been available concerning the potential costs. Clearly, the quality of our decisions on generic regulatory matters will improve as the breadth, scope and generic applicability of available risk information improves.

In addition to decisions on generic issues, each year we make numerous plant-specific decisions--on the appropriateness of a particular license modification, on what aspects of a given facility or licensed activity should be inspected, or, in a given case, on whether a civil penalty should be issued for a violation of NRC requirements. Once again, it is obvious that these plant-specific judgments will better ensure the protection of public health and safety if we can base them on a solid foundation of plant-specific or site-specific risk information.

The Commission has formally documented its position on the importance of using valid risk information in its deliberations and actions. In our Principles of Good Regulation, we state that "Regulatory activities should be consistent with the degree of risk reduction they achieve." Under the principle of "Reliability," we further state that, "Regulations should be based on the best available knowledge from research and operational experience. Systems interactions, technological uncertainties, and the diversity of licensees and regulatory activities must all be taken into account so that risks are maintained at an acceptably low level." In the 1995 Commission Probabilistic Risk Assessment (PRA) Policy Statement, to which I will refer again later in these remarks, we also note that PRA and associated analyses should be used "to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices." These quotes summarize the general principles of the Commission regarding the use of risk information and risk analysis--that we should use these insights to increase our safety focus, to achieve appropriate risk reduction, and to eliminate unnecessary conservatism and any associated unnecessary burden on our licensees or the NRC staff.

When applied broadly, as a coherent approach to the full spectrum of regulatory activities, risk-informed regulation will allow us to maintain a clear sense of the primary NRC health and safety mission, while also being responsive and flexible in the face of change. When such an approach is not in place--even when a clear sense of mission exists--the organizational response to emerging issues can become a patch-work of quick fixes, knee-jerk reactions, and/or redundant programs that quickly balloon into overall inefficiency, ineffectiveness, and a lack of clear priorities. A risk-informed approach provides a structured, systematic, and defensible method that can be applied not only to rulemaking, but also to licensing, inspection, enforcement, and performance assessment--as well as providing a basis for prioritization in the establishment of programs and the allocation of resources.

2. Why Is Research a Key To This Transition?

Let me next provide my perspective on why research is such an important attribute in the pursuit of more risk-informed regulatory decision-making. Information from research programs can aid such decision-making in several ways: (1) by providing new information, (in the form of test results or detailed analysis), that sheds light on the likelihood, consequences, or mode of progression of a given accident; (2) by relating that risk information and analysis to the specific context of a rule, regulatory guide, or generic letter--even when that information leads to the closure of an issue or concern without new requirements; and (3) by providing a risk-informed context relevant to a plant-specific licensing, inspection, or enforcement action.

These enhancements to decision-making can occur in the NRC oversight of either reactor or materials licensees, and can take place independently of whether a formal PRA or partial PRA has been developed for the facility in question. However, when coupled with PRA information or folded into a PRA, research on system or human performance or on accident phenomena can be even more helpful in providing a directly relevant basis for regulatory decision-making. Additionally, research can advance the state-of-the-art of PRA, by reducing or quantifying uncertainties in risk estimates, allowing new phenomena to be incorporated into risk estimates, addressing previously unmodeled operating modes, or providing greater design detail for inclusion in a given risk model.

These aspects emphasize the role that research has played and will continue to play in this vital NRC transition to risk-informed regulation. Clearly, a great deal of work remains to make this transition complete and successful. However, before discussing the new initiatives we need to undertake to accomplish our goals, we should recognize what already has been accomplished and consider the efforts that currently are underway.

3. What Has Been Accomplished To Date--Both By the NRC and By the Nuclear Industry?

Clearly, one of the earliest milestones in advancing our understanding of nuclear safety risk was the publication of the WASH-1400 study, in 1975. WASH-1400 provided risk estimates for two plants--a Westinghouse-designed pressurized water reactor (PWR), and a General Electric-designed boiling water reactor (BWR). Although there were criticisms of WASH-1400 that limited its application in regulatory decision-making, it did represent a significant advance by demonstrating the potential benefit that a more fully developed PRA could have as a regulatory tool.

As time progressed, the NRC and the nuclear power industry continued to conduct risk studies, and PRA methods and insights gradually began to be seen as having direct application to regulatory activities, as a valuable complement to deterministic engineering approaches. In other words, the application of PRA has represented an extension and enhancement of traditional regulation, rather than a separate, different, stand-alone technology. PRA methods were used effectively during the anticipated transient without scram (ATWS) and station blackout rulemakings, and were incorporated into the generic issue prioritization and resolution process. Probabilistic analyses also were used in developing an approach to estimate the Safe Shutdown Earthquake Ground Motion for a reactor site, as part of the rule change to reactor siting criteria in 10 CFR Part 100.

In 1986, the Commission took a key action toward incorporating risk information and risk analysis into an overall framework for decision-making, by publishing the NRC "Policy Statement on Safety Goals for the Operation of Nuclear Power Plants." These Commission Safety Goals set forth quantitative societal health effects objectives, based on the incremental risk of cancer arising from potential accidents at nuclear power plants. The Commission recognized that such goals could be implemented best through the continued maturation of PRA as the mechanism for performing quantitative safety assessments.

In early 1991, the NRC published NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," after extensive peer review. In NUREG-1150, the NRC used improved PRA techniques to assess the risk associated with five nuclear power plants, including the WASH-1400 plants and adding an additional BWR and two PWRs, to examine a range of containment designs. This study was a significant turning point in the use of risk-informed concepts in the regulatory process, and enabled the Commission to improve greatly its methods for assessing containment performance and accident progression after core damage initiators. The methods developed for these studies, and the results that emerged, provided a valuable foundation in quantitative risk techniques.

With the increasing sophistication of PRA techniques, the NRC began to use PRA information in assessing the safety importance of operating reactor events, as well as making risk analysis an integral part of the design certification review process for advanced reactor designs. Some reactor licensees also began using risk-assessment methods to identify plant vulnerabilities--and with the initiation of the Individual Plant Examination (IPE) program and the Individual Plant Examination External Events (IPEEE) program, all reactor licensees began participating in this effort. I should note that all power reactor licensees have completed the IPE program, and to a large measure already have undertaken the plant changes judged to be appropriate based on a weighing of risk reduction versus cost. I also would point out that we are nearing the completion of IPEEEs by all licensees, which, with the resultant plant design and operational changes, will represent another significant milestone in risk-informed regulation.

Since the publication of NUREG-1150 and the continued work of the nuclear industry to enlarge and improve the PRA database, the Commission has continued to develop policies and guidelines on the use of PRA insights. In 1995 the Commission published a major revision of the NRC Regulatory Analysis Guidelines (NUREG/BR-0058, Revision 2). The Regulatory Analysis Guidelines included a formulation of screening criteria for using PRA information, in conjunction with the subsidiary safety goals, as part of regulatory decision-making on generic matters such as rulemaking and generic letter issuance. The guidelines relate the subsidiary safety goals to the criterion of substantial additional protection contained in the NRC Backfit Rule, 10 CFR 50.109. They also lay out criteria for the quality of the risk information needed for such safety goal evaluations.

Perhaps even more significantly, the Commission also published in 1995 the PRA Policy Statement, from which I quoted earlier, setting out the broad principles and goals that the Commission would pursue in the PRA Implementation Plan. With the publication of the PRA Policy Statement, the Commission consciously began to take a more holistic approach toward risk-informed regulation, with the goal of establishing an overall framework for risk-informed decisions in all regulatory functions, as well as on plant-specific licensing issues. The NRC PRA Implementation Plan was established to describe and monitor progress on Commission initiatives, including (1) the development of additional regulatory guidance on risk-informed plant-specific licensing decisions; (2) the incorporation of risk information and analysis into NRC rulemaking, inspection, licensing, and enforcement programs; and (3) the linkage between NRC and industry activities in this area.

Just this past July, the Commission published Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Bases." Whereas the Regulatory Analysis Guidelines provide a framework for generic decision-making <u>initiated</u> by the NRC, this Regulatory Guide provides a general framework for plant-specific NRC decisions that have been <u>requested</u> and <u>initiated</u> by licensees. It sets forth the Commission-approved principles for NRC staff evaluation of such proposals, including expectations for application of the Commission Safety Goal Policy, reliance on traditional defense-in-depth approaches, and maintenance of sufficient safety margins when initiating changes to the licensing bases. In addition, it provides criteria for the scope, level of detail, and quality of the PRA supporting the licensee submittal.

The publication of Regulatory Guide 1.174, while critical, was only a partial step toward the overall use of PRA information for plant-specific decisions. Regulatory guides and standard review plans also have been published that outline risk-informed approaches for in-service testing programs, technical specifications, and graded quality assurance programs. In addition, the staff has provided, for trial use, a regulatory guide on risk-informed in-service inspection.

The nuclear power industry has initiated pilot projects at several reactors that are expected to provide a basis for refining these risk-informed regulatory guides. These pilot applications have included proposed licensing changes such as (1) changes to allowable equipment outage times; (2) changes to equipment testing intervals; (3) changes to the types, locations, and frequency of piping inspections; and (4) reduced quality assurance measures on specified equipment. I will acknowledge that during the development of the PRA regulatory guides, there was insufficient acceleration of plant-specific pilot requests for changes to plant licensing bases. This has changed, especially since the guides now are complete (although still subject to revisions based on actual use). In addition to the pilots, the Nuclear Energy Institute is sponsoring a whole plant study to support the development of changes to 10 CFR Part 50. A significant effort also is underway to develop an industry consensus standard for conducting a PRA, coordinated by the American Society of Mechanical Engineers (ASME). Groups such as the Electric Power Research Institute have contributed greatly to these and other ongoing risk studies. The NRC staff is committed to working with the nuclear power industry to ensure that these efforts achieve the desired advances in risk-informed regulation.

I would like to point out that not all NRC regulatory activities lend themselves as readily to the use of risk analysis event trees and fault trees, found so useful for commercial power reactors. Although the NRC has developed probabilistic methods for performance assessment of waste disposal facilities and decommissioning, we are still evaluating approaches for applying risk analysis methods to medical and industrial uses of radioactive material. While our focus at this meeting is on water reactor safety, it is important to note that the NRC staff also has developed a plan for moving forward on risk-informed regulatory oversight of our materials licensees, and to the degree that resources permit, we intend to implement that plan.

4. What Are Our Areas of Current Focus, Relevant to Risk-Informed Regulation?

Clearly, many of the specific initiatives I have described so far are areas of ongoing activity, in which we have made significant progress but which will continue to evolve and mature. Most of these initiatives have related directly toward risk-informing NRC requirements–either through a risk-informed approach to rulemaking or through guidance on making risk-informed changes to the plant-specific licensing bases. In recent months, however, the Commission has accelerated the transition toward risk-informing NRC processes–that is, establishing a framework for NRC inspection, performance assessment, and enforcement that will more readily accommodate and incorporate risk information and risk analysis. These changes, for the most part, are still works-in-progress; however, the NRC staff has been working intensively in these areas, actively soliciting and receiving input from our stakeholders, and I would like to share briefly with you the general direction of the progress we have made.

Over the past three years, the Commission has placed increasing emphasis on risk-informing these processes. The 1996 Commission-directed Arthur Andersen study of the Senior Management Meeting process resulted in an increased emphasis on using objective, quantitative information as input to the assessment of reactor licensee performance. In addition to developing more objective performance indicators, the Commission directed a more systematic processing and comparison of regulatory performance data in the areas of human performance, enforcement, allegations, and risk. Inspection procedures and the NRC Enforcement Policy were revised to require the explicit consideration of risk information as part of evaluating the significance of problems identified. Senior reactor analysts, trained as PRA experts, were placed in each of the regions and in NRC Headquarters. Reactor inspectors were provided additional training on PRA and PRA applications. Taken together, these efforts helped to lay the groundwork for the increased incorporation of risk information and risk analysis into our reactor regulatory oversight processes.

Current efforts, however, are re-examining our inspection, assessment, and enforcement processes in much more fundamental, comprehensive terms. For example, the NRC staff has sought to answer the question: What is the "risk-informed baseline" level of inspection for reactor licensees? In other words, what is the baseline amount of inspection that the NRC must conduct–even at the best performing reactor sites, in order to have the requisite degree of confidence that licensee safety performance is being maintained?

The first step in answering this question, in keeping with the overall NRC mission of protecting public health and safety, has been to identify the "Cornerstones of Safety"–those fundamental objectives that characterize safe and appropriate reactor licensee performance and plant material condition. When considering light-water reactor safety, these cornerstones basically reduce to the following: (1) minimizing plant transients; (2) preventing accidents; and (3) being able to mitigate accidents, should they occur. Once these cornerstones have been established, we then can proceed (1) to define the inspectable population of facility equipment and activities; (2) to determine monitoring methods that will provide the desired level of confidence that no undue risk is presented by facility operation; and (3) within such a context, to establish and execute an inspection program that can be adapted as necessary to the characteristics and performance of specific licensees.

The NRC staff has been working with industry representatives and other stakeholders to determine how this sort of risk-informed program could be established most effectively. At a recent public workshop, members of the NRC staff worked with industry representatives to define these cornerstones of reactor safety, as well as to discuss how radiation safety and safeguards objectives could be integrated into such a program. In particular, the workshop focused on how a risk-informed NRC reactor performance assessment process could take its input from a combination of objective performance indicators, NRC inspection results, licensee reports, and other data sources. The NRC staff also has been working with our stakeholders to determine the appropriate role of enforcement as an integrated part of an overall risk-informed regulatory oversight framework.

Based on our current schedule, the NRC staff intends to brief the Commission next week on these proposed improvements–and in particular the proposed changes to the reactor assessment process. While the Commission has not yet determined the appropriateness of these changes or the exact features of the processes that will result, we have made clear our commitment toward achieving an overall framework for reactor oversight that is coherent, scrutable, defensible, and risk-informed. This will allow the NRC to apply <u>necessary</u> burden, but not <u>unnecessary</u> burden. Based on the continuing efforts by the NRC staff and the industry, we believe that we are making rapid progress toward that goal.

5. <u>What Initiatives Are Being Planned for the Future?</u>

So far I have described how WASH-1400, NUREG-1150, and industry efforts have provided a substantial body of risk information, how various Commission policies and guidance documents have established a framework for incorporating this information into regulatory decision-making, how we are seeking to reform our reactor oversight processes to be more risk-informed, and, in fact, how a number of risk-informed regulatory decisions have been made. Given the progress that we have made, should we be content with continuing to use existing tools–that is, continuing to rely on the current state-of-the-art in PRA methods?

In my opinion, the answer to that question is "no." Too many issues and decisions still face the NRC and the nuclear power industry that could benefit from advances in the state-of-the-art of

PRA capability. A vigorous research program must be retained in order to achieve these advances. Consider some examples:

- Our ability to understand the risk effects of component and structural aging at nuclear power plants and other facilities will become increasingly important as facilities age, and as we assess the capability of licensees to manage those aging effects.

- Continued efforts to model the effects of human performance and reliability on risk would help to narrow the uncertainties persisting in this area.

- Licensees are continuing to replace analog circuitry with digital technology, including software, in the safety and control systems of power plants. Although we have in place some deterministic acceptance criteria for such replacements, we would benefit from the ability to quantify more accurately the reliability of these new systems through the use of probabilistic methods.

- Some stakeholders have expressed an interest in reducing the burden or providing greater flexibility in NRC fire protection and quality assurance requirements. The ability of the NRC to determine how to proceed would be facilitated by a better understanding of the risk significance of the various facets of these programs.

The NRC Office of Nuclear Regulatory Research has developed plans to advance our capability to analyze all these issues, using PRA methods. If the Congress provides sufficient funding, we intend to pursue those plans.

In addition to these initiatives, the NRC research program will continue to provide the technical bases, including improved calculational tools and data, to support more realistic analyses of safety margins in thermal hydraulics, fuel behavior, reactor physics, engineering, and materials. These efforts will lead to more accurate risk estimates, elimination of unnecessary conservatism, and improved decision-making.

We also are considering several new initiatives. The Commission is considering the issuance of a white paper on "Risk-Informed, Performance-Based Regulation" to facilitate a common understanding among our stakeholders of such key terms as "risk-informed," "risk-based," and "performance-based" regulation. Given that the NRC has been criticized for not having a real definition of safety, as we move to risk-informed, performance-based regulation, a common understanding of fundamental terms is of paramount importance. The NRC staff also is reviewing the possible benefits of a revision to the existing Safety Goal Policy, to add a quantitative safety goal for core damage frequency. We are considering, both internally and in joint efforts with the nuclear power industry, how we could make our reactor regulations in 10 CFR Part 50 more risk-informed, with particular emphasis on developing a more risk-informed 10 CFR 50.59 process. Throughout all of these actions, we are pursuing active interaction with our stakeholders, to ensure that this transition to risk-informed, and, where appropriate, performance-based regulation is completed in a deliberate, sensible, safety-conscious, and scientifically sound manner.

Conclusion

This completes my overview of the ongoing NRC transition to risk-informed regulation, and of the vital role that research must play in that transition. In conclusion, I would note that some members of industry have expressed concerns that the NRC focus continues to be on developing the risk-informed framework, and that as yet our licensees have not realized any substantial positive impact or relief from NRC efforts. In addition to continued development of foundational elements, we are accelerating our efforts to achieve definite, measurable outcomes in all license-initiated requests. Let me assure you, without ambiguity, that the Commission metric for success in this area is not simply the completion of a framework–but in fact is the implementation of that framework, and the use of associated guidance documents. In short, we will not be content simply by having published "outputs." Rather, we will measure success by our achievement of the desired "outcomes." Your discussions and deliberations here will help both to achieve and to measure those outcomes.

I hope that I have given you a clear sense of the strong Commission commitment to developing a risk-informed regulatory framework that will encompass all NRC regulatory oversight functions, and that will underlie and provide a defensible, coherent basis for future decision-making. I hope that I have given you additional perspective on how far we have come, and on where we are headed. I also hope that you understand more fully the value of previous and future research accomplishments as part of this overall effort. The Commission will continue to make every effort to assure that NRC activities, including our research program, are focused on providing the information needed to support risk-informed decision-making. Thank you.

FUTURE OF NUCLEAR SAFETY RESEARCH

Ashok C. Thadani, Director
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

I am pleased to welcome all of you this morning to this plenary session's panel discussion on the future of nuclear safety research. This is a topic of great importance not only to us at the NRC, but also to you and to our other colleagues from around the world, especially in today's climate of dwindling resources.

We have, today, a very distinguished panel of experts in the field of nuclear safety, they are:

Mr. Ralph Beedle, Senior Vice President and Chief Nuclear Officer, Nuclear Energy Institute, Washington, DC;

Mr. Jim Lang, Director, Delivery Energy Conversion Division, Electric Power Research Institute, Charlotte, NC;

Dr. Michel Livolant, Director, Institute of Nuclear Safety and Protection, Paris, France;

Mr. Dennis Harrison Director, Operating Reactor Research Program, Department of Energy, Washington DC; and

Dr. Dana Powers, Vice Chairman, Advisory committee for Reactor Safety and Department Manager for Energy Nuclear Facilities, Sandia National Laboratory, Albuquerque, NM.

In order to conserve time, I will not describe their many accomplishments except to note that their contributions to nuclear technology have, in fact, been noteworthy. Our plan is for each of us to briefly provide our views on the future of research, and allow sufficient time for questions and answers.

I would like to speak to you on the challenges that we face, here at the NRC, and on the direction and the steps that the NRC's Office of Nuclear Regulatory Research is taking to meet these challenges. At the outset let me state what I believe should be our mantra - "Safety research should be pro-active, outcome oriented, and risk-informed."

The nuclear industry is changing and maturing. New technologies are evolving. New information and operational data are being generated, and our operating plants are aging. These factors pose new issues and challenges for nuclear safety research.

Our research needs to be responsive to this changing environment. Research is needed not only to provide the necessary information in order to support nuclear safety but it is also needed to support removing unnecessary conservatism or requirements and providing additional licensee flexibility (i.e, burden reduction decisions). Resource constraints (declining budgets and dwindling qualified staff) and an increased focus on outcome measures, emphasize the need for efficiency and effectiveness. Economic deregulation and restructuring will continue to increase pressures on the industry to gain efficiency. Efficiency will be sought by the industry by employing new technologies, including risk assessment, and improving operational performance in areas such as power operation and extended operating cycles and higher fuel burn-up levels.

In recognizing these changes and their demand to improve the way we conduct our business, and as Chairman Jackson noted yesterday, the NRC is currently in transition to a more risk-informed regulatory process. This transition also requires that our analytical tools should be modified to better understand margins for safety and to remove any unnecessary conservatism that can be determined by research results.

The following identify what I believe is our mission and future role of research at the Nuclear Regulatory Commission:

- Resolve safety/regulatory issues while providing appropriate flexibility to the licensees;
- Reduce uncertainties in areas of potentially high risk or safety significance;
- Recommend improvements to NRC's regulatory programs/processes to achieve enhanced safety while improving efficiency and effectiveness;
- Develop and maintain regulatory tools and data bases;
- Keep pace with new technology such that the licensee's use of technology is not delayed by the regulatory acceptance process;
- Take specific actions to improve the effectiveness of interactions with internal and external stakeholders;
- Using risk-informed thinking throughout Agency programs;
- Sunset activities when sufficient information is available for regulatory purposes; and
- Maintain a high caliber professional staff.

I believe that research remains a critical and vital component in the infrastructure that ensures the safe and efficient use of nuclear technology. The specific challenges for research include:

- Providing the tools and understanding of emerging issues so that they can be properly dealt with in a timely fashion. Research should be pro-active.

  In Hockey the best players don't go to where the puck is, but rather where the puck is going to be.

- Research should be outcome oriented. Updating codes and standards, criteria, practices, processes and analytical methods to reflect experience that leads to improved safety and/or reduce unnecessary conservatism.
- Maintaining the infrastructure of tools and data so that the best available knowledge is used in safety decisions and that lack of such knowledge does not become a hindrance to safe and efficient nuclear plant operation.

Given these challenges, I would like to focus our panel discussion on the following questions:

- What should be the major role for Research in the future?
  - improve operational efficiency
  - respond to emerging issues
  - support advanced designs
- What technical issues most need research to ensure an effective and efficient solution?
- What role should risk information play in defining future Research programs?
- How best can Research capabilities be assured in light of large budget reductions?
- What deciding factors should be considered in terminating a Research program?

FUTURE OF NUCLEAR SAFETY RESEARCH

Ralph Beedle, Senior Vice President and Chief Nuclear Officer
Nuclear Energy Institute
Washington, DC

Let me start with an expression of thanks for being able to come here and talk with you today. But what I'd really like to do is start with a story. Ashok presents this thing, you know, as very dry and may put half of you to sleep after the first couple of minutes. That's why he only wanted us to speak for ten minutes because he didn't want you to sleep too long.

This guy is driving through the Virginia countryside, and he encounters a huge herd of sheep. So he gets out of his vehicle and stands beside the road. And before too long, the shepherd comes along. So he engages the shepherd in conversation and chit-chat and so forth. Finally, he says, "I'll bet you $100 against one of your sheep that I can tell you exactly how many sheep you have in that flock." The shepherd said, "Well, there's no way this guy can do that. I mean, it just looks like a mass of white, all these wooly things moving across the road." So the shepherd says, "Sure, I'll take you on, on that bet," and so the guy kind of does one of these robotic scans of the flock, and he says, "You've got 973 sheep in that flock". The shepherd is absolutely amazed. He says, "I can't imagine that you can do that. That's fantastic." So the shepherd says, "If that's a bet, go pick up one of the animals," and the fellow goes over and grabs one of the animals and heads back to his vehicle. The shepherd says, "Just a minute. I'd like to double or nothing on that bet that I can tell you exactly what you do for a living," and the guy says, "Well, there's no way that shepherd can tell me what I do for a living." "You're on." So the shepherd says, "You're one of those research guys from Washington, DC," and the guy is absolutely amazed because he's right. He says, "Well, how did you do that?" and the shepherd says, "Well, you put my dog down, and I'll tell you." All right.

Joe Callan accused me of never telling a story without a point. And there is a point to this, that, you know, we gather groups like this. And this is really a very important meeting. It helps us all focus on the subject of research. But we get together, you know, and Ashok is leading the band, trying to get us to think about this.

And when you sit in Washington, whether you're in DOE or NEI or NRC, sometimes we're able to tell you folks that are out there herding the sheep, who are trying to get some results out of your research work -- you're running the labs and the various research facilities and really doing the work -- sometimes we come along and we try and tell you how to do your business, you know, like we've really got a lock on how these programs ought to run. And I would remind you that you're the ones that are out there really doing the work, and you know how to go about it. And we, in the Washington environment, need to focus on how to make it possible for you to do your work, rather than telling you how to do the work. And so, that's really the point in that little story.

Well, let me go back to Ashok's mantra here about safety regulation being an outcome-focused effort, and I think that's really where we need to take a look at some of the research that we do. We are in an era where the dollar is becoming harder and harder to allocate toward things that don't have immediate return. I'm not talking about any return. I'm talking about immediate return.

Ashok mentioned the restructuring and the realignment of the industry. We've got utilities that are faced with the prospects of the sale of their plants within the next year or, even in some cases, months. Do you think that those executives are interested in putting a lot of money into research projects when they're anticipating the sale of their assets within the next year? The answer is no, and you can extend that out to about a five-year period where the uncertainty of where the company is going with their program or the uncertainty associated with the activities of the Public Service Commission call into question the wisdom of putting money into a program that doesn't provide immediate return.

When you take a look at the operational folks in these plants, they're looking for solutions to problems today. They're not looking for the solution to the problem that we forecast will face the industry 10 or 15 years from now. We're looking at how do you solve your maintenance problems? How do you solve your equipment problem? How do you solve some operational problems today? And if you can't see an immediate return on that invested dollar in the research program, then it's really difficult to try and get the operational and management people at these facilities to part with their dollar.

I think probably the greatest example -- and Jim may talk on that when he talks about some of the activities that EPRI is involved with -- but, here is a case where the shift in the focus of research for the industry's research organization, EPRI, is moving more and more toward the immediate return on investment philosophy. And I think we need to give more and more thought to that. So if that's the case, then where do we go with this program of research? Because I think it's really essential, if the industry is to grow, that we continue to see effort to develop the technology to find new and better ways to deal with it. And we're dealing with -- as Chairman Jackson mentioned yesterday -- an extremely complex technology. Nuclear, per se, isn't the complexity that we deal with, but it's the interaction of the mechanical and the electrical and the social as they come together with the nuclear processes in order to generate electricity or to provide medical benefits or deal with things like your radiation of food. All those things come together and create that complex problem that we face almost every day in our business. So conferences like this, I think, are absolutely essential in getting the wide variety of people involved in research to focus on it at one point in time and, perhaps, to improve the level of cooperation between NRC and the DOE, EPRI, owners groups. All of them play a part in these research dollars that flow into the various labs in this country and research facilities. So I think it's important that we try and capitalize on that.

I think that as we are looking at a dwindling set of dollars that are available for research, we have to figure out how to capitalize on those that we do have, and that can best be done through a cooperative and a compatible research program by the industry at large.

Thank you very much.

FUTURE OF REACTOR SAFETY RESEARCH

James F. Lang, Director
Delivery Energy Conversion Division
Electric Power Research Institute
Charlotte, NC

I'm from EPRI and research is our business. My boss, Robin Jones, is the Vice-President of the Energy Conversion Division, and he would have loved to have been here because this is really his need. Unfortunately, he's in Japan working on some other collaborative research programs.

I'd like to take just a minute to review what EPRI is. For some of you, this is something you already know well. But we are a nonprofit collaborative R&D organization, and we're funded by the electric utilities, first in the United States, but more recently, internationally. In the nuclear arena, virtually all of the nuclear operating companies in the United States are members, are nuclear collaborative.

We've got an annual budget of about $100 million. Fifty percent of that is in a very broad collaborative program which all the members fund and in which all the members share. It's in that broad collaborative program that some of the longer range activities are embedded. All the nuclear utilities believe that these are to their benefit and support them. Fifty percent of our $100 million budget, though, is embodied in specific research undertaken by subsets of our members, research into specific issues, smaller collaboratives looking at issues that directly affect them. And that's really where I work most of the time in Charlotte, working on near-term activities of interest to subsets of our members.

Our R&D in EPRI focuses first and foremost on economics and improvement in the economics of operating nuclear power plants, and that includes reliability because that's a very important aspect of economic operation of the plants. Safety is one of the things that sets our nuclear culture apart from most other engineering disciplines. And that, of course, is a very important part of the research that we do. But as Ralph said, it's a dwindling part. It doesn't represent as large a fraction of what we do as it once did.

Collaboration with other industry organizations is something that was something of a tradition in the United States. If we look back to the '70s, EPRI, DOE, NRC research all worked together, not always smoothly. But we did work together. We co-funded activities, et cetera. In the late '80s, I think our relationship with the NRC hit a trough. There was a real concern within the NRC about independence of research and a concern about having their activities tainted by industry involvement. And that concern really ushered in a decade of lack of cooperation, defensive research where, "For every program you've got, I've got one that protects me from what you're likely to find," et cetera. And that has continued until quite recently. I think the economics that both Ashok and Ralph mentioned has sent us a wake-up call. We can't afford to do things over and over again. We can't afford to do everything twice. It's really very wasteful. It's not even a very rewarding thing for researchers to do.

So in late 1997, we signed a memorandum of understanding -- EPRI did -- with the NRC to coordinate our research activities in the safety arena, being very careful to concentrate on data collection. We all ought to be able to agree on the facts. Interpreting the facts is another matter, and we don't collaborate and cooperate. I guess I would say we don't collaborate on that. Working with NEI, we may cooperate on that, but we don't collaborate on that. And that really is the essence of our understanding with the NRC. It's not just EPRI that works in that arena. The NRC, working with EPRI, also has agreed to work with other industry research organizations, work that may be done by owners groups, et cetera. So we have in place now a broad framework to regain that cooperation with the NRC. Our cooperation with DOE has never waned, although DOE's funding certainly has, and we've been working hard with Dennis and the rest of the DOE crew to support regaining DOE funding for nuclear research, not for the university areas and longer range activities, but research to try to make license renewal a more economic and practical activity, research to help make operations more economical and practical and so on.

Now, I'd like to spend just a moment surprising Ashok in addressing his questions. What is the role of research? I'll do it in my own way because I can't remember what they were. I'll go back to something I said earlier and something that Ralph referred to.

I've been at EPRI now 19 years. When I first went to EPRI, virtually every research program was justified on the basis of reducing the margins in Appendix K. About ten years ago, that changed. Today, virtually every research program is justified on the basis of reducing O&M costs. Now, I'm the first to admit there's a certain amount of salesmanship in that. Some of the relationships are tenuous, at best. But in addition to reflecting what our members want, I think it reflects also that change reflects what our members need. I'm not naive enough to think that we understand the answer to all safety questions, that there won't be issues that arise and so on, but the balance has shifted, and the number of open safety issues has shrunk while the potential for more economic operation through research has increased.

Risk informed approaches can help us decide what to do, not only in the safety area, where to spend our limited budget, but in the O&M area, as well. Risk informed approaches are really some of the basis of some of the O&M work, and that helps guide some of our very practical near-term research into cutting O&M costs.

The biggest problem we may face, I think, is one of the last ones that Ashok has on there, is how do we decide when to stop research? You know, some of our research into safety areas and the O&M areas may be like peeling back an onion. You know, if the question is do we stop phase I and start phase II, that gives us a break in which we can address that question.

The real problem ends up when the research is a continuum and we reach the point of diminishing returns, and I wish there were an easy answer, but I think it falls into the category of almost every other activity in life -- there isn't an easy answer. It's hard. It's a management issue. It takes management attention. There isn't a formula. And those of us who are in those management positions just have to step up and accept that, and those who oversee what we do have to hold us accountable for making those decisions. And I can readily admit that I've done some right, but I've done an awful lot wrong. It just takes a lot of attention.

That's all I have. Thank you.

FUTURE OF REACTOR SAFETY RESEARCH

Michel Livolant, Director
Institute of Nuclear Safety and Protection
Paris, France

I am Director of IPSN, which is an institute for safety analysis. We do the safety analysis of the French reactors and nuclear installations, and at the same time, we are an institute for research and we receive money from the French government to do research, which puts us in a situation which is not so much different from what I hear from my American colleagues and maybe allows us to be stealing a position, which is what I understand here and what we see in Europe and other countries. We are in a relatively good position to continue and to push some research, and also in a position to make it effective because we have the other way of work, which is safety analysis of the real cases.

So my presentation is, in fact, probably more technical because we have still time and possibility and money to do some research and to make some future programs. Nevertheless, to predict the future is never easy. I tried the usual way to see by what way we arrived at the research which is done now, and I discovered, which is not very original, that the research at the present time is not a pure, logical instruction, but the result of various events. Some of them were unexpected.

It's clear that severe accident research and the use of probabilistic risk assessment were really stimulated by TMI, and on the other hand, protection studies were stimulated by the Chernobyl accident. Some other research topics were pushed by the technical evolution, like human factors studies, software safety, and hydrogen control in containment, and from time to time, experience brought to light some specific questions which generated the corresponding research.

This actual research was the best starting point to try to predict the future of research, and I will do that by distinguishing three categories. The first category is the continuation of the reactor research in progress up to a correct answer of the problem initially raised. Typically, the severe accidents fall in that category. It is especially complex research which needs expensive, very expensive sometimes, experimental programs and high-level modeling capabilities. There is sometimes a tendency, and what we hear here is going more or less in that direction, to stop in front of the difficulty of the task. But my conviction is that the aptitude of the safety nuclear community to solve those problems is a measurable component of its credibility. As today we hear about credibility, I think this is a part of it. In that domain, I estimate that the research should continue in various plants on severe accidents like, for example, core degradation. If the first part of the degradation is relatively well known, the late phase of the core melting is not well understood. There are no large experiments in progress in that field, but we need to make progress in modeling for that, and some analytical experiments on physical and chemical behavior of core materials are really needed.

Fuel interactions -- there is also a tendency to say that everything is said on that point. We had some difficultly with our reactor. We have to demonstrate to the safety authority that there is no risk of fuel interaction, and we find very difficult to do that without very large conservatism. So we still consider that there is some work to do in that field.

I will go quickly through the other topics in this domain. Efficient product transport in aerosols. We have our famous efficient product program, which is largely open internationally with the participation of NRC. EPRI, for the first time, but they stopped quickly. I don't know why. Hydrogen and risk, the kinetics of hydrogen distribution in the containment has to be improved, and for that, there is some research, like NRC and in Russia, we do experiments with some colleagues and with the German colleagues. And this is important for us. It's research which has direct impact because we have taken a decision in France to put recombiners in the large part of our plant, and so we need to do that the best way possible.

There are other topics in this domain of severe accident research, but I won't go further. I want to speak of the second category of research. The first is the research -- continuation of the actual research -- in topics where we have not obtained sufficient results in our position, and the second category is the basic research. In that category, generally good answers, good answers on the ongoing problems have been obtained, but the capabilities have to be kept in order to react to new unexpected events. Particularly, we found in this category the thermal-hydraulics LOCA conditions, structural integrity, human factors, software safety, and PRA. Thermal-hydraulics in LOCA conditions is one of the success stories of the reactor safety research. It was one of the main topics of this reactor safety meeting in the past. And the computer codes which are now available, RELAP and CATA, are validated with analytical and global experiments and are in use for treating safety cases, especially if they are associated with uncertainties and estimations.

But such success can be a danger if it draws the conclusion that no more research is needed. This could have the result of killing the capabilities, with a risk of becoming unable to reach into unexpected situations. It is essential to maintain the research activity in this field, where new improvements are expected like neutron calculations. I understand that NRC has some thinking about this topic to maintain thermal hydraulic capabilities, and we are thinking the same in Europe. There will be a presentation. I don't know if it is today or tomorrow on that. The same for research on structural integrity. They have a long history with many results, and in the software for mechanical calculations. The development of techniques allows a more precise evaluation of the life of structures and components. So this is one of the disciplines which is important concerning the aging problem, and I suppose it's a part of the research which is still financed because it can be a short-time result system. But it is typically a domain where the capabilities have to be maintained on a national basis very clearly. Recently, in France, for example, we had the pipe rupture on the French reactor Sevu, which is the latest design of French reactors. We are not very proud of that, but we have to do with it and to make changes and to be sure we have really understood what happened.

There is a third part in the future of research, and this is limited in my presentation because it is the part located and connected to new design. And that's one of the problems we have in the reactor domain -- there is not so much new design. Nevertheless, we can see that AP600, or EPR for Europe have stimulated some research in new fields like specific thermal-hydraulic conditions or corium cooling techniques or containment resistance to accidents. If the reactors don't change -- there is some evolution in the fuel nevertheless. The fuel is improving, and the concern is more present now. And this needs to be defined in normal and in accident conditions, and corresponding criteria need to be established. And this is the aim of our project that we propose for international collaboration at the present time.

Just to finish, a word about international collaboration. All that I've said is possible only if you have sufficient reserves available and a condition such that you can be sure to maintain your effort for some years. It's not possible to go into such efforts like big experimental research on the reactor if you have not some assurance of money, of the resource continuation during the same time. But as you know, there is a general tendency to decrease the research budgets, especially the governmental budgets in many countries. And this makes for very difficult choices which are in favor of short-term efficiency. And what I have said about tomorrow is not contradictory with that. And this is to the detriment of more basic or long-term research. And in such a situation, it's clear that international collaboration can be helpful by sharing the cost of some expensive projects. But there is a limit for that. And in many countries, many countries are now in a survival situation where the first priority is to save some national capabilities, and it's quite impossible to participate in an international research program. So international collaboration is a necessity, but it can be successful for a country, only if some national research is maintained at a sufficient level. That was one of my main messages I wanted to give to you.

Thank you very much.

FUTURE OF NUCLEAR SAFETY RESEARCH

Dana Powers
Vice Chairman, Advisory Committee for Reactor Safety and Safeguards
and Department Manager for Energy Facilities, Sandia National Laboratory
Albuquerque, NM

I'm an engineer. I can't talk without viewgraphs. You should see what my wife says when I try to bring a view graph projector in. I'm here representing the ACRS. The ACRS has an obligation to annually review for Congress the needs for research within the nuclear safety program. We've also been asked to do that for the Commission.

We've just produced a report on our review for last year for the research program. This report does delve into both the definition of research programs and the individual research programs themselves. I wouldn't presume to try to summarize all the information in that report because I couldn't and because I think things are changing. So today, what you're going to get is my personal views as we embark on our next review of the program. As Margaret Federline often points out, we're the ACRS and we're definitely here to help you.

Let me make some observations. First, let us not lose sight that research has been absolutely essential to the NRC in the past. And let me also offer my personal observation that I have an opportunity to work with a lot of government agencies in my non-ACRS work. And I have to say, NRC is blessed with some of the most technically capable individuals in its research programs that I've encountered in all of government.

They have carried out a program that's been absolutely essential to establish the credibility of the NRC's regulations. They have defined the framework on what we mean by safety. The research program that they pursued in the past has been exceptionally broad in scope. They seem to have somebody delving into every single topic. At the same time, the research program in the past has been characterized by addressing some very big topics. And I've just listed a few of them here, the successes of the past.

Quantitative risk assessment has been developed by the NRC, revised short-term safety goals – all of these have been major programs, they have produced major changes. The line organization at the NRC has had to respond to these findings from research. More importantly, the industry has had to respond, sometimes a little bit reluctantly. I think this is a mode of operation that has worked well in the past, but I think its time has come to an end. The times are changing. Could we ever not say that? But I think that's a real feature of the times. The society, itself, is changing. And there is a societal imperative that its public institutions begin doing things better, faster, cheaper and safer.

The previous speakers have mentioned numerous changes that are occurring within industry which are very real changes. We also have to recognize that technology is changing. When people speak of technological change within the nuclear industry, they often cite digital electronics. And that, for sure, is going to be a big change. But there are other changes. Fuels are changing. It is these changes that I think refute the notion that the nuclear industry is static, where the plants are fixed, that we don't need to do any more research. In fact, I think there is more need for research today than there was ten years ago.

Well, now we come to where I think the research ought to go. How should it change? I think we hear over and over again that Research is going to have to go from being an independent part of NRC that is prompting change to one that becomes a support organization for the line organizations. It has to be in a position to provide the line organizations with tools to do risk-informed regulation better, faster and cheaper. I think this means that there is going to be an end to the entitlement process of funding. That is, funding in the past for NRC Research has been done by saying, "Okay, we have this block of money. We'll give it to Research." And they go do what they think they have to do. They have to prioritize their activities. I think the problem you run into when you are in that kind of mode, when you have a body of money and you set out to prioritize some activities in there, automatically telegraphs to the world that your activities are a luxury and that if they want to cut back your funding, it's okay. It's just a luxury. You can do without it. We can do without it because we can always impose a little more conservatism.

I think Research is going to have to evolve so it's not the constant whipping boy for any reductions in funding but, rather, portray its activities as essential for the NRC to carry out its mission. In other words, Research is going to have to carry out and organize its programs so that they are justified on the basis of the return on investment to the agency. In order to do that, they're going to have to argue strenuously that imposing conservatism to compensate for a lack of knowledge is an unacceptable thing to do and an unnecessary thing to do. In fact, in contrast to most of the speakers, I don't think there's unnecessary conservatism embodied in the regulations today. I think there is, is a need for Research to say, "We can reduce the necessity of conservatism in these regulations by improving our knowledge." The consequence of this is, I think, that many of the historical research activities within NRC will have to be closed, and responsibilities for them will have to be assumed by the industry. And I think this is taking place -- I cite, as an example, reactor fuels again. The NRC has certainly established the limits under which it will allow fuels to operate now and said that anything that goes beyond that will have to be done by the industry. Still, I think there are a few technical capabilities that NRC will have to maintain -- Research will have to maintain for the NRC, itself, a few of these where they need expertise that they cannot acquire from the outside. What should be the research topics? I think the Commission has done an outstanding job of setting a new direction for the NRC and defining risk-informed regulation.

I am quick to point out that it is Research that developed the technology that has allowed the Commission to turn to risk-informed regulation. Implementing this represents a challenge of enormous magnitude. I, in fact, share Commissioner McGaffigan's suspicion that it may not be possible to do risk-informed regulation.

First, I note that right now we are unable to calculate risk in a defensible fashion. That is, if we calculate risk at NRC and the industry calculates risk using its methods, we will get unresolvable differences because we have a lack of knowledge. I think in all likelihood, we're not going to have this conformed to regulation. I think we're going to have core damage frequency informed regulation. We may, in fact, only have core damage frequency under operating conditions informed regulation because there are substantial shortfalls in our ability to calculate core damage frequency. And I've listed down a few of them. I think people are aware of these. Even when we try to do core damage frequency under operating conditions informed regulation, the processes are extraordinarily slow. I've recently been reviewing a particular case in which three separate determinations of risk had to be done on looking at a particular issue for prioritization. It took years to prioritize, just to prioritize an issue, because we did not have in hand processes for calculating core damage frequency under operating conditions to make an informed decision. In other cases, we run into problems imposing core damage frequency informed regulation because there's an inherent conservatism in the system that says, "Ah, don't relieve this regulation. It's there for defense in-depth." We have not yet defined a good relationship between core damage frequency and the traditional thinking of defense in-depth. My personal view is that we tend to do defense in-depth at too microscopic a level. But without a routine capability and a reliable capability to calculate core damage frequency, this conflict perceived conflict between defense in-depth and risk-informed regulation, is going to slow the progress. And it is going to be the responsibility of Research to change that situation.

What do I recommend? Well, my first recommendation is to establish a division of the tools and technology that should be available to the NRC line organizations in the next 20 years to implement core damage frequency. I hesitated to put the word "vision" up there because if you haven't been comatose over the last 15 years, you have frequently heard people imploring you to develop vision statements. And, God, they're awful. They're saccharine and pointless. Those are not the kind of visions I am talking about here. I am talking about visions that say, "Do we want inspectors to have the ability to routinely calculate the core damage frequency of a particular configuration under shutdown to make an assessment of its safety? Do we want to have line organizations in headquarters to routinely have the capability to do computational fluid dynamics of flows through particular valves or to assess the vulnerability of secondary piping systems to flow-induced corrosion as a matter of routine in their jobs? Do we want them to have those tools?"

If the answer is yes, that's a vision of what you want, and the research that you have to pursue to do it can be readily defined. You have to sell that vision to the line, develop the research, and then demand you get the funding to do it rather than using it as, "Here's one component to draw upon a set or a block of money that we've gotten by entitlement." It will take some courage to pursue this point because I think you have to curtail or limit research that does not contribute to the individual visions that are established in the various areas of NRC Research. I want to also point out that NRC has offices that do assessment of regulatory effectiveness. And I think that is research and that it is essential to have that research to support any vision of the future.

I think it's also going to be necessary for the NRC to define those few competencies that it has to maintain in order to serve the agency's needs as the industry changes and brings new technologies to apply to the nuclear power. I think those are few in number and that we have to guard against defining everything as an essential competency. And that concludes what I have to say.

Thank you.

FUTURE OF NUCLEAR SAFETY RESEARCH

Dennis Harrison, Director
Operating Reactor Research Program
Department of Energy
Washington, DC

Good morning. I welcome this opportunity to present the Department of Energy's perspective on this important panel session. First, I'd like to say that Bill Magwood asked me to express his apology for not being able to provide DOE's perspective on the future of reactor safety research. Bill was looking forward to this meeting. However, a last minute change that he couldn't correct occurred.

Before discussing DOE views, I would like to discuss some changes in DOE, and it's primarily so that you can understand our recommendations. Speaking of change, just last night I looked at the first view graph. We had Bill as acting -- just late yesterday, they made the announcement Bill is now the permanent director.

When we look at DOE, our mission really hadn't changed in terms of supporting the key elements in DOE. But I think the things to recognize are, one, the management. We are under new leadership. There is new policy and guidance. And by that, I mean -- I think you heard Jim mention earlier about sometimes we're marching to the same tune, DOE and the industry, DOE and NRC. The new policy and approach to this is that we propose, and we have been, instituting more of an outreach program to work with industry, to work with government agencies prior to developing our strategies, prior to implementing them. That's a key change for us.

The second thing is in terms of our focus. You can think of it in terms of three key elements there. One, the transfer of international nuclear safety programs to the Office of Non-Proliferation and National Security. The second, which is very important, the shift in our focus away from the national security programs back to what we've done traditionally in the past, and that is nuclear technology research and development. And then finally to establish that policy and a new Office of Technology to manage our initiatives in terms of NEPO, the university programs and international collaboration.

I have on there, if you look at the energy -- the NEPO -- there was some mention earlier of that, that's the Nuclear Energy Plan Optimization program. It was included in our 1999 request, but it was not funded. It was technically an effort that involved, as I say, EPRI, NEI, the national labs, the utilities, in essence, the nuclear industry, whether it's public or private. That activity is included -- should be included in our request for 2000.

The other activity under the energy program which I think directly relates to this panel is the nuclear energy research initiative. And I'll just take a few minutes to tell you what that is. Prior to that slide, though, one other ongoing program is our university program, which is basically our program to provide support for university research reactors and facilities, grants, fellowships and scholarships.

Next slide, please. I think the key here is referred to as NERI, Nuclear Energy Research Initiative, an independent peer review to help select the best research. The key here is to break away from the old ways of thinking. That is, instead of DOE directed research, we're looking towards that openness to work with the labs and universities to try and select the best appropriate research.

The other item I think is important is that we talk about the focus caused by changing environments and the challenges as far as the utilities concentrating on shortterm research for survival. Well, we think the role of DOE is in the area of long-term research, especially advanced technologies which limit near-term commercial application.

International collaboration, we heard about that. That was a cornerstone to help leverage Federal dollars to not only support the research, but also the nuclear infrastructure in this country.

And finally, the NERI program serves not only in terms of looking over the horizon at the nuclear technology that's necessary to continue operation of existing plants -- but also future plants primarily -- it's also the birthing place. And that's where that transition occurs in terms of the crossover between the long-term research and potential short-term research.

The next slide. And here I'll try to answer those questions that were raised in terms of what should be the direction of future research? What should be the scope of research? And I kind of think there's a third one there, too. How should we conduct that needed research? The other panel members have stressed, I think, the first two. And in our recommendations, I think we're trying to address the third one, as well as the other two.

As was mentioned, there are limitations on resources. We have competing national policies on energy, on security costs and environment. And all of these are challenging how we approach what resources we have to resolve the key issues. We must take into consideration -- I think this is very important -- each stakeholder's limitations and institutional roles, as I mentioned, DOE for traditionally long-term R&D, EPRI for short-term, and so on. We must look at those and try to combine those and try to come up with a consensus in terms of a national strategy where we can all maximize the limited resources.

Finally, we must also recognize that the nuclear industry is a world industry now. I think as we look at our vendors as they try to compete in the world, as we look at our research and the closing of research facilities, it also follows that international collaboration is a must. I'm not going to read each of those items up there. But I think, in essence, the three key things, as I say, is that limitations on resources and competing policies say we have to work together. At the same time, we must recognize the unique positions of each of the organizations involved in this. And then finally, international, because of the change in nuclear energy becoming a world industry.

Thank you.

THE CONVERGENCE PROCESS

Nils J. Diaz, Commissioner
U.S. Nuclear Regulatory Commission

In 1964, the nuclear engineering community thought reactor safety had more to do with the reactor core than with the heat transfer problems. Nevertheless, we were working on the latter.

Then, that year, the AEC issued a white paper with the conclusion that the knowledge of light water reactors was solid and there was reason to believe in the technology of breeders. Therefore, the AEC would focus on developments in those areas. Work on heavy water reactors ceased. We started to concentrate on light water reactors. The AEC's decision was not a fully-informed decision.

In 1967, at the University of Florida, I became a full-time AEC contractor on light water reactor safety. We worked on the Large Core Dynamics Program. The National Reactor Testing Site was designing the PBF and the LOFT.

However, early in the seventies, the AEC's Milton Shaw, wrote a memorandum concluding in essence that there was enough knowledge about reactor safety. Work ceased on a number of projects, especially at the National Reactor Testing Site. The decision was not risk-informed. Two years later, the ECCS controversy developed, and four years later we had the ECCS criteria. Then, in 1975, Rasmussen was examining risk probability. In 1979, we had TMI.

TMI shocked the reactor safety community. We were trained to think that adequate care of the design basis accident would take care of everything else.

Since then, we have learned that reactivity accidents are infrequent and, if they happen, are not very significant. The early emphasis on power excursions and reactivity control had paid off.

The fundamental reactor safety issue is that you have to have heat removal approximately equal to heat generation. Nothing else comes close in importance. We should be challenging the every-day use of the design basis accident as the foundation of reactor safety and the regulatory framework, and place it in the context of today's science, engineering, and operational know-how.

It has taken thirty years to get good technology to handle the phenomena associated with the light water reactor. In 1964, we did not know the interaction of light water with everything that was around. We still have problems every time this coolant is in two phase -- when it is no longer liquid or vapor, but becomes a mixture.

Today, we have a base of knowledge that allows us to state much better what we know and what we don't know. We have to put this ability in the right regulatory context and in the right operational context.

The title of this talk,"The Convergence Process," stems from the need to develop a safety envelope, based on current knowledge, that can guide plant operation, research, and regulation. Convergence should minimize the deviations from acceptable safety standards and also minimize unnecessary burden.

This concept of a safety envelope is more useful than a definition of safety per se because there is no zero factor in risk, zero deviation or zero defect. Zero is not a functional quantity. It does establish positive or negative deviations for control. Plants are not perfect. A mandate of perfection is an assurance of failure.

Light water reactor safety research can become a controller because it looks at deviations between know-how and the need to know and minimizes the difference between reality and expectation.

The Commission has begun a series of initiatives that will change profoundly the way we regulate. The initiatives are not only risk-informed, but they are also directed toward assurance of consistency, transparency, accountability and due process.

Change in the world and in the nuclear industry is profound and accelerating. It cannot be avoided, but it has to be orchestrated. We have risk technologies and information technology to do this. The only thing we have to fear is lack of change itself.

Unfortunately, we have hardened the requirements for nuclear power plants beyond adequate protection. They have become hardened by crises; inconsistent perspectives; insufficient resources; and lack of growth.

We have to become more relevant and more functional by becoming risk-informed. We have to move from a patchwork regulatory fabric, created in an ad hoc fashion, to a multidimensional matrix, a woven fabric. We have to move to make the entire Part 50 of the Code of Federal Regulations risk-informed so that we can accommodate change and make improvements, like graded quality assurance, that work within the framework of our regulations. We can no longer base our regulatory structure and plant operations on the design-basis accident.

Connectivity has to exist between light water reactor safety research, regulations and operations. They are not independent parameters. It is indispensable that research be connected to operation and regulation. For instance, people in operations have to realize that research is vital in creating better operations, and clear definition is actually more than half of the game to achieve good operation -- to know what you have to do and what you have to avoid.

Research needs connections to the past, the present and the future. We have accumulated enormous conservatism that started from initial conditions, boundary conditions, the design basis and other things. We want to be realistically conservative now.

Research needs to be a driver toward convergence. It must help us avoid the gaps and the differences that have plagued the industry and the regulator. At the NRC, research needs to be directed by a systematic planning process that is tied to assuring safety and maintaining only necessary burden.

Research can help us at all levels, from top to bottom: systematically determine how we see our role in protecting public health and safety; determine the performance areas that we must address; set the cornerstones for our inspection and enforcement; and check these activities through confirmatory research.

Timely and independent technical bases for all our regulatory process is also critical to the credibility of safety standards and safe plant operation.

For these reasons, it is indispensable that research be aligned with the outcomes of risk-significant regulated activities.

Overview of the OECD Halden Reactor Project

Carlo Vitanza, Director
OECD Halden Project
Halden, Norway

It's a great pleasure for me to be here today and try to explain how we are doing things in our small place in Halden. I always call it a small place, as we are sort of unpretentious in many respects. Halden is a small place, but it is also a window on the nuclear world, which has been there for a considerable number of years. However, rather than reflect on the past, it's better to look at the challenges that lie ahead and focus on the best way to tackle them.

Later this afternoon, there will be a session devoted to the technical achievements of the Halden Project, so I will not spend time on that now. I will, however, try to present what the Project is today, how it works, its challenges and its perspectives.

Yes, it's true that many people see the Halden Project as a success story, that is, an organization that has been able to maintain momentum throughout many years of activity. And I would also like to make the point clear that we want, and our firm intention is, to maintain this position of building on the things that we have been able to achieve so far.

In this endeavour, we realize that we need the help of people around us, primarily the licensing organizations and the industry. We need their input, their suggestions, and their needs spelled clearly. We also need their forgiveness at times because no one is perfect. We make our mistakes and try to learn from them. Now, some of you may be here trying to get inspiration on how an organization should be successful internationally and hear what the key to success is. I have a figure that tries to explain these conditions for success, and I want to show it to you. The reason for having this blank page is that I don't think that direct recommendations can be transmitted from one organization to another. Said in other words, an organization will have to find in itself, in the human infrastructure it has, in its technical capabilities and infrastructure, in its ability to renew itself and the relations with its surrounding partners, the key for its own successful operation. I don't want to give a lesson to anyone on success. First of all, assuming that we have been successful, then again let's not indulge on that. Let's look ahead because there are plenty of hurdles and problems to solve in order to maintain and build on what has already been achieved.

The USNRC has always been an important supporter to the Halden Project. We have worked together for a number of years and have a very good understanding of each other, on what work needs to be done and on how that work needs to be carried out.

The NRC involvement in the Halden Project started three decades ago and, at that time, focussed on the fuel activities, that is, on tests devised to understand nuclear fuel behaviour. These tests, conducted in the Halden reactor, made use of the extensive instrumentation that was already available at Halden. This has always been one of our specialities, to be able to measure things inside a reactor directly during operation, while things are happening. This is why the Halden data are so important, because they disclose the full story of fuel behaviour.

When you do post-irradiation examination, you are getting valuable data, but only at one point- in-time, that is, at the end of irradiation. With the instrumentation that we have, you can build an entire history and a good understanding of what is going on in the fuel during service. That is why the USNRC used our facility when it needed data to build up the first generation of fuel performance codes.

But then, at one point, the USNRC decided that they were not interested in fuel anymore. It was just after the TMI accident, when their interest switched from one day to another, to other scenarios, to other items. After TMI, the USNRC priorities became human factors, instrumentation and control, operator support systems, alarm filtering, alarm handling. That came, of course, for a reason.

Our Institute had, at that time, started working in the area on instrumentation and control for the petroleum offshore industry, an industry that was then emerging in the Norwegian North Sea. We put that expertise at service for these new needs in the nuclear industry. Today, human factors, instrumentation and control, and control room support systems represent as much as 40 percent of our activities.

Meanwhile, we didn't forget the fuel at all, as other members of the Halden Project kept that as a priority item. This turned to NRC advantage when the NRC returned on fuel issues. We had not seen Ralph Meyer for years. We didn't know where he was and all of a sudden, he was back in business, with new spirit and a fresh re-start. I believe it was good to have data available to Ralph at the right time when he needed to review his codes for the high burn-up applications.

I should also say that in this country, we have had excellent collaboration with the industry, that is, with EPRI, General Electric, and Combustion Engineering. We tried to be in touch with the other side of the coin, that is, the industry interests also.

In this country, there has always been a separation, a more clear separation than there is in other countries between utilities and licensing. We try to be a link between these two realities. I don't know how things will change in the future or if time will bring these two realities closer together. We feel we have done our best to serve interests from both sides in a balanced way and will continue to do it in the future.

The Halden Project is placed in a small country up north in Europe, a country called Norway. The Norwegians like to be on their own. There is a feeling for independence in that country that is somehow unique. I don't want to elaborate too much on this, but Norway didn't join the European community, for example. So I guess the next step would be a referendum on whether the European Union wants to join Norway in the future.

I said Norway is a small country. It only has 4 million people, scattered around a very interesting country. Being an Italian, I've been able to enjoy a lot of things there, especially the Norwegian nature. Norwegian people are also special in that they are sort of pioneers in many different ways. For example, they claim that a guy called--I think it was-- Leif Eriksen travelled over the Ocean and discovered America before Columbus, or before Amerigo Vespucci. That might be true, but he didn't report his findings. And this is what I often tell people at Halden. You have to report things. It does not help with a good discovery if you keep it to yourself; it's just as if it didn't exist. In summary, Norway is also a nice country to do work in. The society is structured in such a way that things work. The Norwegian government supports the Halden Project. The Government changed one year ago. We had two ministers visiting the Halden Project two or three weeks ago and the outcome was, we believe, very positive. The interest from the Norwegian government is on safety, and we have to keep this priority in mind all the time.

Every three years the Project is close to bankruptcy. This is a threat and challenge but, at the same time, it is what has kept the Project going. This constant threat makes everyone at Halden more alert. We don't have marketing people as such at Halden, but we tried to build the perception we have throughout the organization that everyone is working for a common objective.

I said before that we have been around for some time. We believe at the same time, that we are a young organization, young in that we have always been able to create the conditions for a new start. There were ten countries participating in the Halden Project only eight years ago. There are twenty now. You can take this as a sign of dynamics in the organization, and it doesn't come only because one or two guys are very good and clever. It comes because there is a sound technical basis in executing the program. You cannot sell, so-to-say, if you don't have that technical basis behind you. We are not selling paper. We are executing experiments which very often require complex and advanced technologies for their execution. Experimental data of unique quality are our products and, if they are not good, people will not come back to us the next time. This is why it's important that everybody works with dedication and good spirit.

We have a simple and straightforward organizational chart. We have a Board of Management which is constituted by representatives from all the countries participating in the Project. The Board determines the direction of the Project and the overall priorities of the program, and approves the technical program together with the administrative part of it. We also have the so-called Halden Program Group, which monitors the technical execution of the program. This is a large group with typically 50 representatives.

Both the Halden Program Group and the Board meet twice a year, separately from each other. Regarding the Halden Board of Management, I am very pleased that we will have Margaret Federline as the representative of the United States and of the USNRC, starting with the next meeting in December. We look forward to that.

In addition to fuel safety, we are doing work on aging issues, in particular, stress corrosion cracking especially when on-line measurements are needed. This is, again, our specialty. We also do work on man-machine system research which includes human factors, instrumentation and control systems, and software safety.

I was mentioning before the international character of the Project. There are 16 countries in Europe, and then the U.S., Japan, the Republic of Korea and Brazil. We are also talking to Canadian organizations. With the U.S. and Japan, we had a long-term 30-year association, whereas Brazil joined only one year ago. Please note that there are also some central European countries and Russia in the Halden Project.

We believe that in addition to the technical part, there is something that is more difficult to explain, something non-technical here. It is–how you want to call it–a sort of safety culture, that is, working and talking together, exchanging experience, and finding solutions to problems that concern everyone.

You might say that the Halden Project is also a network at service for all the organizations that are in the Project. I don't know how much importance you attribute to these more abstract things, but one doesn't live with only bread. There are also other aspects that are important in everyday life as well as in our work. We don't want to be the only organization doing R&D, just to put it bluntly. There would be no confidence in a single organization managing everything. We would like, instead, to maintain and also transmit things whenever possible to the different environments.

We are now in the process of establishing a new three-year program, which will start in the year 2000. At the moment, we are addressing the technical content, that is, the issues that need to be covered in the program. To this end, we are travelling from country to country aiming at defining a common basis for the Joint Program. This will be more or less completed by the end of the year. Thereafter, the financial aspects of the program will be addressed, aiming at concluding the entire process in the summer of 1999.

We have reasons to believe that the international support will continue undeminished and the support from Norway will continue. Please note that Norway, as host country, covers as much as 33% of the total budget which is not bad.

I don't know what more I have to say on this except, perhaps, emphasising the principle of cost-sharing which enables us to keep the Halden Center available to the international community at a reasonable cost. The total budget of the Joint Program is approximately $15 million per year.

Note, however, that in addition to the Joint Program, we conduct contract work which also amounts to about $15 million. Also, in this case, the areas of work are man-machine/control room engineering, fuel safety, and in-core materials aging. A significant part of the contract work on man-machine/control room work derives from non-nuclear applications, for instance in Norway.

There was some discussion this morning about the long-term and short-term R&D and that in today's world, you need to come up with quick results. Yes, it's true. You have to always come out with quick results. But, as a manager of a facility, you have a bigger responsibility than that. You have to maintain the facility and plan ahead.

Ashok mentioned the hockey game at this morning's plenary. I am not familiar with hockey; I like football, European football. It is not very smart that everyone runs after the ball. If everyone does that, there is no fun in football games. There must be some guy who goes where no one else has gone but where the ball is coming later. So, you have to prepare this thing ahead of time. Regardless of challenges, we feel we have the duty to plan for some time ahead if we want to have the expertise available at the right time as well as the confidence of our participants.

And the Board encourages us in this effort. The Board has put out a document called "The Views on the Long-Term Direction of the Halden Project," up to the year 2010. That isn't a program; it's a long-term view. It's up to us, in consultation with participants, to transform their views into a meaningful program.

So the outcome—the message of our Board was very straightforward and if I can simplify it, it says, "Build on your point of strength and capabilities, keep the facilities in good order and listen to what the nuclear community has to say on priorities. Further, focus on and use the qualities you have, and they are primarily flexibility and being able to understand the need of your users, taking care of the qualified staff, and transferring the know-how to the younger generation of scientists and technicians." In essence, that is the message.

The Board's remark on staff is important. When running an R&D Center, one must have good people, and at Halden we have gone through a generation change. Now, the average age at the Halden Project is 38-39 years.

There is one thing that I am proud of, and that is when I walk in the corridor and see these 20 or 30-year old women and men from different countries working together, it's sort of a unique site. It's a different atmosphere than it was some years back. These are very competent people, and the responsibility of the people that work with me is to keep these good people well focussed. In order to do that, we need some perspective in front of us. As to short-term return of R&D, I got the message this morning. I hear of challenges, problems and so on, but I cannot go back to my people and talk about problems all the time. We also have the duty to maintain a good spirit in the working place.

To the R&D managers who are here today, one can say that if the Halden Project didn't exist, one would need to invent it. I think we have a good Center there, and I count on the continued support from our participants.

I don't know how much you were expecting from this presentation, I tried to go through it explaining how we are doing our work in a simple way. I also tried to give you our perspectives and, perhaps, our limitations. Flexibility, for example, is important. But, there is a limit to flexibility. You can change a mountaineer into a sailor, but you cannot change a mountain into the sea. So you have to start with the facilities and capabilities that you have and make them adaptable and flexible, keeping focus on what you have on one side and on what is needed on the other.

Thank you very much.

The History of Safety Research Programs

Herbert Kouts
Defense Nuclear Facilities Safety Board
Washington, DC

One of the benefits of passing years, if there are any benefits at all, is that they give me a license to stand in front of a group of younger people like this and tell about things that happened before you were involved. In fact, there will even be occasions when some of the audience will be captive audience and some where people will even have paid to attend, as is the case here, for the privilege, so to speak.

Now, I am going to take full advantage of the situation by going over some ancient history of the reactor safety research program, telling how some of the things began, drawing some conclusions, and even, at the end, giving a recommendation or two. And that is what I do quite well these days, because that is what the Defense Nuclear Facilities Safety Board does, it issues recommendations to the Secretary of Energy concerning safety in defense activities. Incidentally, if you happen to hear a lot in the first person singular in what follows, it will be because I am going to confine what I have to say to things in which I had a part to play and, therefore, where I can speak with some authority.

Now, let me just say at the outset that this meeting is the 26th in a series which I began in 1973, and I will say a little bit more about that later. Let me start the story in the middle 1960s, somewhat more than 30 years ago. I was a member of the ACRS at the time. It may even have been a time when I was Chairman of the ACRS. The initial flood of orders for nuclear plants had set in in this country. Orders were growing in number and designs were quite mobile.

The regulatory staff of the Atomic Energy Commission, which was later to become the Nuclear Regulatory Commission, was quite small in those days. A few of the staff members came into a meeting of the ACRS at one point and told us that some scoping calculations which had been done at the National Reactor Testing Station in Idaho, as it was then called, showed that in a shutdown reactor, if there were a complete loss of coolant, the core would melt, the molten core would fall on the bottom of the containment vessel, would in time melt through, fall on the floor, melt through and continue on. That was the birth of the "China Syndrome."

I left ACRS a little bit after that, but I stayed in touch. The next important step in the sequence occurred a couple of years later when Harold Price, who was head of the regulatory staff, called me to say that he had decided to form a group of people to look into the reality of that threat and to see what could be done about it. I told him, "don't look at me," and suggested William Ergin of Oak Ridge to head the study.

The Ergin Report, when it appeared, confirmed the existence of the threat and stressed the absolute importance of what later became called "engineered safety features" for nuclear plants to prevent the occurrence of a loss of coolant at all. Incidentally, these engineered safety features had already been proposed by designers of nuclear plants. Every designer had his own kind of emergency core cooling system, and it was in later years that I came to realize that the fact that these were being proposed to us at the ACRS was indication that the very good theorists who worked for these designers, were very much aware of the problem all along, and had taken the steps on their own.

The Ergin Report led the Reactor Development Division of the Atomic Energy Commission to initiate several programs of research directed to determining effectiveness of the emergency core cooling systems. There were loop-sized experiments called semiscale at Idaho. These were quite small facilities. They led to a large scale experiment with a pressurized water reactor called LOFT. The plan was that once codes for prediction of behavior of emergency core cooling systems had been constructed, the LOFT would test whether or not these codes were all right.

Programs for testing emergency core cooling of boiling water reactors were conducted by the industry at the Moss Landing facility of the Pacific Gas & Electric Company, and these were done under the aegis of an industry consortium headed by General Electric. Some related experiments were held on simulated fuel bundles at Oak Ridge and at Westinghouse, and the latter experiments were jointly funded by Westinghouse and the Atomic Energy Commission. Their purpose was to establish the behavior of fuel under the reduced pressure and increased temperature that would be experienced in the course of a loss of coolant accident. All of these operations were conducted in accordance with a program plan which had been written by people at the Idaho plant.

Now, at the time, the primary attention of the Reactor Development Division was focused on development of the fast neutron breeder, and the design and eventual operation of the Fast Flux Test Facility at Hanford, which was part of the fast breeder reactor program.

A prevalent view was that if the nuclear plants could be built and operated according to strict standards, there would be no loss of coolant accidents and no large accidents of any kind and, so, stress was placed on getting standards issued. A number of scientists engaged in the research program chafed under what they viewed to be inadequate funding, inadequate support of the safety research program underlying effectiveness of ECCS. And that unrest eventually reached the ears of Ralph Nader. Ralph Nader promptly instituted a case in court seeking to shut down the entire nuclear industry on the basis that one could not guarantee the safety of the nuclear plants. To address that question, the Atomic Energy Commission called a hearing, a public hearing. The courts decided at that point that they would defer decision on the outcome of the court case until the hearing was over.

Now, as is customary, regulatory hearings are of the judicial form. That is, intervenors act like prosecutors. They can call their own witnesses and they can cross-examine other witnesses. They did so with a vengeance in the ECCS hearings. There were about 26,000 pages of hearing record, and I know because I read every word of every one of those 26,000 pages.

A number of the researchers in the program testified and were cross-examined. Many of them expressed views that the information underlying the effectiveness of the ECCS systems was not adequate. Some of them even said that the analytical tools underlying analysis of effectiveness indicated that the ECCS would not be effective.

Early on, the five Commissioners, who were then headed by Jim Schlesinger, realized that they would not be able to arrive independently at conclusions as to the meaning and results of that hearing, so they asked two people, Herbert McPherson of Oak Ridge and the University of Tennessee, and me to review the results of the hearing and inform them what we thought ought to be done. We did so. The two of us, after long and hard work, issued a report to the Commission, which the Commission discussed and accepted, and issued verbatim as their opinion. The central feature of that opinion was a set of recommendations on characteristics of computer codes that were to be used to analyze the effectiveness of emergency core cooling systems for reactors.

The regulatory staff promptly converted that opinion to a regulation which they issued as Appendix K to 10 CFR 50. Appendix K became notorious after a time because it was extremely prescriptive. It told exactly how the calculations ought to be done, and it contained a number of strong conservatisms, but, unfortunately, all of that was really necessary at the time, because we had to circumscribe the knowledge of those characteristics of phenomena that had been brought to light in the course of the hearings.

The opinion also promised a safety research program to firm up the basis for acceptable features of computer codes and to permit relaxation of the Appendix K requirements in due time, if that were possible. The courts concluded at that point that the Commission's opinion satisfied them. The actions which were being taken were responsible and the safety research program offered to solve the remaining problems. Conduct of the safety research program came to be called "paying off the mortgage" on the nuclear power reactor program. The mortgage was going to be considered paid off if and when it was shown that ECCS design codes were capable of predicting the performance of these systems.

By this time, Schlesinger had left the Atomic Energy Commission and Dixie Lee Ray had become the Chairman. She drafted me in early 1973 to head the research program as Director of a new Division of Reactor Safety Research and, after ineffective struggle to escape, I agreed to do so.

With the argument that the regulatory staff needed an independent basis for deciding the validity of technical propositions brought to it, I managed to obtain resources needed for a rapidly growing program of research in reactor safety. The central feature of this program didn't differ too much at first from what had been found in the old program under the Reactor Development Division, but it was certainly much more intensive.

We added to the programs that were directed toward designing the characteristics of the codes, programs to develop new codes, and we even added studies in new areas such as fast breeder reactor safety and the safety of high temperature gas-cooled reactors.

I would be remiss if I did not mention at this point the singularly important contributions of Dr. Lon Sun Tong in the course of all this. Dr. Tong was my righthand man in the design of the program, and I find with great sorrow that he is quite ill now and wish him the best.

I left the program in 1976, a year after it had been transferred to the new Nuclear Regulatory Commission. It continued to be enlarged under Saul Levine and Bob Budnitz, and the mortgage was finally paid off in the series of LOFT tests that were conducted, when the codes that had been developed predicted the performance of ECCS in the LOFT tests conservatively. I won't go into any details on this because many of you participated in those phases of the program and I was off doing other things at the time.

The important thing to note is that the nuclear power plants operating today owe their continued existence to the Safety Research Program that is represented here today. In the course of conducting that program, you developed an entirely new branch of engineering, which is the engineering of nuclear reactor safety, and it is new.

Now, I would like to turn to another feature of nuclear plant safety in which I have been a fringe player over many years. I go even farther back in time for this. In 1956, the first commercial nuclear power plants were about to go online in the United States. Congress struggled with legislation to protect the fledgling industry from severe financial damage that might occur if an accident to one of the nuclear plants took place. The Commission asked Brookhaven National Laboratory to conduct a study to estimate the consequences of such an accident and, if possible, the probabilities.

A group was formed within my division at Brookhaven, a group headed by Kenneth Downes, to attempt to do what had been asked. The result was published as a document called "WASH 740." Unfortunately, WASH 740 was not extremely helpful, although it was truthful. It concluded that it was not possible to predict the probability of an accident of this kind, partly because the designs didn't even yet exist for the commercial plants.

It concluded that if the containment building remained intact following an accident, there would be essentially no consequences at all to the surrounding public, but if the containment could not be relied on, you could only estimate upper limit consequences, and these upper limits, based on release of all fission product gases and a large portion of the iodine and some of the other fission products, would be tremendous – would be horrific.

In an example of common sense legislation, Congress issued, following this, what was known as the Price Anderson Act. The Price Anderson Act sensibly ignored the higher limit estimates of WASH 740. The Act established a fund which was contributed to by the nuclear utilities, who directed a fraction of their income from nuclear electricity to the fund, and the fund was to cover expenses following a conceptual accident and to be used for public compensation if such an accident took place. It turned out to be enough for the industry to proceed, and the construction of the plants went on. In 1966, the Act was to expire since it had to be renewed every ten years, and a second study at Brookhaven for this purpose led to the conclusion that the first results could not be improved upon appreciably.

Some years later, as time approached for a second renewal, I got another phone call from Harold Price and he asked me if I would head a study to try to get some estimate of probabilities. I told him I didn't think it could be done. I said, no, thank you, get somebody else, and I suggested Norman Rasmussen of MIT. I thought he might be able to do it if anybody could. Well, you all know the results. Norm, with the help of Saul Levine, and other contributors, did an incredible job. He succeeded in accomplishing the impossible and WASH 1400 was born.

Dixie Lee Ray also asked me to fold the management of Rasmussen's operation into my new Division of Reactor Safety Research, which was easy because Saul Levine at the time was my deputy. Before the report was issued, I had read every word of it and of its voluminous appendices, and I did much of the technical editing on that report, so I felt that I understood it quite well.

After I left the Safety Research Program, I continued active in numerous meetings and discussions on WASH 740 and on the successor program that led to NUREG-1150. And this all by way of my leading up to some remarks on the use of risk assessment. Some of these may not be agreed to by people who are here today.

There is a lot of discussion these days directed to risk-based regulation. There is such a thing, but I think there is more danger of overuse of this concept than is to be attached to its under-use. Rasmussen and Levine were always firm in their statements that the large error bars in any such conclusions should be the basis for care in use of bottom line estimates of risk. Risk estimates should be used primarily as the basis for regulation that remains solidly founded on mechanistic requirements, with emphasis on "a" basis rather than "the" basis. It has to be one of a number of bases for decisions to be made. I thoroughly agree.

Risk analysis can be very powerful in a number of ways. It can be used to identify vulnerabilities in design or in operation of plants, by searching for major contributors to accident probability and consequences. It can also be used as an aid to choice among alternatives in design or operation because many contributors to the error bars cancel out in such a comparison. But where risk analysis is used, even in ways like this, conclusions should always be subjected to reality tests. That is, you should always ask the question -- Does this result make sense? Sometimes it won't. And if it does not make sense, what is the source of that problem? And you can learn a lot by investigating the source of the problem that way.

It is not only important that regulators recognize these limits to the techniques, it is important that they continue to pass recognition of that information on to the less technically prepared groups who try to push use of risk methods too far, and I read that there are now pressures from some parts of Congress to do risk-based regulation. I don't know what they mean.

Some final remarks. The first of these meetings of the reactor safety research community was held in 1973. It was attended only by a sparse number of researchers from a few laboratories. This was just after the trauma of the ECCS hearings. Attendees were afraid to talk about their research because they had the view that if whatever they had to say could be interpreted as critical of the safety of nuclear plants, their jobs were at stake.

When I asked for people to speak, I got silence. I had to harangue the group, tell them that I not only had to hear what they had to say, but I insisted they had better start publishing the results of what they were doing in the open literature or there would be hell to pay, and then I got a little action. A few people did begin to talk about their work, and a trickle of publications began to appear in successive months, which eventually swelled into a torrent.

Well, this meeting, like so many of its predecessors, shows that the problem of reluctance in 1973 no longer exists, and I say to that, just -- hallelujah. And I thank you all for the opportunity to review a little memory today. Thank you.

Evaluation of Margins in the ASME Rules for Defining the P-T Curve for a RPV

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Abstract

The pressure-temperature (P-T) curve controls the upper-bound to the permissible operating envelope for a reactor pressure vessel (RPV) during the normal start-up and cool-down transients. The P-T operating envelope is progressively restricted because of irradiation embrittlement of the RPV material. In recent years, a number of electric utility companies have reported that the plant-specific P-T operating envelope has become so restricted that operation of the reactor during the heat-up and cool-down transients has become very difficult. An evaluation of the inherent margins in the current American Society of Mechanical Engineers (ASME) Code P-T curve rules was made to determine if they can be modified so as to increase the available P-T operating envelope while retaining adequate safety factors. The evaluation was made in the U. S. Nuclear Regulatory Commission (USNRC)-funded Heavy Section Steel Technology (HSST) program at Oak Ridge National Laboratory (ORNL) using results from a number of NRC-funded research programs. Best estimate allowable pressure (P_{BE}) calculations included all loading conditions and crack-front locations with the safety factor on pressure loading set to 1.0. The P_{BE} value obtained when the material fracture toughness was set at the lower-bound to the shallow-flaw database for RPV materials was higher than the allowable pressure obtained when K_{Ia} is replaced by K_{Ic} in the P-T curve rules. This finding supports opening of the P-T operation envelope by using K_{Ic} instead of K_{Ia} in the ASME P-T curve rules. It is important to recognize, however, that lower-bound to the shallow-flaw fracture toughness database is controlled by results from clad cruciform biaxial-loading tests conducted at normalized temperatures ($T\text{-}RT_{NDT}$) not less than -40°F. A potential exists for the estimated shallow-flaw lower-bound fracture toughness to be further adjusted if data from clad cruciform biaxial-loading tests become available for the normalized temperature range $-200°F \leq T\text{-}RT_{NDT} \leq -170°F$.

1. Introduction

Principal features of the RPV P-T operating envelope are shown in Fig. 1. The upper-bound to the P-T envelope is defined by the vessel material P-T curve [1], modified by the operating characteristics of the low temperature overpressure protection (LTOP) system required by the USNRC Standard Review Plan

[2]. The lower-bound of the P-T operating envelope is set by the pressure required at a given temperature to prevent cavitation of the main coolant pumps, and/or activate the pump seals. Adjustment of the P-T curve to accommodate the effects of irradiation embrittlement of the RPV material results in a severe restriction of the P-T operating envelope. A concern exists that the potential for operator errors and opening of relief valves is greater when the operating envelope is severely restricted. This concern led to an investigation to determine if a basis exists for opening-up the P-T operating envelope while preserving essential RPV safety margins. The specific objective for the investigation was to determine if margins in the P-T curve analysis procedures were sufficient to support the substitution of the ASME K_{Ic} curve in place of the K_{Ia} curve in the analysis procedures of Ref. 1. The allowable pressure determined using this substitution is designated $P_{NEWCODE}$. The approach taken in the P-T curve margin study was to use best estimate data from RPV structural integrity research programs as input to calculate a best estimate allowable pressure P_{BE}. Demonstration of margins sufficient to justify the proposed ASME Code change requires that $P_{BE} \geq P_{NEWCODE}$.

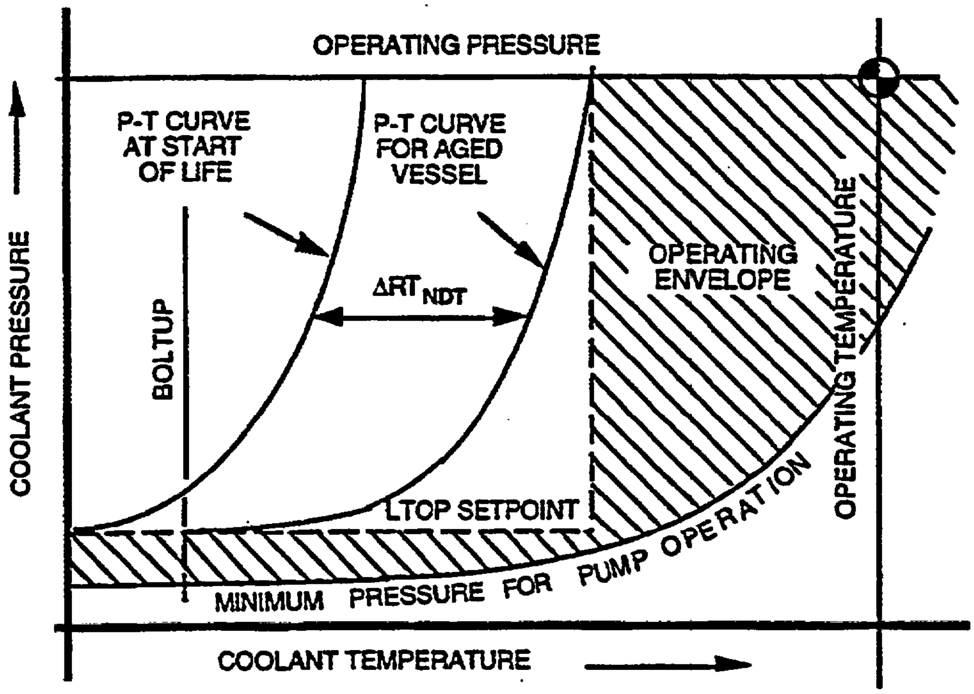

Fig. 1 The P-T operating envelope for a RPV is progressively restricted because of adjustment of the P-T curve to accommodate the effects of irradiation embrittlement of the RPV material.

2. P-T Curve Margin Assessment Rationale, Method and Scope

2.1 Rationale

Fracture technology research conducted since the P-T curve rules of Ref. 1 were formulated has identified a number of areas where those rules include unnecessary levels of conservatism, and some areas where the level of conservatism was less than had been anticipated. Examples of unnecessary conservatism include the use in the P-T curve analyses of, (a) the lower-bound (K_{Ia}) crack arrest fracture toughness curve, and (b) a conservative inner surface flaw, having a depth corresponding to 25% of the RPV wall thickness (1/4t). Use of the K_{IR} curve reflects an early concern [3] that fracture could originate from local brittle zones in the RPV as a pop-in and propagate in a dynamic manner to the 1/4t depth. Fracture toughness tests on irradiated weld material, however, showed that data from specimens which had pop-ins fell within the scatter band of data from specimens which failed with no prior pop-ins [4]. Use of the K_{Ia} curve to safeguard against failure initiating from a local brittle zone was shown to be unnecessary.

Specification of the 1/4t flaw depth was made in 1972, when data on flaw-size-distribution and flaw density in RPV's were scarce [3]. At the time it was specified, the 1/4t flaw was considered to be very conservative. Since that time, data from non-destructive examinations (NDE's) of a significant number of "LWR vessels, made essentially to ASME III rules alone," have been published [5]. While a number of flaws in the size range 0.5" ≤ a ≤ 1.0" were reported, no flaws with a radial depth greater than 1.0" were found. A similar finding was obtained from the recently completed NDE and destructive examination, conducted by Dr. S. Doctor of Pacific Northwest National Laboratory, on the RPV from the Pressure Vessel Research User's Facility at ORNL [6]. Results from these investigations support the use of a 1" deep flaw in the P_{BE} evaluations.

The rules of Ref. 1, and the sample problem defined later in this paper, both define a 6:1 aspect ratio surface-breaking flaw. A check was made in this evaluation to determine if the assumption of a surface-breaking flaw introduced an element of unnecessary conservatism. Any significant flaws in the inner surface structural material of a RPV (plate, forging, and weld) would be introduced during fabrication operations, which would have been completed before the application of stainless steel cladding to the inside surface of the RPV. A surface-breaking flaw, which existed in the inside surface of the RPV prior to cladding, would be converted to a sub-clad flaw by the cladding process. Stress intensity factors induced by P-T loading on a sub-clad flaw would be substantially lower than those induced in a surface-breaking flaw by the same loading. Cleavage-controlled allowable pressures for a subclad flaw could, therefore, be significantly higher than the allowable pressures for a surface-breaking flaw. For P-T curve margins based on a subclad flaw to be valid, however, it must be shown that the cladding above the flaw remains integral throughout the operating life of the RPV. An investigation of the structural integrity of the cladding above a sub-clad flaw, based on both tensile instability and ductile tearing failure modes, was included in this P-T curve margin assessment.

2.2 Method

The P-T curve analysis rules of Ref. 1 use the following equation to define the P-T curve for a RPV. K_{Im} values are determined using conditions at the deepest point on the flaw and the applied stress intensity factors (K_{Im} and K_{It}) are determined at the same location.

$$2K_{Im} + K_{It} < K_{Ia} \qquad (1)$$

Where: K_{Im} is the stress intensity factor produced by pressure-induced membrane loading in the RPV shell (ksi√in.).

K_{It} is the stress intensity factor produced by a radial thermal gradient through the wall of the RPV (ksi√in.).

K_{Ia} is the lower bound stress intensity factor obtained from crack-arrest tests (ksi√in.).

The factor of two applied to K_{Im} in equation 1 is the means by which allowance is made to accommodate sources of stress intensity factor not included in equation 1. Sources of stress intensity factor not included in equation 1 include, residual stresses in the RPV structural welds (K_{IRS}), stresses produced by pressure on the crack face (K_{IPCF}), and stresses resulting from differential thermal expansion between the stainless steel cladding and the low-alloy steel RPV shell material (K_{ICB}). In the following evaluations, K_{IRS} and K_{ICB} are included, but K_{IPCF} is not included because it has only a minor influence on the results.

Equation (1) can be modified as follows to produce the allowable pressure (P_{CODE}) at a given normalized temperature ($T\text{-}RT_{NDT}$).

$$P_{CODE} = (K_{Ia} - K_{It}) / (2C_P) \tag{2}$$

Where: C_P is the stress intensity factor at the deepest point on the 1/4t flaw produced by a 1 ksi pressure loading in the RPV (ksi√in.)

For the calculation of P_{BE} equation (2) was modified as follows to, (a) include all significant sources of applied stress intensity factor, (b) eliminate the safety factor of 2 on pressure loading, and (c) consider a number of definitions of the material fracture toughness (K_{Ix}).

$$P_{BE\theta} = (K_{Ix\theta} - K_{ISC\theta}) / C_{P\theta} \tag{3}$$

and

$$K_{ISC\theta} = K_{It\theta} + K_{IRS\theta} + K_{ICB\theta} \tag{4}$$

Where: $P_{BE\theta}$ is the best estimate allowable pressure (ksi).
$K_{Ix\theta}$ is the normalized-temperature-dependant fracture toughness for material definition x (ksi√in.).
$K_{ISC\theta}$ is the sum of all strain controlled stress intensity factors (ksi√in.).
$C_{P\theta}$ is the stress intensity factor produced at a specified location on the reference flaw in a RPV by internal internal pressure of 1 ksi.
$K_{It\theta}$ is the applied stress intensity factor produced by a through-the-wall temperature gradient (ksi√in.).
$K_{IRS\theta}$ is the applied stress intensity factor produced by residual stresses in the RPV structural welds (ksi√in.).
$K_{ICB\theta}$ is the applied stress intensity factor produced by clad-base material differential thermal expansion (ksi√in.).

The designator θ in the above definitions indicates the critical location on the crack front. Introduction of this designator is necessary because the position of the critical location on the crack front is influenced by the choice of the material fracture toughness definition (x). The critical location for a relatively high material fracture toughness ($K_{Ix\theta}$) is usually at the deepest point on the crack front, but it moves to a location near the clad-base metal interface when $K_{Ix\theta}$ is lower.

2.3 Scope

Fracture toughness of the RPV material was the primary variable in this P-T curve margin assessment. Fracture toughness curves used in the margin assessment included the K_{Ic} and K_{Ia} curves defined in the ASME Code, plus additional curves designated, K_{Icm}, K_{Ic}', and K_{Icsf}. The mean curve (K_{Icm}) through the

EPRI K_{Ic} database was defined in Ref. 7. The data plots of Ref. 7 also show a number of points below the ASME K_{Ic} curve in the normalized temperature ($T-RT_{NDT}$) range of interest in a P-T curve analysis. The K_{Ic}' curve was obtained by adjusting the ASME K_{Ic} curve downwards by 4.5 ksi√in, so that it became a true lower-bound curve for $T-RT_{NDT}$ values down to –200°F. The 1.0" deep 6:1aspect ratio flaw specified for the reference problem has an absolute value of flaw depth, a, that is slightly greater than most of the flaw depths in the shallow-flaw specimens used to generate the shallow-flaw fracture toughness database for RPV materials [8-13]. The reference flaw does, however, have an a/W ratio of 0.11 which is in the mid-range of a/W values for these shallow-flaw specimens. The judgement is then that toughness values developed in the shallow-flaw test programs should envelope the minimum toughness value expected for the reference flaw. The lower-bound to the shallow-flaw fracture toughness data (K_{Jcsf}) is 7.5 ksi√in above the ASME K_{Ic} curve at all $T-RT_{NDT}$ values. The K_{Icsf} curve was generated by adjusting the ASME K_{Ic} curve upward by 7.5 ksi√in so that it became a lower-bound curve for shallow flaw fracture toughness data.

3. Materials Properties

3.1 Fracture toughness bounding curves

The mean fracture toughness curve used in these studies was taken from the evaluation of the EPRI K_{Ic} database as described by Nanstad, et al., in Ref. 7. After a thorough check of the toughness values reported in the original EPRI database, some corrections and deletions of invalid data were made. Using the corrected database, a mean curve was developed using a non-linear regression analysis and an equation of the form of the ASME Section XI K_{Ic} curve. The database and mean curve, with constants as reported in Ref. 7, are shown in Fig. 2.

The ASME Section XI K_{Ic} and the K_{Ic}' curves are also shown in Fig. 2. The K_{Ic}' curve was constructed by shifting the Section XI K_{Ic} curve downward to pass through the lowest data point of the EPRI data set. Only data above $T-RT_{NDT}$ = -200°F were considered since lower normalized temperatures are not relevant for P-T curve evaluations. Using the philosophy of a "below all points" curve, the controlling data point was found to be for HSST Plate 01 material tested at $T-RT_{NDT}$ = -170°F, K_{Ic} = 29.4 ksi√in. This point is 4.5 ksi√in below the ASME Section XI K_{Ic} curve at that normalized temperature. The equation for the K_{Ic}' curve is then,

$$K_{Ic}' = 28.7 + 20.734\exp[0.02(T - RT_{NDT})] \tag{5}$$

Note that the curves discussed above represent the toughness for deep-flaw specimens. For this study, it was desired to develop a shallow-flaw lower-bound fracture toughness curve comparable to the ASME Section XI K_{Ic} curve. The curve developed could then be used for the P-T curve margin evaluations. Test programs to investigate shallow-flaw fracture toughness in RPV materials have been performed by ORNL and the Naval Surface Warfare Center (NSWC) [8-13]. The specimens used were beam-type structures with either 2-D or 3-D through-surface flaws. For the ORNL testing, specimens were subjected to uniaxial or biaxial loading. In general, it was concluded that shallow flaws have greater scatter and higher mean fracture toughness than deep flaws, but that there appeared to be little difference in the lower-bounds for either data set [14]. For the purposes of this study, the shallow and deep-flaw data sets were re-evaluated to quantify, as best possible, the comparative effects on fracture toughness of shallow

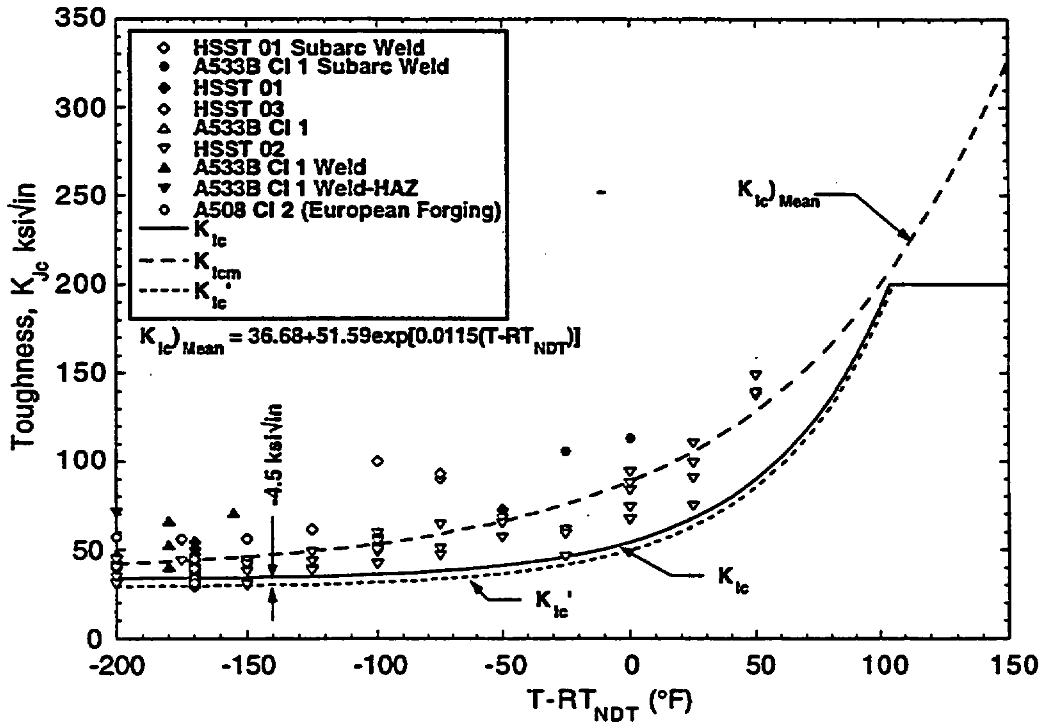

Fig. 2 EPRI database showing position of K_{Ic}, K_{Ic}', and K_{Icm} fracture toughness curves.

versus deep flaws. The value of the test programs performed by ORNL and NSWC is that comparative shallow and deep-flaw specimens were tested using the same materials and test conditions. The materials were all A 533 B steel with heat treatment applied to some heats to achieve elevated yield strength. Thus, direct inferences may be drawn as to shallow-flaw effects. To evaluate shallow-flaw effects, the data were restricted to specimens with 2-D surface flaws to have similarity in specimen geometry. The 2-D flaw data are shown in Fig. 3. Examination of this figure reveals immediately the conclusion that shallow-flaws exhibit greater scatter in fracture toughness, than do deep flaws. To quantify the difference in the mean toughness between shallow and deep-flaws, these data were analyzed using standard non-linear regression methods. Equations of various forms were tried, but the best correlation was obtained using a simple exponential equation of the form,

$$K_{Jc} = a_1 \exp[a_2(T - RT_{NDT} + a_3)]. \tag{6}$$

An equation of this form penalizes the mean toughness values at low temperatures since, in the limit, it approaches zero as an asymptote. In this sense, its use is conservative. It does, however, have the advantage that no artificially imposed lower-bound is applied. Regression analyses yielded the constants as shown in the Table 1 below. The mean curves for both shallow and deep-flaws are shown in Fig. 3. The lower-bound for the shallow-flaw curve was constructed by inspection, rather than statistical treatment, i.e., utilization of some multiple of the standard deviation of the fit. The lower-bound curve was established as a fractional multiplier of the mean curve, with the multiplier being selected such that the resulting curve bounded all shallow-flaw data. The resulting lower-bound is shown in Fig. 3. It was

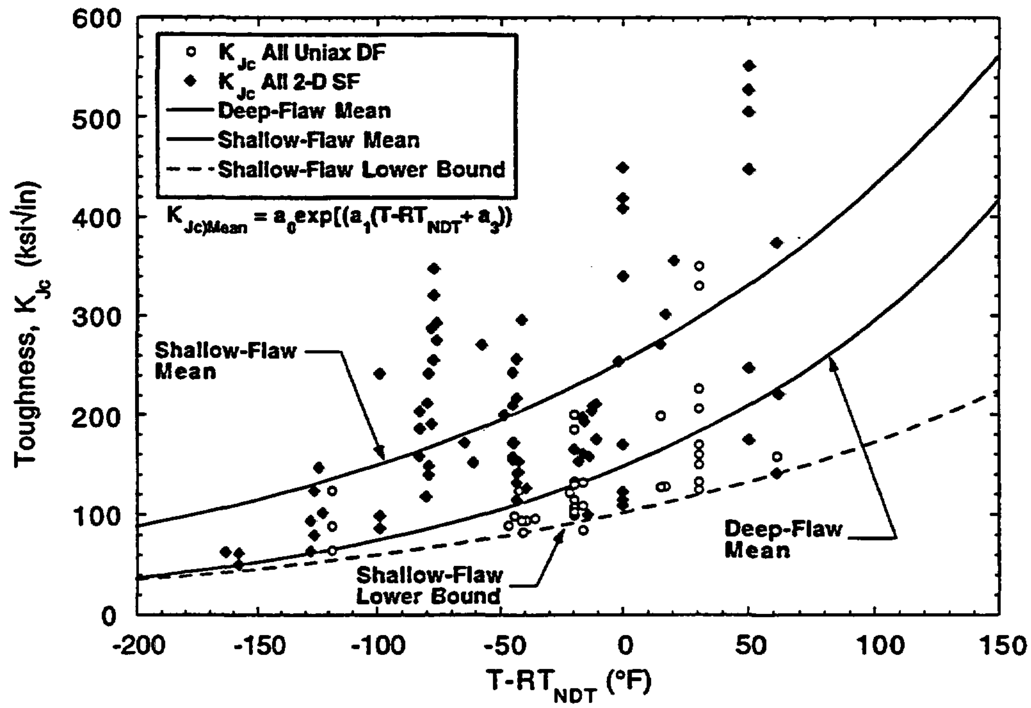

Fig. 3 Comparison of shallow- and deep-flaw data sets with mean curves showing elevation of shallow-flaw fracture toughness.

observed from the data shown in Fig. 3 that, where sufficient numbers of shallow and deep-flaw specimens were tested under the same conditions, the mean value and lower-bound of the shallow-flaw data are greater than the mean and lower-bound of the deep-flaw data. (See for example the tests performed over the range $-50°F \leq T\text{-}RT_{NDT} \leq 0°F$.) For P-T limit evaluations contained in this paper, the lowest normalized temperature ($T\text{-}RT_{NDT}$) is greater than −170°F. At −170°F, the shallow-flaw lower-bound curve shown in Fig. 4 is 7.5 ksi√in above the ASME Section XI curve. Since the Section XI K_{Ic} curve is programmed into the analysis procedures, for ease of application the K_{Jcsf} lower- bound curve was established by raising the K_{Ic} curve by 7.5 ksi√in as is shown in Fig. 4.

Also, included in Fig. 4 is a set of 3-D shallow-flaw clad cruciform beam test results for RPV weld material. These specimens were tested under uniaxial (0:1) and biaxial (1:1) loading from which it was determined that the constraint associated with biaxial loading reduces fracture toughness compared to uniaxial loading, as shown in Fig. 5. The K_{Jcsf} curve in Fig. 4 bounds all the clad cruciform results, with those lying closest to the curve being biaxially loaded tests. It is then concluded that the K_{Jcsf} lower-bound curve shown in Fig. 4 provides a rational and conservative lower-bound to RPV material shallow-flaw toughness data.

As was previously discussed, the K_{Ia} curve was specified for P-T curve analysis because of concerns about the effect of local brittle zones on static fracture toughness [7]. Recent test results indicate that local brittle zones have no significant effect on static fracture toughness [4]. It is then concluded that current activities within the ASME Code to permit use of the K_{Ic} curve instead of the K_{IR} curve have a sound technical basis.

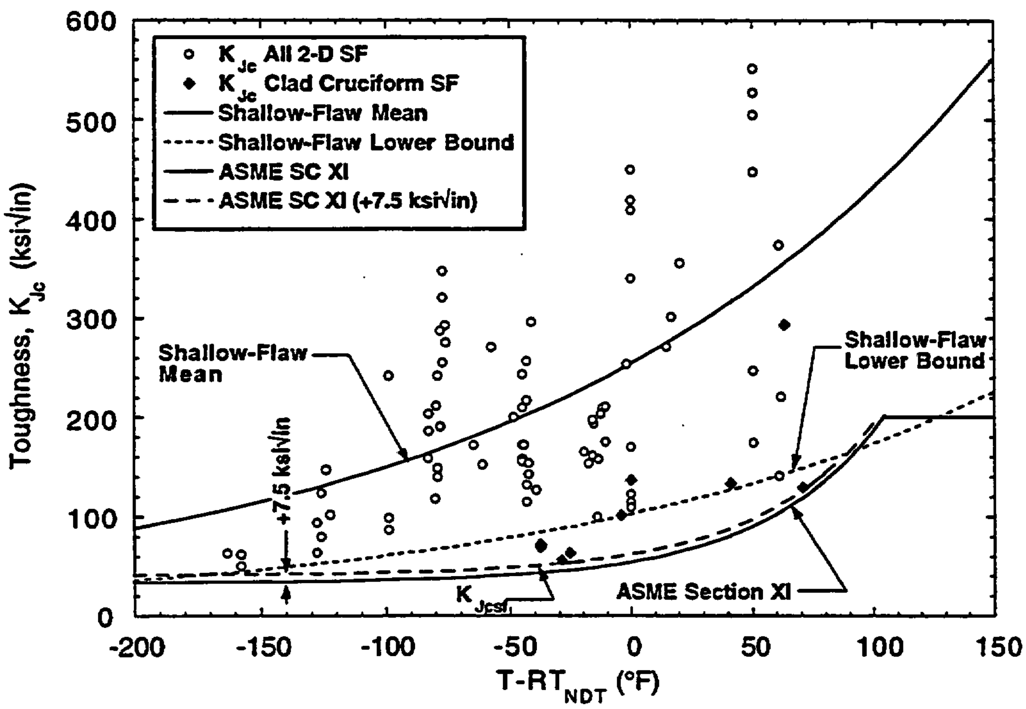

Fig. 4 Shallow-flaw fracture toughness data set with recommended lower-bound, K_{Jcsf}.

Table 1

Constants for the Shallow-Flaw and Deep-Flaw Mean Curve Equations

Flaw Type	a_1	a_2	a_3
2-D Shallow	51.211	0.0069	153.79
2-D Deep	80.465	0.0053	216.96

3.2 Fracture Behavior of Clad Material

In order to evaluate the behavior of subclad flaws and their influence on RPV integrity, some definition of the clad layer failure criteria was needed. For this purpose, data generated using a special Jo-Block specimen was used. The Jo-Block specimen was first conceived for the purpose of evaluating the fracture properties of cladding over a subclad flaw. The specimen consists of two machined steel blocks with the ends butted together to form a "crack." The name Jo-Block was derived from Johansson blocks, which are precision-machined gage blocks used for calibrating instruments, etc. Two machined blocks are butted together, and cladding is deposited on opposite faces of this assembly across the interface between the two blocks. Subclad flaw tips are then generated at the intersection of the block interfaces and the

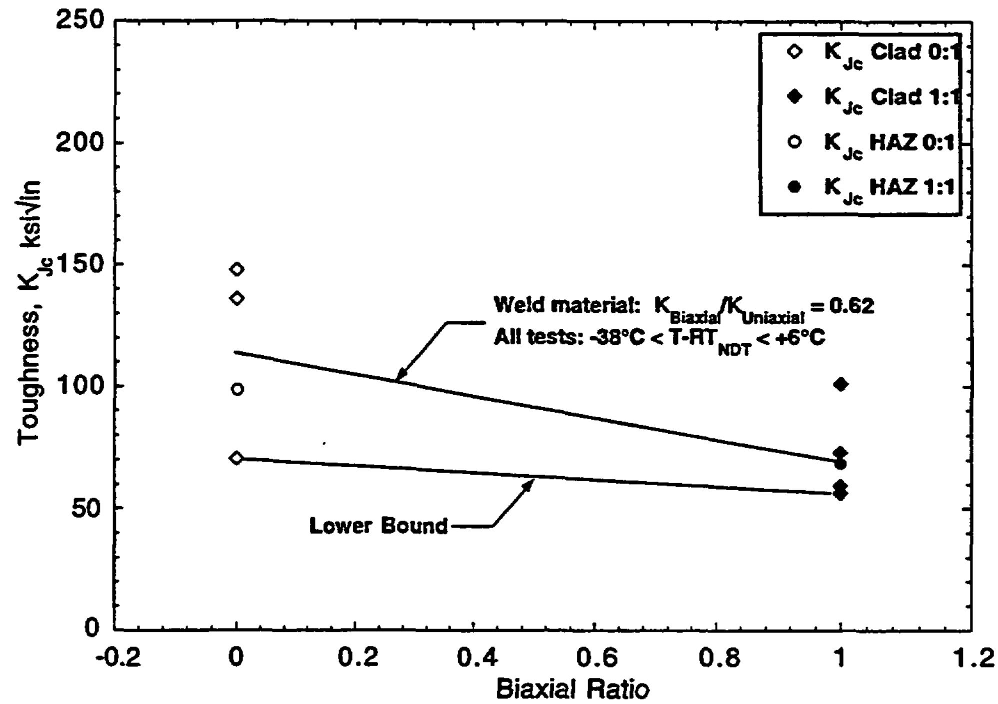

Fig. 5 Biaxial loading causes a reduction in the mean and lower-bound fracture toughness in RPV weld material.

overlying cladding. The quality of the butted-machined surfaces, the care used in fit-up and the restraint against distortion during cladding determine the final width of the crack. In practice, essentially zero-width cracks were obtained with relatively "sharp" flaw tips, i.e., tip radii in the range of 0.0008–0.003." Shrinkage of the cladding during cooling usually caused additional subsurface crack extension (microcracking) such that the flaw tip had characteristics of a "true" crack. The final crack configuration resembles a cross-section of the clad/base metal interface region of a two-dimensional subclad flaw in a vessel wall. Since the cladding is applied in the same manner that vessel cladding is applied, the cladding retains many of the characteristics that cladding on a vessel wall would have. The details of fabrication of these specimens are contained in Ref. 15. An isometric drawing of the Jo-Block specimen is shown in Fig. 6.

The effective yield stress and the rupture strain of the cladding and the crack opening displacements (COD) beneath the cladding were determined. The amount of clad surface stretching directly over the flaw tip was measured using conventional foil-type strain gages, and clip gages on the sides of the specimen were used to measure COD beneath the clad layer and at the specimen midplane.

The specimens were tested at both room temperature, and at –200°F. A typical plot of front-surface clip-gage readings versus load for a specimen tested at –200°F is shown in Fig. 7. The maximum load for this test was 31.7 kips giving an engineering ultimate strength of 83.5 ksi. Plastic instability in the clad layer occurred for crack-tip opening displacements in the range of approximately 0.010- to 0.020." Failure of the cladding occurred at a COD greater than 0.032." Specimen failure was characterized by failure in only one ligament of cladding. When yielding occurred in one clad surface as shown in Fig. 7, there was little further increase in load. The yielded surface became the "weak link" and continued to stretch up to

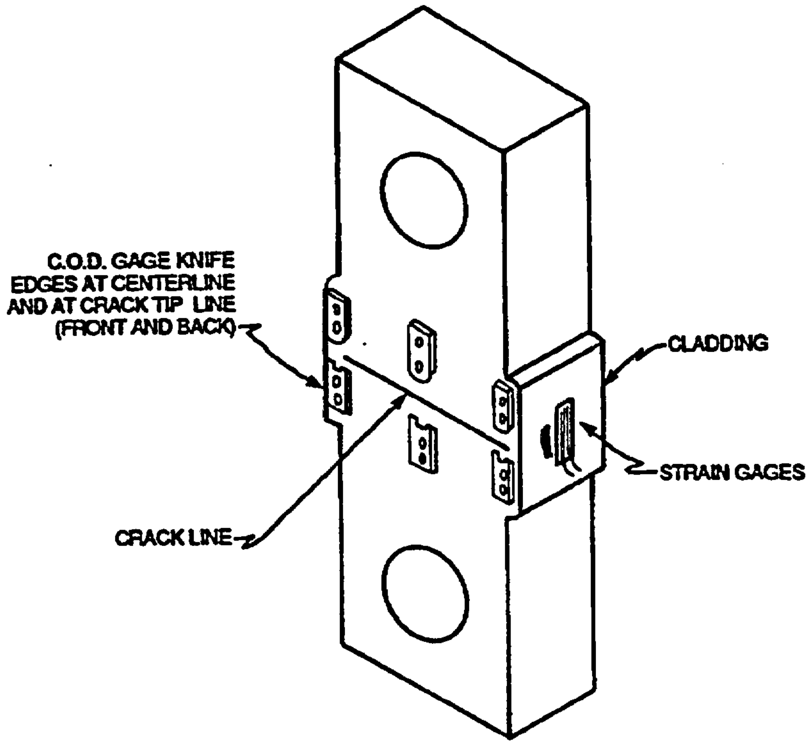

Fig. 6 Isometric drawing of Jo-Block specimen used to measure properties of clad layer over subclad flaw.

failure of the cladding. Since there was little additional load increase, axial deformation and COD of the opposite surface essentially stopped. Figure 8 is an enlarged view of the initial part of this loading curve, which shows three distinct regions of behavior. The first of these, which extends up to a load of approximately 6 kips is the elastic response of the entire specimen (base metal and cladding), and is due to the pre-load caused by the cladding residual thermal stress. This is indicated as the "Elastic Bar" line in Fig. 8. The second region is the near linear elastic behavior of the cladding acting alone. Nonlinear response of the cladding initiates at approximately 18 kips. The last region is the fully non-linear plastic behavior of the cladding. For the cladding, taking the deviation from non-linear behavior as the "yield" point, an effective clad yield stress for this configuration of 47 ksi would is obtained.

The failure of these specimens was characteristically by ductile tearing of the cladding and plastic instability even at –200°F. There was no evidence of cleavage-type fracture. The deformation results above are used in the assessment of clad instability given in Section 4.

3.3 Residual Stresses in Cladding and Weld

Residual stresses were measured in the longitudinal weld and in the clad layer of an RPV shell segment. Procedures utilized and the results obtained are briefly described below.

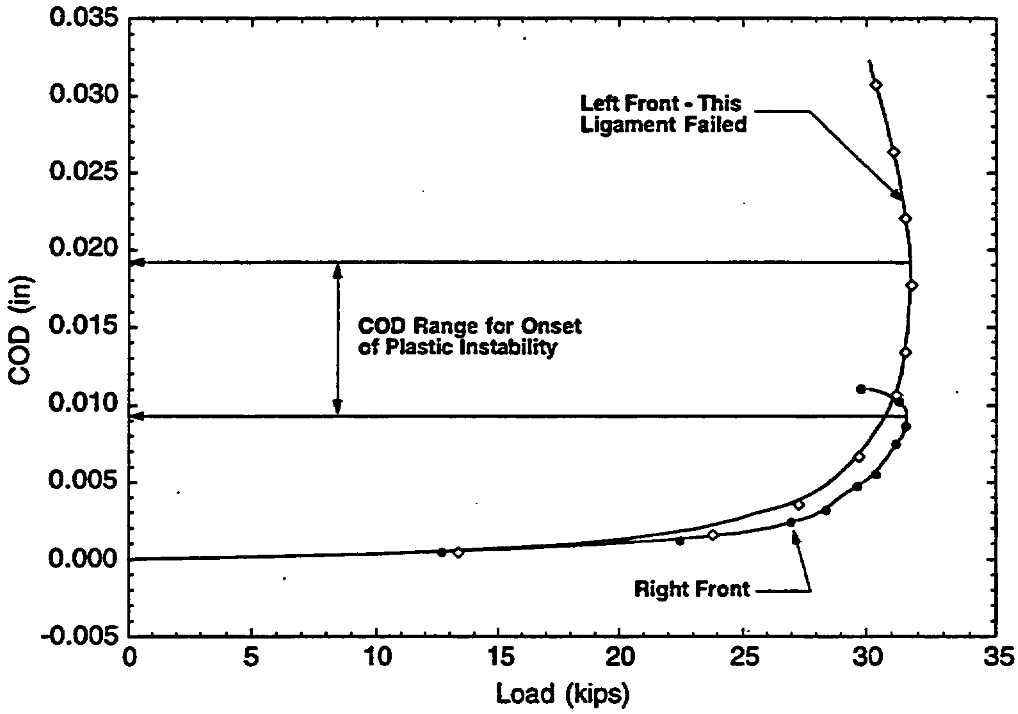

Fig. 7 Load-COD behavior of typical Jo-Block specimen showing highly ductile deformation characteristics of clad layer.

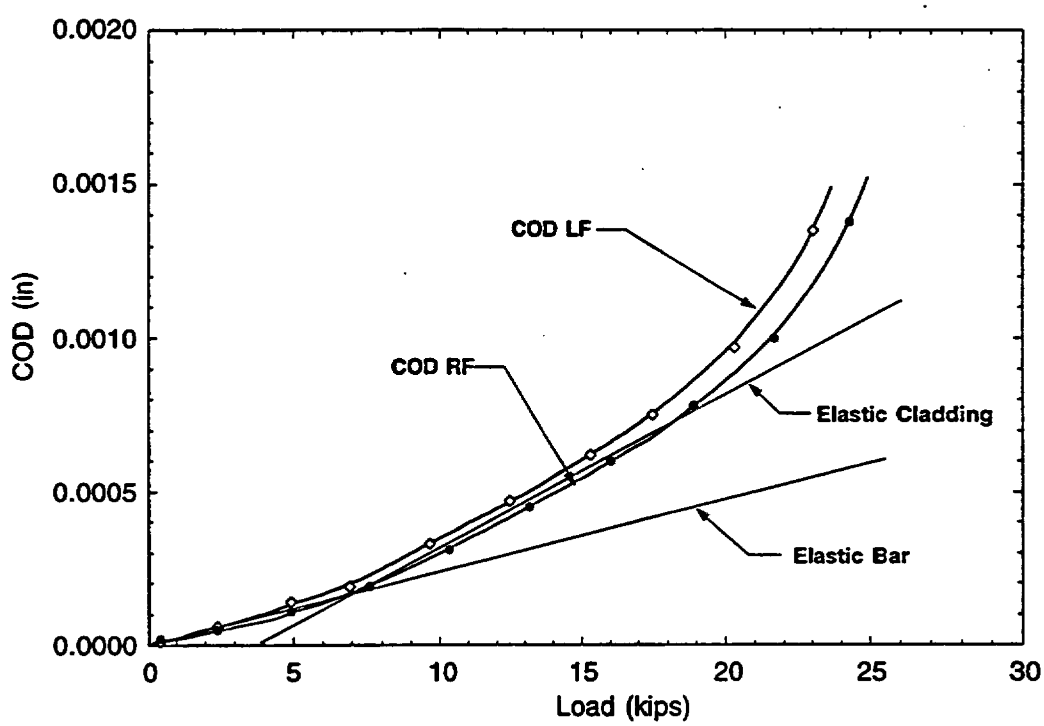

Fig. 8 Initial elastic region of Load-COD behavior for Jo-Block specimen shown in Fig. 8.

3.3.1 Longitudinal weld residual stress

The residual stresses in a RPV structural weld stem from (a) the clad-shell differential thermal expansion (DTE) and (b) the residual stresses, generated by the structural welding process, that are not completely relaxed by the post-weld heat-treatment [11]. Data required for calculation of these residual stresses were obtained by cutting a radial slot in the longitudinal weld in a shell segment from a RPV, and measuring the deformation of the slot width after cutting. The measured slot openings are assumed to be the sums of the openings due to the clad-base material differential thermal expansion (DTE) and the weld residual stresses. To evaluate the residual stresses in an RPV structural weld, a combined experimental and analytical process was used. Slot opening measurements were made during the machining of full-thickness clad beam specimens with 2-D flaws. The blanks measured 54" long (circumferential direction), 9" wide (longitudinal direction), and 9" thick (radial direction). The blanks were cut so as to have a segment of a longitudinal seam weld from the original RPV at the mid-length of the blank. Using the wire-EDM process, a slot was cut along the weld centerline in a radial direction from the inside (clad) surface of the blank. Measurements were made on three specimens having final slot depths of 0.045," 0.90," or 4.50," respectively. After machining, the widths of the slots were measured along each radial face of the blanks. The results for the specimen with a 4.50" deep flaw are shown in Fig. 9. Finite element analyses were used to develop a through-thickness stress distribution that gave a deformation profile matching the measured values. This distribution is shown in Fig. 10, where the contribution from clad and base DTE has been removed.

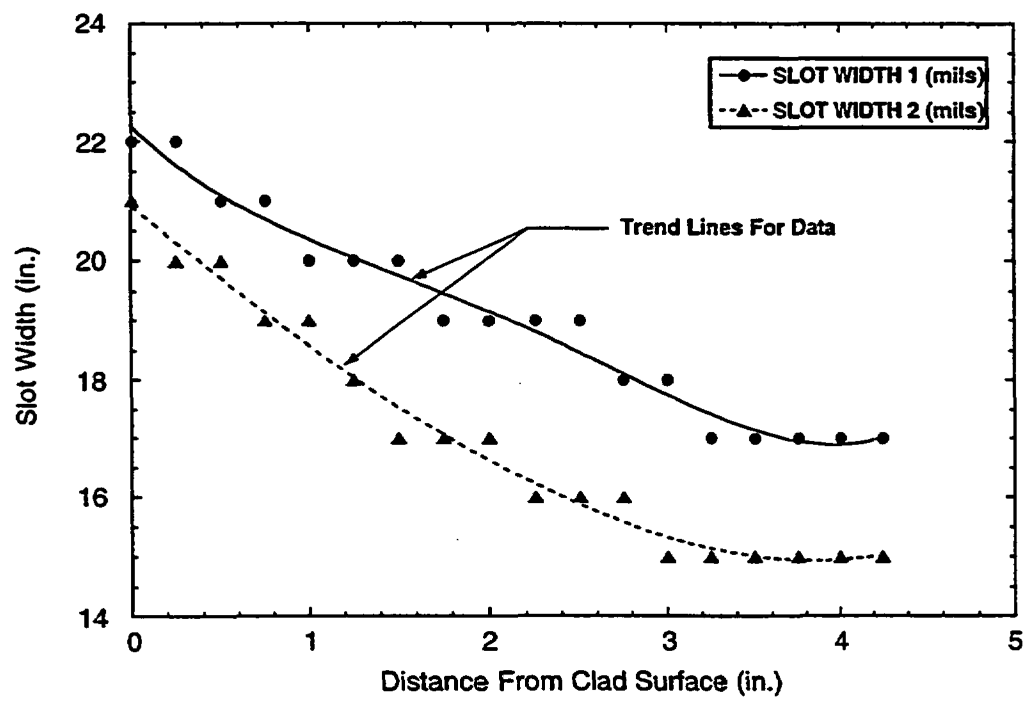

Fig. 9 Slot-opening measurements made for RPV weld specimen with 4.5" deep flaw.

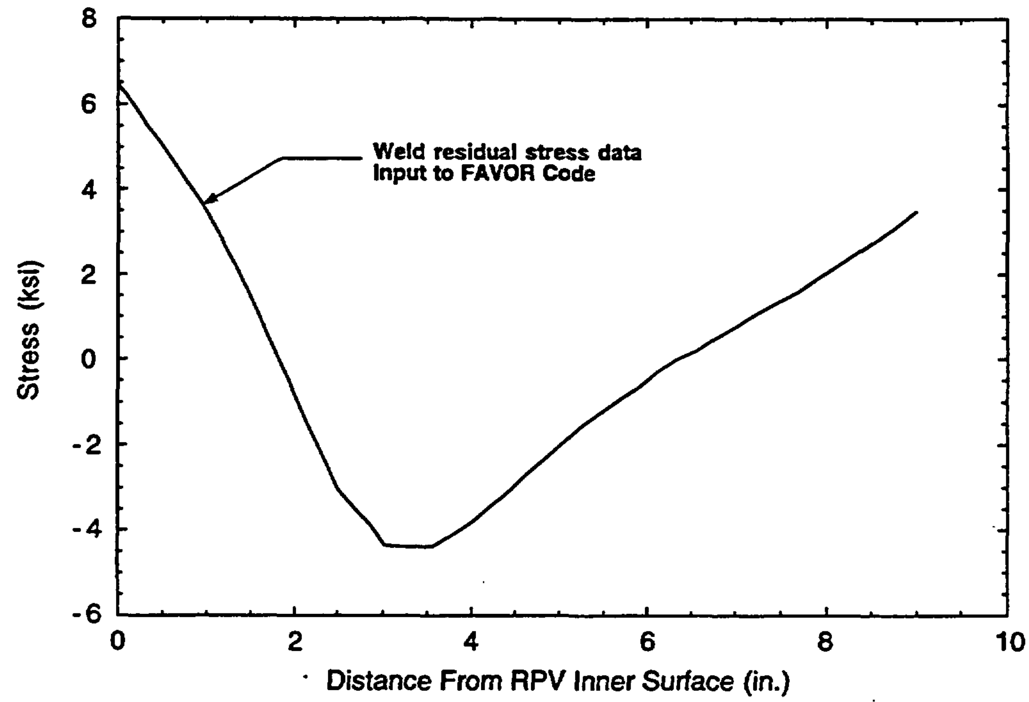

Fig. 10 Final weld residual stress through-thickness distribution developed for use in RPV integrity analyses.

3.3.2 Cladding residual stress

The procedure used for a determination of the residual stresses in the clad material was to separate the clad layer from the underlying plate material, measure the change in geometry during this procedure, and calculate the clad stresses relieved. As a test article, a block of the clad-over-plate shell material measuring 10" square and full-RPV-wall thickness was used. Two smaller full-thickness blocks were then saw-cut from this block. Each of these smaller blocks was machined to a parallelepiped 4" x 1" x 9 1/4" thick (shell thickness). One block was taken with the clad layer length (4" dimension) in the circumferential direction of the shell (parallel to the cladding deposition direction) and the other in the longitudinal direction (transverse to the cladding deposition direction). The sequence of operations and the orientation of the clad strips relative to each other are shown in Fig. 11. As part of the machining process, inspection points (fiducial marks) were applied at points on the original and machined-end surfaces of the clad layer. These points were drilled to a depth not exceeding 0.010" using a conical point drill. This procedure assured an inspection point with uniform dimensions for repeatability of the subsequent measurements.

Precision dimensional inspections were performed to measure the x, y, and z coordinates of each of these fiducial marks with the blocks in the parallelepiped geometry. A set of coordinate axes is shown in Fig. 11 for reference. After the initial dimensional measurements, the clad layer was machined away from the plate material until the clad/base metal fusion zone was completely revealed. To estimate the mid-thickness of the fusion zone, visual inspection was used to judge when approximate equal amounts of

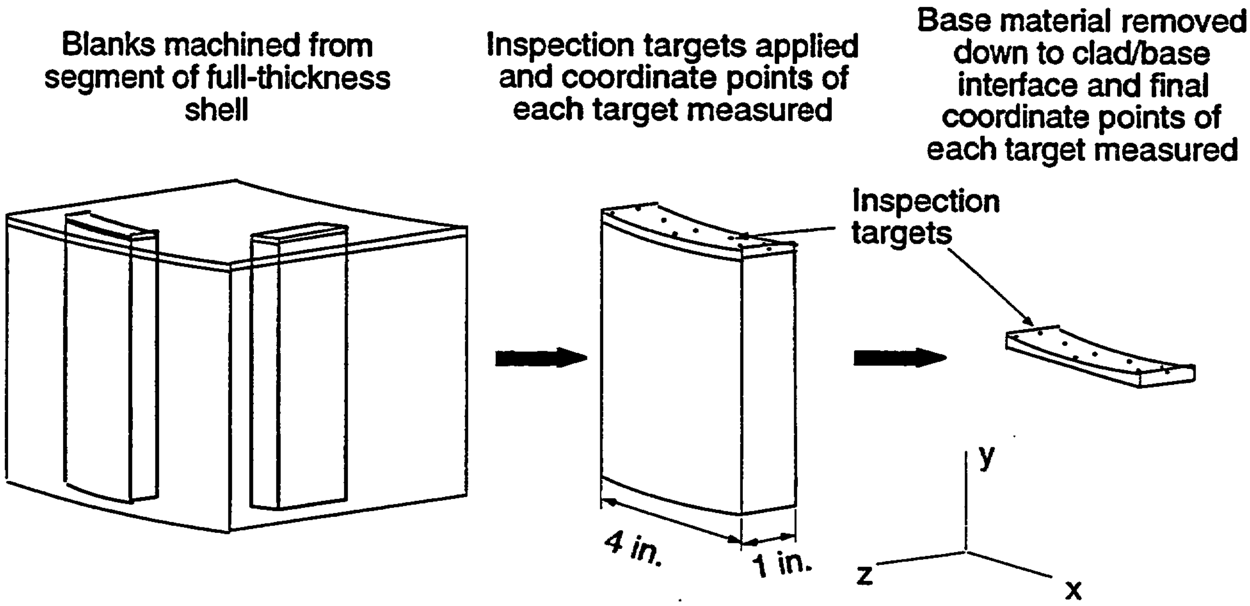

Fig. 11 Schematic of operations used to obtain strain measurements from relaxation of residual stresses in clad layer.

plate and clad materials were exposed on the machined surface. The clad layer parts were re-inspected to measure the distorted shape as compared to the original shape. The change in shape of the parts was then determined by subtracting the initial coordinates of each inspection point from the final (deformed) coordinates. Using the deformed coordinates for each fiducial mark, maximum and average values (for the sets of measurements) of circumferential and longitudinal strain were calculated and are shown in Table 2.

Table 2

Strains Measured when Cladding was Released from the Base Material of a RPV Shell Segment

	Strain ε_x (in/in)	Strain ε_y (in/in)
Maximum	-0.00086	-0.00041
Average	-0.00070	-0.00033

These strains were used as boundary conditions in finite-element analyses to determine the maximum and average values of clad residual stress components. It was first necessary, however, to determine the appropriate value of elastic modulus for the clad strip since, as was described above, the final clad strip contained both clad and fusion zone material. From tensile tests of both clad and fusion zone material at –30°F, the effective modulus of the clad strip was determined to be,

$$E_3 = 24.85 \times 10^6 \text{ psi.}$$

The modulus, E_3, is for the composite clad strip at -30°F, while the residual stress measurements were performed at room temperature. To calculate the elastic modulus at room temperature, a third-order polynomial in temperature was fitted to the ASME data for 18 Cr - 8 Ni steel; see Table TM-1, p. 664, ASME Boiler and Pressure Vessel Code, Section II, Part D, 1992. The equation was then normalized to the value of E_3 at –30°F yielding a temperature dependent modulus for the clad strip of,

$$E = a_0 + a_1 T + a_2 T^2 + a_3 T^3, \tag{7}$$

where T = temperature in °F,
and $a_0 = 24.71$
$a_1 = -0.005257$
$a_2 = -4.6257\times10^{-7}$
$a_3 = -3.5109\times10^{-11}$.

The value of modulus for the equivalent clad layer at room temperature is then,

$$E_{RT} = 24.30\times10^6 \text{ psi.}$$

Using this value of modulus and a Poisson's ratio of 0.3, the resulting residual stresses from the finite-element analyses for the two cases considered (maximum and average strains) are shown below.

Strain Combination	Circumferential Stress (ksi)	Longitudinal Stress (ksi)
Maximum	26.2	17.8
Average	21.3	14.4

It is appropriate to also consider the contribution of the "curling" deformation of the clad strip to the overall residual stress-state. Considering the strip as a thin, cantilever beam with applied displacement of the free end, the residual bending stress in the clad strip was calculated to be in the range of 1 ksi. The bending stresses in the clad layer were then considered to be an insignificant part of the of the total residual stress state and would have minimal impact on the determination of a stress-free temperature.

3.3.3 Stress-free temperature

The residual stresses determined for the clad layer were used to calculate the stress-free temperature for the vessel. Since the circumferential and longitudinal stresses will not necessarily go to zero simultaneously, only the circumferential (larger) stress component was used to determine the stress-free temperature. Also, since the margin assessments performed and described below were to represent "best estimate" results, the average value of circumferential stress was used. Using the FAVOR Code, the stress-free temperature was calculated to be,

$$T_{stress\text{-}free} = 468°F.$$

4. Clad Stability Evaluation

The evaluation of cladding integrity under P-T curve loading conditions was done to determine if the cladding above a subclad flaw would remain structurally intact throughout the operating life of a RPV. Pressures associated with potential breaching of the cladding are designated (P_c) in order to distinguish them from the best estimate allowable pressures (P_{BE}) for the RPV shell. The failure modes of primary concern relative to the evaluation of P_c for irradiated cladding above a subclad flaw are (a) tensile instability at room temperature, and (b) ductile tearing at the RPV operating temperature. The process of cladding over a subclad crack has been shown to produce sharp micro-cracks in the cladding above the flaw [15]. These micro-cracks could be further extended by ductile tearing of the irradiated cladding.

ORNL has performed an elastic-plastic finite element analysis of a subclad flaw in a RPV using the finite element analysis model shown in Figs. 12 (a) and (b). COD results from that analysis are shown plotted as a function of pressure loading in Fig. 12(c). The response of cladding above a flaw to tensile loading was described in Section 3. COD results obtained from the tests are shown in Figs 7 and 8. Fig. 8 shows that the onset of tensile instability occurred when the COD reached approximately 20 mils. This COD value is substantially higher than the value (2.2 mils) obtained in Fig. 12 (c) at the operating pressure (2.2 ksi). Tensile instability of the cladding above a subclad flaw would not, therefore, be expected to occur under a single application of either operating loads or hydrotest loads.

Since the cladding process can introduce a sharp crack into the cladding immediately above a subclad flaw [15], the potential for further propagation of that crack by ductile tearing must be evaluated. Ductile tearing initiation data (K_{Jc}) for 3-wire stainless steel cladding, irradiated to a fluence of 2.41 x 10^{19} n/cm^2, are shown as a function of temperature in Fig. 13 [16]. The geometry of the subclad flaw in the elastic-plastic finite-element analysis model of Figs. 12 (a) and (b) was extended .025" into the cladding to permit the calculation of K_J values applied to the cladding. Results from the analysis of K_{JSC} values for all strain-controlled loading are shown superimposed on the tearing toughness curve of Fig. 13. The temperature-dependent difference $K_{JPC} = K_{Jc} - K_{JSC}$ is the ductile tearing toughness available for accommodation of pressure-induced stress intensity factors. A curve of K_{JPC} is also included in Fig.13. The minimum value of K_{JPC} was obtained as 48 ksi$\sqrt{\text{in}}$. at a temperature of approximately 480°F. Analysis results show the stress intensity factor (C_P) produced by a 1 ksi pressure loading in the RPV to be 30.3 ksi$\sqrt{\text{in}}$ for the stainless-steel portion of the sub-clad flaw. The limiting pressure for prevention of the onset of ductile tearing in the cladding above the subclad flaw is, therefore, $K_{JPC}/C_P = 48/30.3 = 1.6$ ksi. Since the operating pressure for the RPV at a temperature of 550°F would be of the order of 2.2 ksi, tearing of cladding above the subclad flaw would be predicted.

The RPV manufacturing process would tend to produce subclad rather than surface-breaking flaws. Results from the above evaluation, however, indicate that the microcracks introduced by the cladding process and ductile tearing could result in breaching of the cladding, thereby converting the subclad flaw to a surface-breaking flaw. This finding prevents the use of a sub-clad flaw for P-T curve margin assessments. Limited evidence exists showing that cladding over a pre-existing sharp crack can produce a reduction in fracture toughness of the crack-tip material, by the action of locally intensified strain aging (LISA) embrittlement. The LISA embrittlement mechanism could be of concern if subclad cracks convert to surface cracks by the process described above. At this time, however, the body of data available on cladding-induced LISA embrittlement is not sufficient for an evaluation of its potential impact on P-T curve margins.

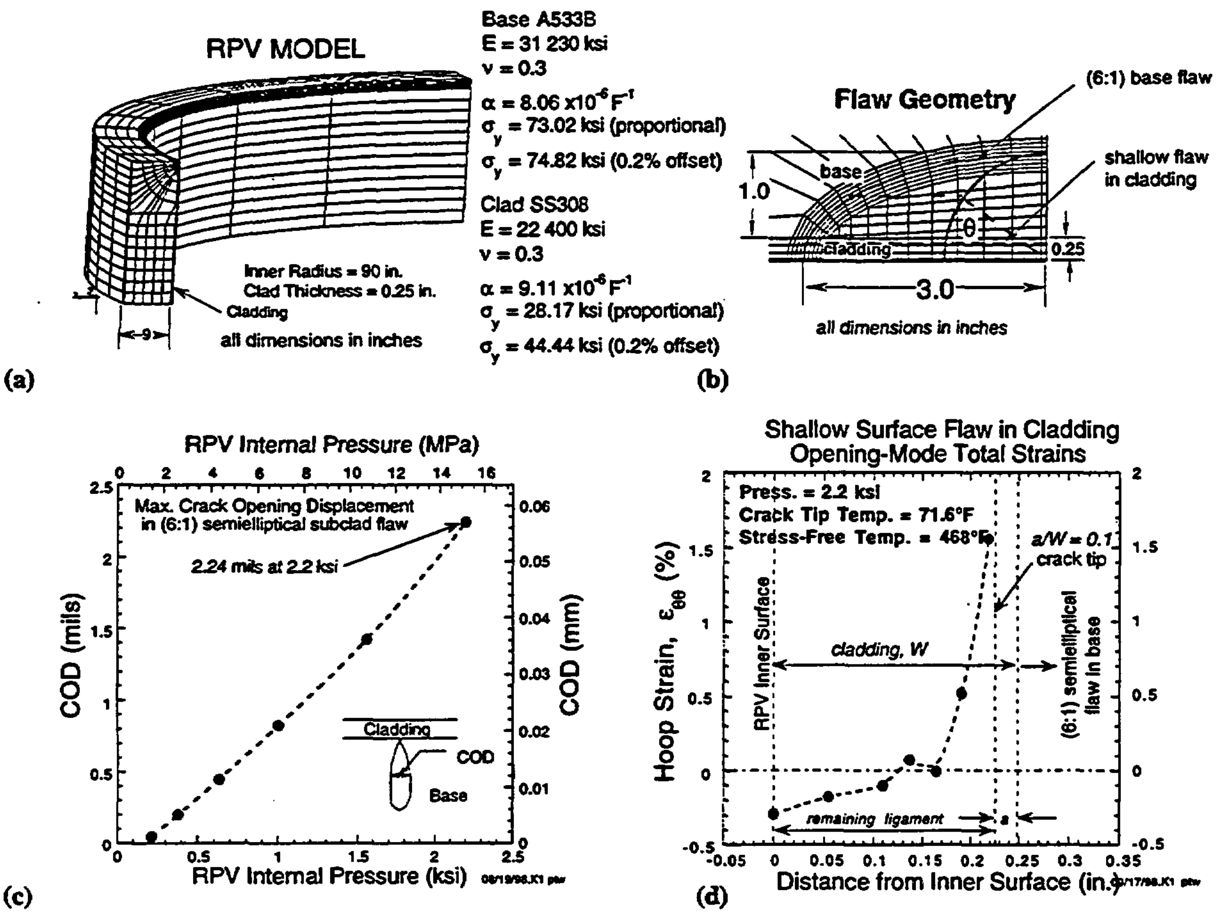

Fig. 12 Finite-element analysis of a semielliptical subclad flaw: (a) RPV model, (b) subclad flaw geometry, (c) crack-opening displacement, and (d) hoop strain profile across cladding with shallow flaw (a/W=0.1) opening to larger flaw in base.

5. Margin Assessments

A reference problem was defined under NRC guidance such that various organizations could perform deterministic fracture analyses to benchmark solutions. The objective for this study was to compare the fracture margin derived from a best-estimate-analysis that includes all of the loads, to the fracture margins derived from the current ASME code and the proposed change to the current code, both of which include only the load due to pressure and the through-wall thermal gradient. The description of the benchmark reference problem is as follows:

The RPV specified for the sample problem has an inside radius of 90," a wall thickness of 9," and a clad thickness of 0.25." Thermal-elastic properties are specified for the sample problem as given in Table 4. The temperature-time history of the cool-down transient is as shown in Fig. 14. The neutron fluence at the inner surface of the RPV was specified as 1.01 x 10^{19} n/cm^2; the copper and nickel weight percent concentrations were set at 0.30 and 0.86, respectively; and an initial unirradiated value of RT_{NDT} defined

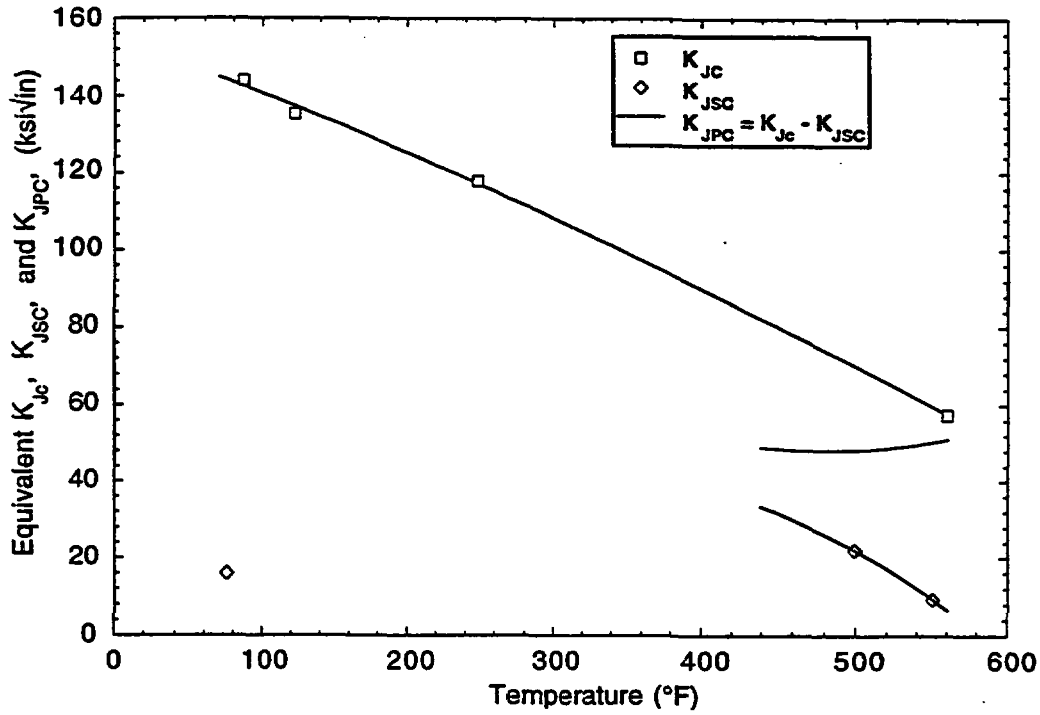

Fig. 13 Stress intensity factor available to resist tearing in the cladding.

Table 4
RPV thermal-elastic material properties

	clad	base
Thermal conductivity (BTU/hr-ft-°F)	10.0	24.0
Specific heat (BTU/lb-°F)	0.12	0.12
Modulus of elasticity (ksi)	22800	28000
Thermal expansion coefficient (°F^{-1})	9.45 x 10^{-6}	7.77x10^{-6}
Density (lb/ft^3)	489	489

was defined as 0°F. This combination of parameters produced a value of RT_{NDT} at the inner surface of the vessel of 236°F. The convective heat transfer coefficient at the inside surface of the RPV was set at 1000 BTU/ hr-ft^2-°F.

The postulated defect is an axially oriented semielliptical flaw with an aspect ratio (total length/depth) of 6:1 with a depth of 1" (t/9). The postulated flaw is assumed to be a through-clad inner-surface defect. Figure 15 illustrates the postulated defect. The elliptical parametric angle (θ) is measured around a semicircle the origin of which is at the center of the flaw on the inner surface of the vessel. The semicircle has a radius equal to the flaw depth. The angular crack front location is measured from the inner surface of the vessel (θ=0°), to the deepest point (θ=90°). For this postulated crack, the clad-base

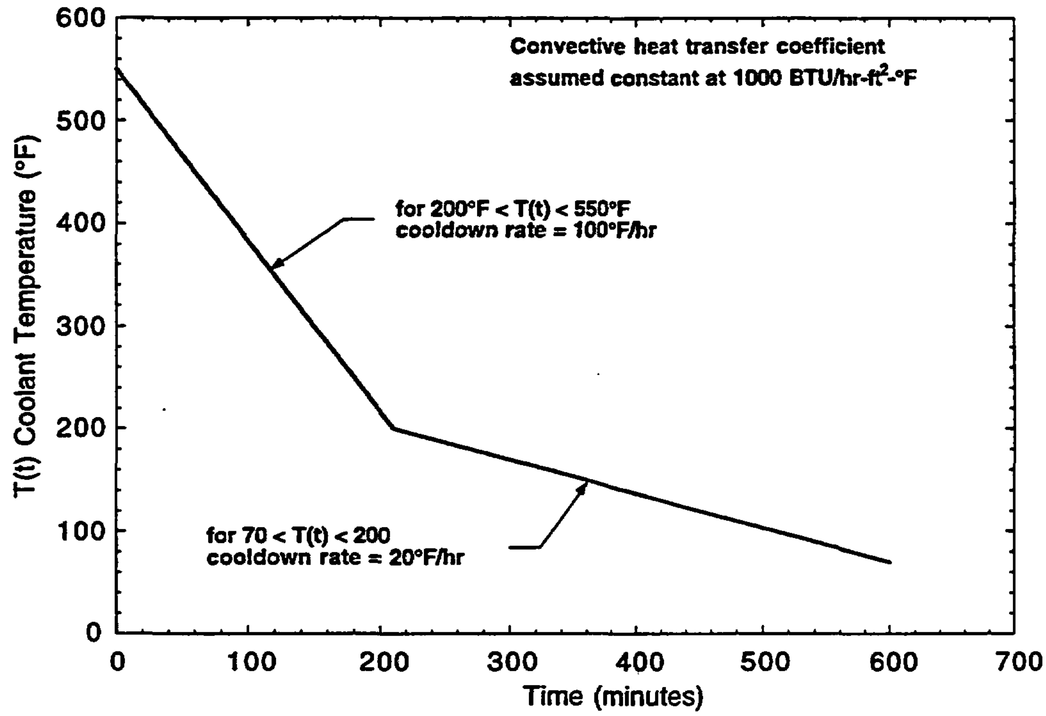

Fig. 14 Benchmark problem cool-down transient.

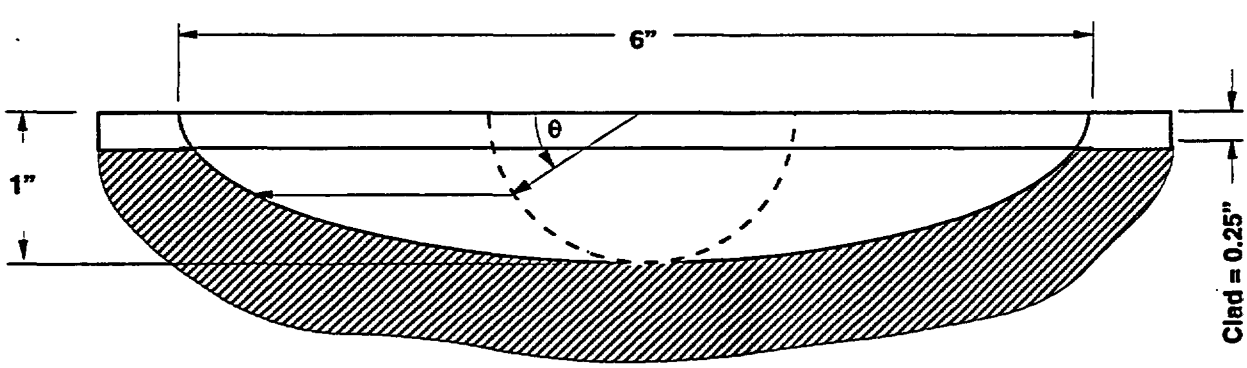

Fig. 15 Schematic showing relationship between circular and elliptic angles used to define points along crack front. The deepest point is at θ = 90°.

interface is at a parametric angular location of approximately 15° degrees. The best-estimate analysis includes searching for the point of initiation around the crack front from the clad-base interface to the deepest point of the flaw, since there are gradients in both the stress intensity factor and cleavage fracture initiation toughness around the crack front.

All deterministic fracture analysis results reported below were generated with the FAVOR computer code [17]. The FAVOR code uses the finite element method to perform thermal and stress analyses, utilizes stress intensity factor influence coefficients [18-20], and superposition to calculate K_I values. FAVOR has been validated to generate solutions that are within approximately 1-2% of those obtained by direct ABAQUS [21] 3-D finite element solutions [22]. ABAQUS is a nuclear quality assurance certified (NQA-1) general purpose multidimensional finite element code that has fracture mechanics capabilities.

Figure 16 shows the superposition of the time histories of stress intensity factor at the clad-base interface (at 15° degrees) due to weld residual stress, clad-base differential thermal expansion, and the through-wall thermal gradient produced by the benchmark transient in Fig. 14. The through-wall weld residual stress was derived in the HSST program from a combination of experimental measurements taken from a RPV shell segment made available from a cancelled pressurized-water reactor plant and finite element thermal and stress analyses [23]. A stress-free temperature of 468°F, derived using the room temperature clad stresses developed in Section 3.3.2, was used in the derivation of the load due to clad-base differential thermal expansion. Figure 16 also shows the time history of the total stress intensity factor at the deepest point (90°) of the flaw. After a time of 200 minutes, the total stress intensity factor is higher at the clad-base interface than at the deepest point of the flaw.

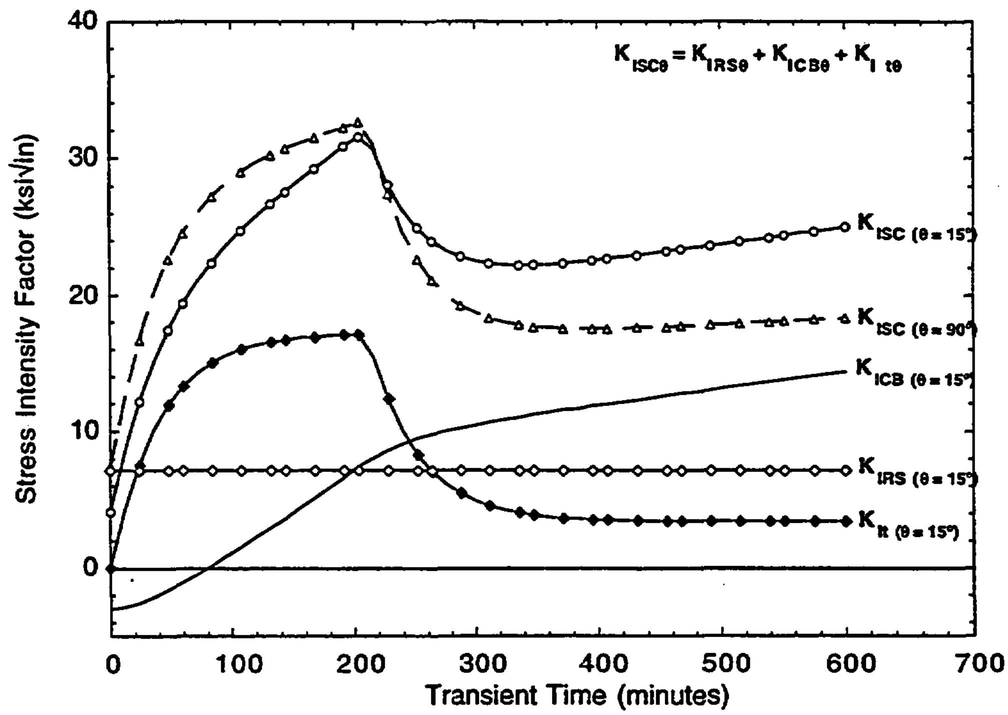

Fig. 16 Superposition of strain-controlled K_I at clad-base interface (θ = 15°) and total strain-controlled K_I at deepest point of flaw (θ = 90°).

Figure 17 illustrates P-T curves derived for the reference benchmark problem using the five models defined in Section 2.3 above and as specified in Table 5. Models 1 and 2 are the current Code methodology and the proposed modified Code methodology, respectively. In both of these cases, the flaw depth is 2.25" (t/4) and the K_I/K_{Ic} ratio is evaluated only at the deepest point of the flaw. The only loads included in models 1 and 2 are those produced by pressure and the through-wall thermal gradient. Models 1 and 2 include a safety factor of 2 on pressure loading. The minimum allowable pressure derived using model 1 is P_{CODE} = .43 ksi. The minimum allowable pressure derived using model 2 is $P_{NEWCODE}$ = 0.53 ksi.

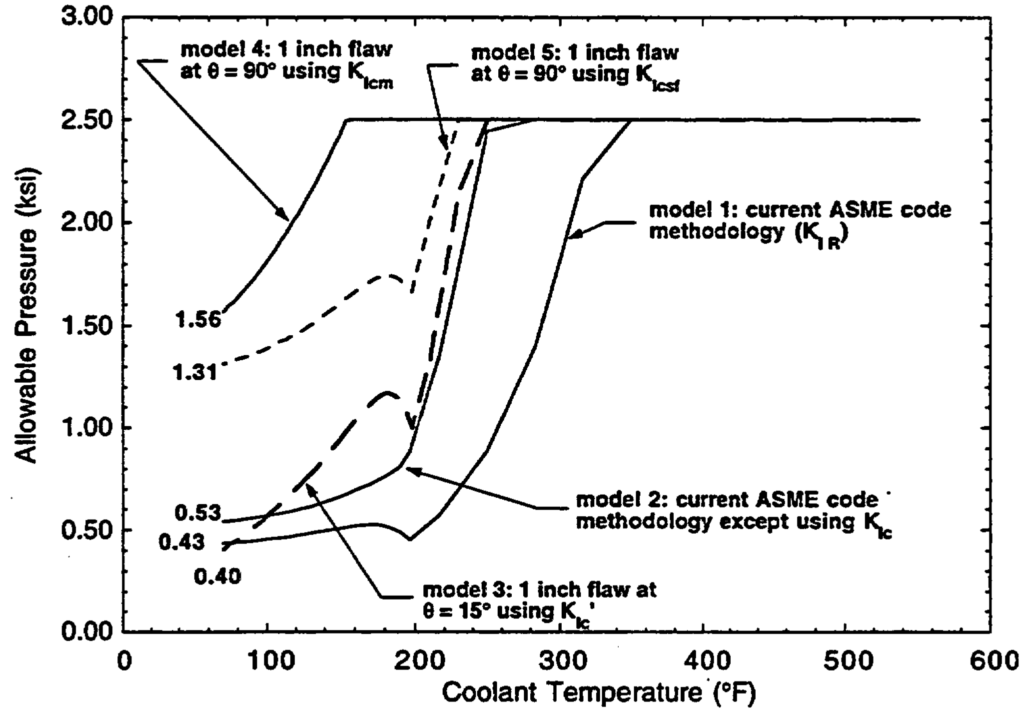

Fig. 17 Allowable pressure curves for 5 fracture models in Table 2 subjected to benchmark transient in Fig. 14.

Models 3-5 utilize the 1"deep flaw illustrated in fig. 15, the stress intensity factors illustrated in fig.16 and a safety factor of 1.0 on pressure loading. The only difference in models 3-5 is the representation of the fracture toughness. In each of these models, the entire crack front from the clad-base interface to the deepest point is considered.

Model 3, which applies the lower-bound curve to the EPRI K_{Ic} database (from which the ASME K_{Ic} curve was derived) in the region of interest [-200 °F ≤ (T-RT_{NDT}) ≤-150 °F], has a minimum best estimate (P_{BE}) pressure of 0.40 ksi.

Model 4, which applies the mean curve to the EPRI K_{Ic} database has a minimum best estimate (P_{BE}) pressure of 1.56 ksi. Results from this analysis, which utilizes a mean K_{Ic} curve in a deterministic analysis, are provided for comparative purposes only. Use of a mean curve was not, per se, within the stated objective of this paper which was to compare fracture margins from best-estimate analyses with those derived from the current and proposed Code methodology, both of which use lower-bound (K_{Ia} and K_{Ic}) curves.

Model 5, which applies the lower-bound curve to the shallow flaw K_{JC} database for RPV materials, (see figure 4), has a minimum best estimate (P_{BE}) pressure of 1.31 ksi. A summary of results from each of the five cases evaluated in this study is given in Table 5.

Figure 18 shows the allowable pressure as a function of crack front angular location for models 3-5 at a transient time of 600 minutes, which is the time that the coolant reaches the ambient temperature and is also the time at which the lowest allowable pressure occurs. For model 3, the lowest allowable pressure on the crack front occurs at the clad-base interface whereas for models 4 and 5, the lowest allowable pressure occurs at the deepest point on the crack front.

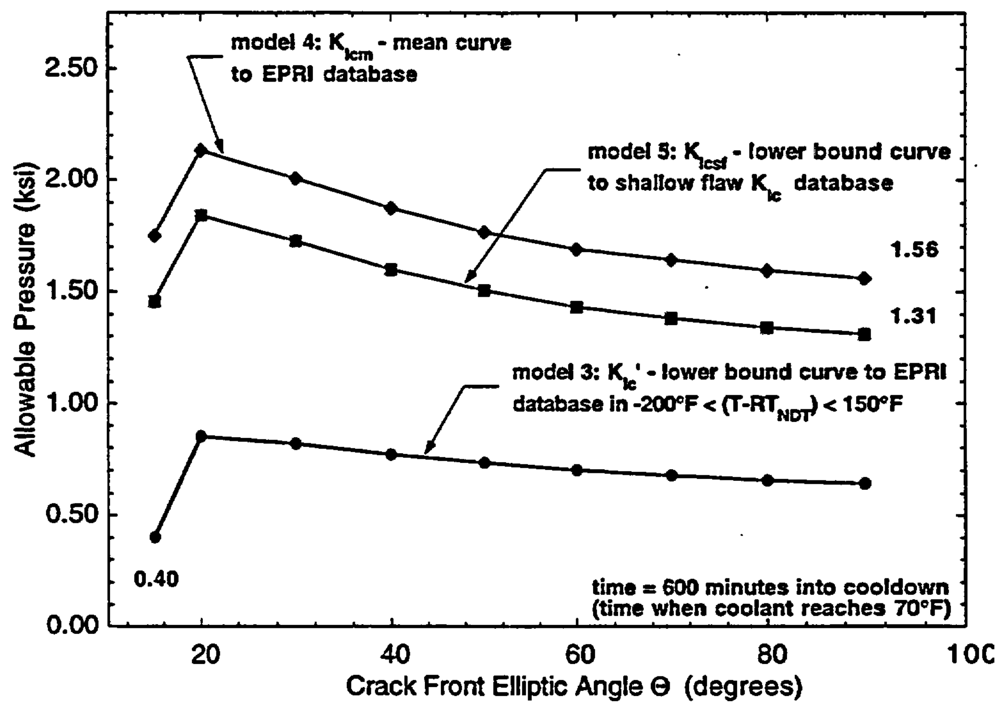

Fig. 18 Allowable pressure as a function of θ for reference benchmark problem at time = 600 minutes for three different representations of K_{Ic}.

As shown in fig. 17, the proposed change to the ASME code to replace the K_{Ia} curve with the K_{Ic} curve accomplishes the desired objective of opening-up the P-T operating envelope. The minimum allowable pressure for the benchmark problem is increased from the P_{CODE} value of 0.43 ksi to a $P_{NEWCODE}$ value of 0.53 ksi at the lowest coolant temperature. The increase in allowable pressure is larger at higher coolant temperatures. The P_{BE} result from Case 5 of this study shows that adequate margins against brittle fracture of the RPV will be maintained if K_{Ia} is replaced by K_{Ic} in the ASME code P-T curve analysis procedure.

Table 5
Minimum Best Estimate Allowable Pressures (P_{BE}) for the Five Sample Problems

Case Number And Description	Flaw Geometry	Fracture Toughness	Loading	Safety Factor on Pressure	Allowable Pressure P_{BE} ksi
1. ASME Sect. XI, Appendix G	1/4T 6:1 surface, deepest point	K_{IR}	P+T	2	0.43
2. As (1) but with K_{Ic} replacing K_{Ia}	1/4T 6:1 surface, deepest point	ASME K_{Ic}	P+T	2	0.53
3. All loading plus SF=1 plus lower bound K_{Ic}'	a=1" 6:1 surface, deepest point & near clad	Lower-bound to the EPRI K_{Ic} data (K_{Ic}')	P+T+R+C	1	0.40
4. As (3) but with K_{Icm} replacing K_{Ic}'.	a=1" 6:1 surface, deepest point & near clad	Mean curve through the EPRI K_{Ic} data (K_{Icm})	P+T+R+C	1	1.56
5. As (3) but with shallow-flaw fracture toughness K_{Jcsf} replacing K_{Ic}'	a=1" 6:1 surface, deepest point & near clad	Lower-bound to the ORNL/David Taylor K_{Jcsf} data	P+T+R+C	1	1.31

P = Pressure, T = Thermal gradient, R = Residual stress in the structural weld, and C = clad-base material differential thermal expansion

6. Interim Conclusions

- Justification for changing the fracture toughness used in the ASME P-T curve analysis procedure from K_{Ia} to K_{Ic} requires a demonstration that $P_{BE} \geq P_{NEWCODE}$.
- $P_{BE} \geq P_{NEWCODE}$ has been demonstrated using the lower-bound to the shallow-flaw uniaxial- and biaxial-loading fracture toughness database for RPV materials.
- It is important to recognize that lower-bound to the shallow-flaw fracture toughness database is controlled by results from clad cruciform biaxial-loading tests conducted at normalized temperatures ($T-RT_{NDT}$) not less than -40°F. A potential exists for the estimated shallow-flaw

lower-bound fracture toughness to be further adjusted if data from clad cruciform biaxial-loading tests become available for the normalized temperature range -200°F $\leq$ T-RT_{NDT} $\leq$ 170°F.

- A preliminary evaluation of the stability of a sub-clad version of the reference flaw indicates that the irradiated cladding over the flaw may be breached by the ductile tearing, thereby converting the subclad flaw to a surface flaw. The P-T curve margin assessment should, therefore, be based on a surface-breaking flaw.

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TECHNICAL BASIS FOR REVISED P-T LIMIT CURVE METHODOLOGY

Warren Bamford and Bruce Bishop
Westinghouse Electric Company

Abstract

The startup and shutdown process for an operating nuclear plant is controlled by pressure-temperature limit curves, which are developed based on fracture mechanics analysis. These limits are developed in Appendix G of Section XI, and incorporate nine numbers of safety margins; one of which is a lower bound fracture toughness curve.

There are two lower bound fracture toughness curves available in Section XI, K_{IA}, which is a lower bound on all static, dynamic and arrest fracture toughness, and K_{IC}, which is a lower bound on static fracture toughness only. The only change involved in this action is to change the fracture toughness curve used for development of P-T limit curves from K_{IA} to K_{IC}. The other margins involved with the process remain unchanged.

The primary reason for making this change is to reduce the excess conservatism in the current Appendix G approach that could, in fact, reduce overall plant safety. By opening up the operating window relative to the pump seal requirements, the chances of damaging the seals and initiating a small LOCA, a potential pressurized thermal shock (PTS) initiator, are reduced. Moreover, excessive shielding to provide an acceptable operating window with the current requirements can result in higher fuel peaking and less margin to fuel damage during an accident condition.

Technology developed over the last 25 years has provided a strong basis for revising the ASME Section XI pressure-temperature limit curve methodology. The safety margin which exists with the revised methodology is very large, whether considered deterministically or from the standpoint of risk.

Changing the methodology will result in an increase in the safety of operating plants, as the likelihood of pump seal failures and/or fuel problems will decrease.

Introduction

The startup and shutdown process, as well as press testing, for an operating nuclear plant is controlled by pressure-temperature limit curves, which are developed based on fracture mechanics analysis. These limits are developed in Appendix G of Section XI, and incorporate four specific safety margins:

1. Large flaw, ¼ thickness
2. Safety factor = 2 on pressure stress for startup and shutdown
3. Lower bound fracture toughness
4. Upper bound adjusted reference temperature (RT_{NDT})

There are two lower bound fracture toughness curves available in Section XI, K_{IA}, which is a lower bound on all static, dynamic and arrest fracture toughness, and K_{IC}, which is a lower bound on static fracture toughness only. The only change involved in this action is to change the fracture toughness curve used for development of P-T limit curves from K_{IA} to K_{IC}. The other margins involved with the process remain unchanged. There are a number of reasons why the limiting toughness in the Appendix G pressure-temperature limits should be changed from K_{IA} to K_{IC}.

Use of K_{IC} is More Technically Correct

The heat-up and cool-down process is a very slow one, with the fastest rate allowed being 100° per hour. The rate of change of pressure and temperature is often constant, so the stress is essentially constant in this case. Both the heat-up and cool-down and the pressure testing are essentially static processes. In fact, all operating transients (levels A, B, C and D) correspond to static loadings, with regard to fracture toughness.

The only time when dynamic loading can occur and where the dynamic/arrest toughness K_{IA} should be used for the reactor pressure vessel is when a crack is running. This might happen during a PTS transient event, but not during heatup or cooldown. Therefore, use of the static toughness K_{IC} lower bound toughness would be more technically correct for development of P-T limit curves.

Use of Historically Large Margin No Longer Necessary

In 1974, when the Appendix G methodology was first codified, the use of K_{IA} (K_{IR} in the terminology of the time) to provide additional margin was thought to be necessary to cover uncertainties and a number of postulated but unquantified effects. Almost 25 years later, significantly more is known about these uncertainties and effects.

Flaw Size

With regard to flaw indications in reactor vessels, there have been no indications found at the inside surface of any operating reactor in the core region which exceed the acceptance standards of Section XI, in the entire 28 year history of Section XI. This is a particularly impressive conclusion when considering that core region inspections have been required to concentrate on the inner surface and near inner surface region since the implementation of Regulatory Guide 1.150. Flaws have been found, but all have been qualified as buried, or embedded.

There are a number of reasons why no surface flaws exist, and these are related to the fabrication and inspection practices for vessels. For the base metal and full penetration welds, a full volumetric examination and surface exam is required before cladding is applied, and these exams are repeated after cladding.

Further confirmation of the lack of any surface indications has recently been obtained by the destructive examination of portions of several commercial reactor vessels, for example the Midland vessel and the PVRUF vessel.

Fracture Toughness

Since the original formulation of the K_{IA} and K_{IC} curves, in 1972, the fracture toughness database has increased by more than an order of magnitude, and both K_{IA} and K_{IC} remain lower bound curves, as shown for example in Figure 1 for K_{IC}[1] compared to Figure 2, which is the original database[2].

In can be seen from Figure 1 that there are a few data points which fall just below the curve. Consideration of these points, as well as the (over 1500) points above the curve, leads to the conclusion that the K_{IC} curve is a lower bound for a large percentage of the data.

Local Brittle Zones

A third argument for the use of K_{IA} in the original version of Appendix G was based upon the concern that there could be a small, local brittle zone in the weld or heat-affected-zone of the base material that could pop-in and produce a dynamically moving cleavage crack. Therefore, the toughness property used to assess the moving crack should be related to dynamic or crack arrest conditions, especially for a ferritic pressure vessel steel showing distinct temperature and loading-rate (strain-rate) dependence. The dynamic crack should arrest at a ¼-T size, and any re-initiation should consider the effects of a minimum toughness associated with dynamic loading. This argument provided a rationale for assuming a ¼-T postulated flaw size and a lower bound fracture toughness curve considering dynamic and crack arrest loading. The K_{IR} curve in Appendix G of Section III, and the equivalent K_{IA} curve in Appendix A and Appendix G of Section XI provide this lower bound curve for high-rate loading (above any realistic rates in reactor pressure vessels during any accident condition) and crack arrest conditions. This argument, of course, relies upon the existence of a local brittle zone.

After over 30 years of research on reactor pressure vessel steels fabricated under tight controls, micro-cleavage pop-in has not been found to be significant. This means that researchers have not produced catastrophic failure of a vessel, component, or even a fracture toughness test specimen in the transition temperature regime. The quality of quenched, tempered, and stress-relieved nuclear reactor pressure vessel steels, that typically have a lower bainitic microstructure, is such that there may not be any local brittle zones that can be identified. Testing of some test specimens at ORNL has shown some evidence of early pop-ins for some simulated production weld metals, but the level of fracture toughness for these possible early initiations is within the data scatter for other ASTM-defined fracture toughness values (K_{IC} and/or K_{JC}). Therefore, it is time to remove the conservatism associated with this postulated condition and use the ASME Code lower bound K_{IC} curve directly to assess fracture initiation. This is especially true when the unneeded margin may in fact reduce overall plant safety.

Overall Plant Safety is Improved

The primary reason for making this change is to reduce the excess conservatism in the current Appendix G approach that could in fact reduce overall plant safety. Considering the impact of the change on other systems (such as pumps) and also on personnel exposure, a strong argument can be made that the proposed change will increase plant safety and reduce personnel exposure for both PWRs and BWRs.

Impact on PWRs:

By opening up the operating window relative to the pump seal requirements, as shown schematically in Figure 3, the chances of damaging the seals and initiating a small LOCA, a potential pressurized thermal shock (PTS) initiator, are reduced. Moreover, excessive shielding to provide an acceptable operating window with the current requirements can result in higher fuel peaking and less margin to fuel damage during an accident condition.

The proposed change also reduces the need for lock-out of the HPSI systems, which improves personnel and plant safety and reduces the potential for a radioactive release. Finally, challenges to the plant LTOP system and potential problems with reseating the valves would also be reduced.

Impact on BWRs:

The primary impact on the BWR will be a reduction in the pressure test temperature. BWRs use pump heat to reach the required pressure test temperatures. Several BWR plants are required to perform the pressure test at temperatures over 212°F under the current Appendix G criteria. The high test temperature poses several concerns: (i) pump cavitation and seal degradation, (ii) primary containment isolation is required and ECCS/safety systems have to be operational at temperatures in excess of 212°F, (iii) leak detection is difficult and more dangerous since the resulting leakage is steam and poses safety hazards of burns and exposure to personnel. The reduced test temperature eliminates these safety issues without reducing overall fracture margin.

Reactor Vessel Fracture Likelihood is Very Low

It has long been known that the P-T limit curve methodology is very conservative[3,4]. Changing the reference toughness to K_{IC} will maintain a very high margin, as illustrated in Figure 4, for a pressurized water reactor. This figure shows a series of P-T curves developed for the same plant, but with different assumptions concerning flaw size, safety margin and fracture toughness.

The results shown in Figure 4 were obtained for a sample problem which was solved by several members of the Section XI working group on Operating Plant Criteria, for both PWR and BWR plants. The sample problem requires development of an operating P-T cooldown curve or the pressure test for an irradiated vessel. Two P-T curves were required, one using K_{IA} and the second using K_{IC}. In both cases the quarter thickness flaw was used, along with the appropriate safety factor on pressure.

To determine the margins (pressure ratios) that are included in these curves, a reference P-T curve was developed, using a best estimate (mean) K_{IC} curve, and no safety factor on stress, along with a flaw depth of one inch. Typical results are shown in Table 1. Comparing the reference or best estimate curve with the two P-T curves calculated using code requirements, we see that there is a large margin on the allowable pressure, whether one uses K_{IA} or K_{IC} limits in Appendix G.

For PWRs, another important contribution to the margin, which cannot be quantified, is the low temperature overpressure protection system (LTOP) which is operational in the low temperature range. The margins increase significantly for higher temperatures, as seen in Figure 4. .

Impact of the Change on P-T Curves

To show the effect that the proposed change would produce, a series of P-T limit curves were produced for a typical plant. These curves were produced using identical input information, with one curve using K_{IA} and the other using the proposed new approach, with K_{IC}. Since the limiting conditions for the PWR (cool-down) and the BWR (pressure test) are different, separate evaluations were performed for PWRs and BWRs.

The results are shown in Figure 5 for a typical PWR cool-down transient.

Summary and Conclusions

Technology developed over the last 25 years has provided a strong basis for revising the ASME Section XI pressure-temperature limit curve methodology. The safety margin that exists with the revised methodology is still very large.

From the standpoint of risk, changing the methodology will result in an increase in the safety of operating plants, as the likelihood of pump seal failures, need for HPSI systems lock-out, LTOP system challenges and/or fuel margin problems, and personnel hazards and exposure will all decrease.

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1. VanderSluys, W.A. and Yoon, K.K., "Transition Temperature Range Fracture Toughness in Ferritic Steels and Reference Temperature of ASTM", prepared for PVRC and BWOG, BAW 2318, Framatome Technologies, April 1998.

2. Marston, T.U., "Flaw Evaluation Procedures, Background and Application of ASME Section XI, Appendix A", EPRI Special Report NP-719-SR, August 1978.

3. Chirigos, J.N. and Meyer, T.A., "Influence of Material Property Variations on the Assessment of Structural Integrity of Nuclear Components", ASTM Journal of Testing and Evaluation, Vol. 6, No. 5, Sept. 1978, pp 289-295.

4. White Paper on Reactor Vessel Integrity Requirements for Level A and B conditions, prepared by Section XI Task Group on R.V. Integrity Requirements, EPRI TR-100251, January 1993.

Table 1

Summary of Allowable Pressures for

20 Degree/hour Cooldown of Axial Flaw

at 70 Degrees F and RT_{PTS} of 270 F

(Typical PWR Plant)

Type of Evaluation	Allowable Pressure* (psi)	Pressure Ratio
Appendix G with t/4 flaw and Ki_a Limit	420	1.00
Appendix G with t/4 flaw and Ki_c Limit	530	1.26
Reference 1 inch flaw for pressure, thermal, residual and cladding loads	1520	3.61
Reference 1 inch flaw for pressure, thermal and residual loads	1845	4.38
Reference 1 inch flaw for pressure and thermal loading only	2305	5.48

* Note: Comparable values of allowable pressure were calculated by the ASME Section XI Operating Plant Working Group Members from Westinghouse, Framatome Technologies and Oak Ridge National Laboratory

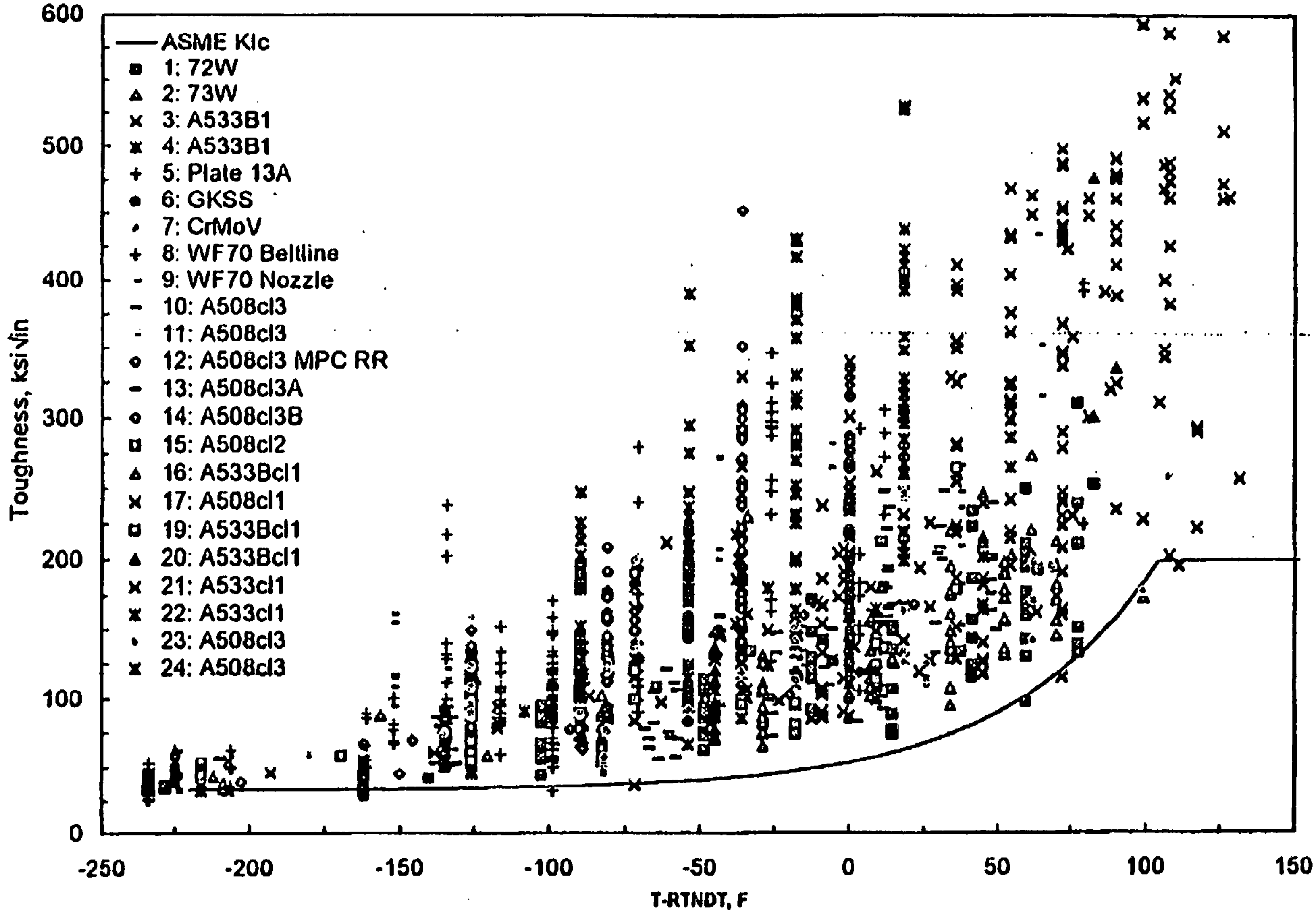

Figure1. Static Fracture Toughness Data (K_{Ic}) Now Available, Compared to K_{Ic} [1]

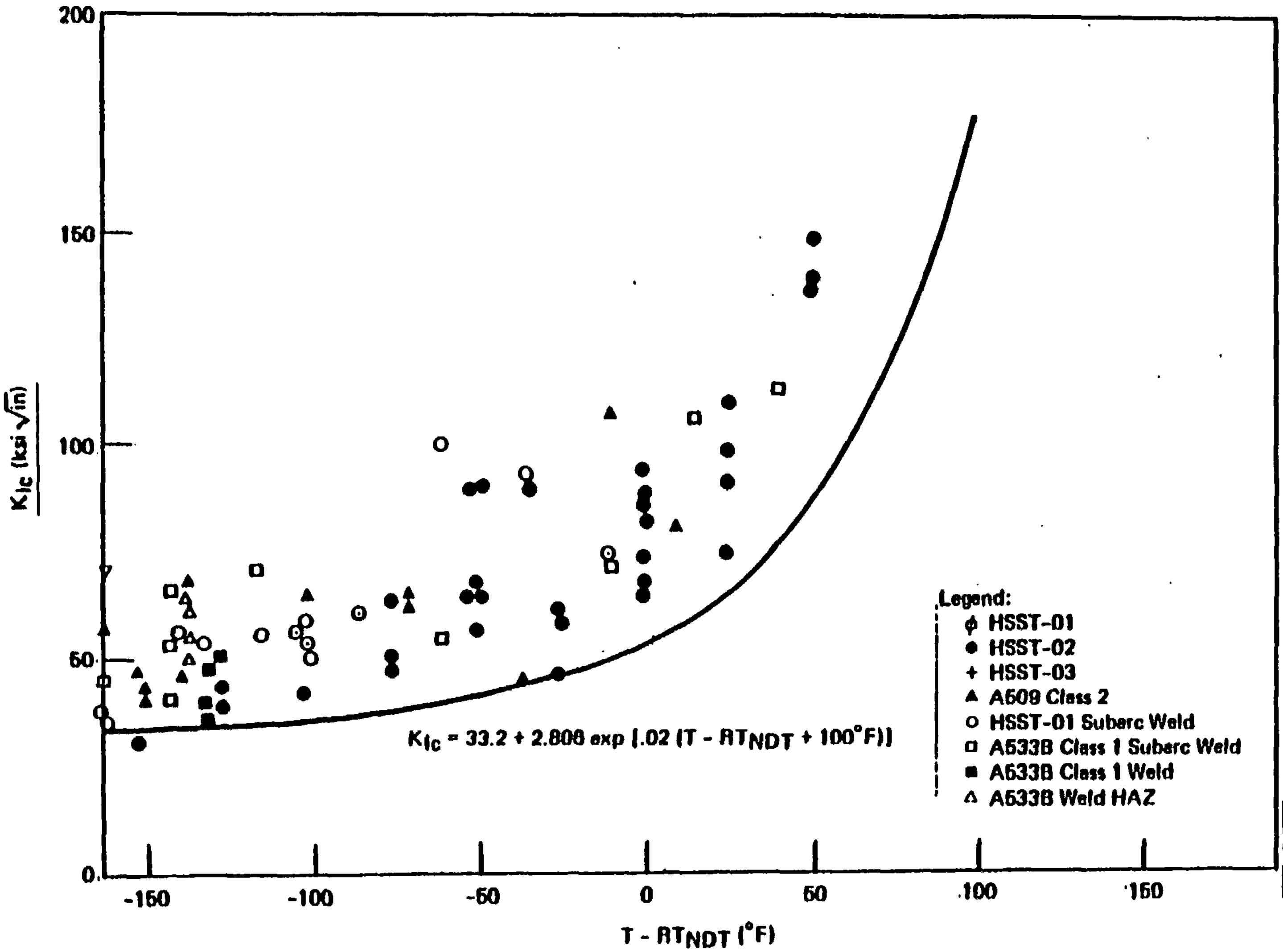

Figure 2. Original K_{IC} Reference Toughness Curve, with Supporting Data [2]

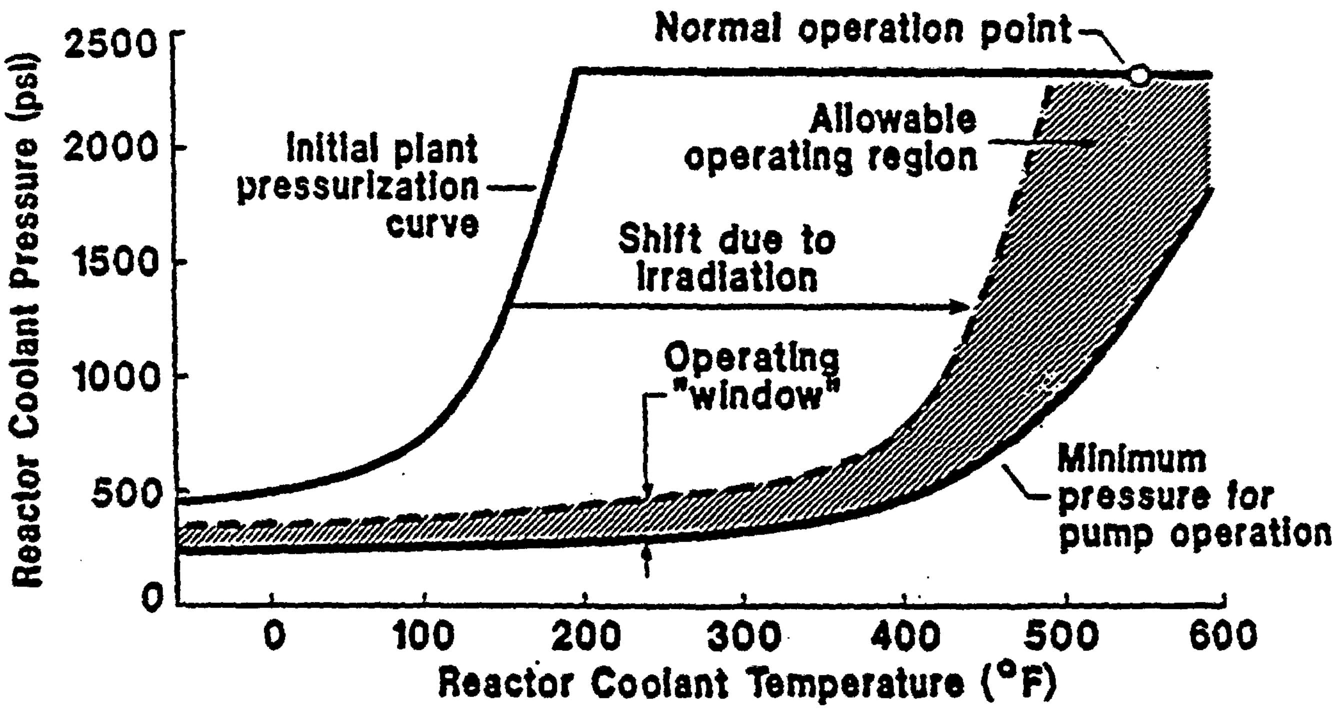

Figure 3. Operating Window From P-T Limit Curves [4]

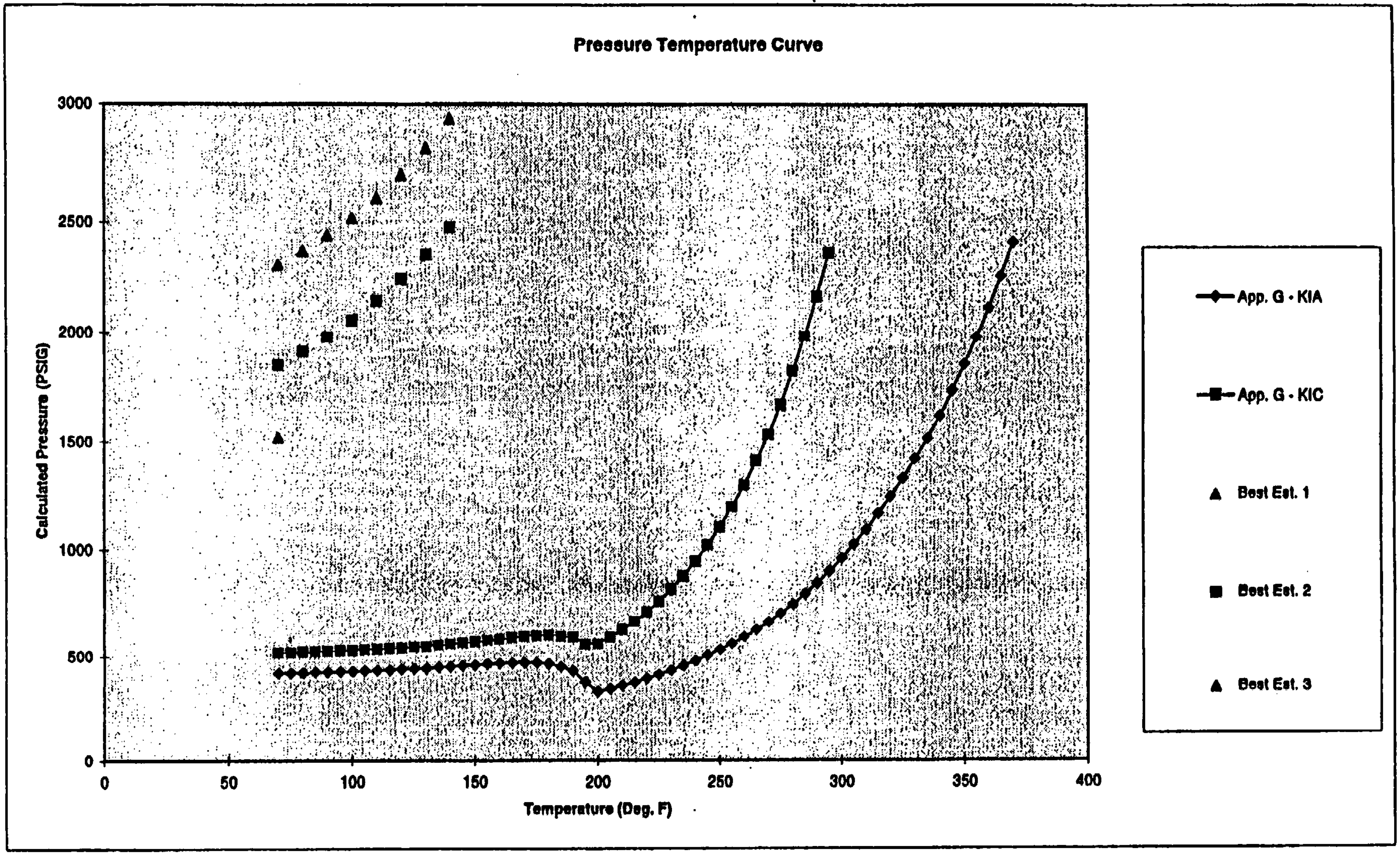

Figure 4. P-T Limit Curves Illustrating Deterministic Safety Factors

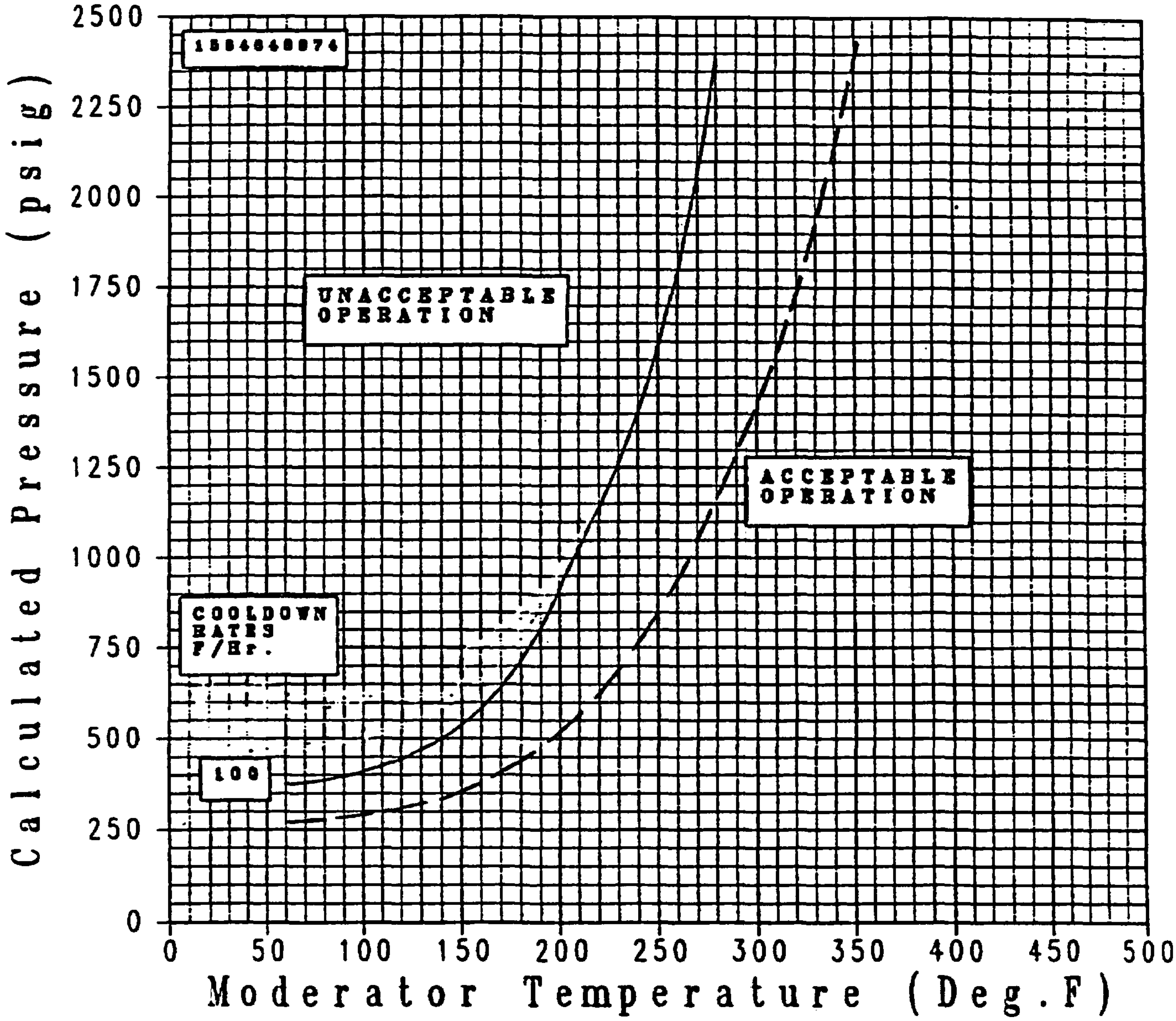

Figure 5. Comparison of Cool-Down Curves for the Existing and Proposed Methods
[Dashed Curve = Existing (K_{IA}) and Solid Curve = Proposed (K_{IC})]

Fabrication Flaws in Reactor Pressure Vessels[1]

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Abstract

Pacific Northwest National Laboratory (PNNL) under contract to NRC has performed nondestructive and destructive examinations of welds taken from reactor pressure vessels of cancelled nuclear power plants. One such vessel is the Pressure Vessel Research User Facility (PVRUF) vessel, which was located at the Oak Ridge National Laboratory (ORNL). The objective has been to determine the numbers, locations and sizes of flaws in the vessel welds, and to develop empirical estimates of fabrication flaw rates for use in fracture mechanics structural assessments. Simulations have also been performed with the RR-PRODIGAL computer code to predict the number and sizes of flaws in the welds with the objective to provide a basis for generalizing the observed flaw rates to other U.S. RPVs, and for estimating rates for flaws having depths significantly greater than those observed. The very sensitive SAFT-UT (Synthetic Aperture Focusing Technique for Ultrasonic Examination) technique was used to detect and characterize flaws. The total number of detected flaws was about 2500 with most of these flaws having through-wall dimensions of less than 3-mm. Larger flaws were confirmed and characterized on the basis of the detailed SAFT-UT, radiographic examinations, and destructive examinations. The observed flaw distribution and those predicted by RR-PRODIGAL were found to be in relatively good agreement for flaw depths greater than 5-mm. The data were also compared with estimates from the Marshall flaw distribution. An assumed flaw density of 400 flaws per cubic meter was found to provide a reasonable approximation to the PVRUF data for flaw depths in the range of 5-20 mm. Future work at PNNL will measure flaw occurrence rates for welds removed from other unused reactor vessels with a focus on BWR vessels (River Bend, Shoreham, and Hope Creek).

Introduction

The estimated number and sizes of flaws in reactor pressure vessel (RPV) welds are important inputs to probabilistic fracture mechanics calculations for predicting failure probabilities for reactor pressure vessels. But unfortunately, they are also inputs which are believed to have the greatest levels of uncertainty. To reduce this level of uncertainty, NRC has supported research to establish a better basis

[1] Work support by the U.S. Nuclear Regulatory Commission under contract DE-AC06-76RLO 1830; NRC JCN W6275, Ms. Deborah Jackson Program Monitor; NRC JCN L2606, Dr. Shah Malik Program Monitor

for estimating the distributions of flaws in RPV welds. Estimates of flaw rates from the research are intended to serve as inputs to fracture mechanics assessments, such as those relating to Pressurized Thermal Shock (PTS) events.

Probabilistic fracture mechanics computer codes such as FAVOR (Dickson 1994) and VISA-II (Simonen, et al. 1986) require accurate estimates of the flaw rates to determine the likelihood of vessel rupture during a PTS event. Fracture mechanics calculations during the 1980s at Oak Ridge National Laboratory (ORNL) estimated vessel failure probabilities for PTS events (Selby et al. 1985). These calculations (in support of Regulatory Guide 1.154) concluded that the inputs for flaw densities and size distributions were the largest source of uncertainty in failure probability calculations. The ORNL inputs were based on results of the Marshall Committee Study (1984), and involved a number of approximations and conservative assumptions such as arbitrarily placing all flaws at the inner surface of the vessel.

Given the difficulty of improving on the well-known and extensively used Marshall distribution, little research progress was made until the early 1990s. The literature shows the development of two complimentary approaches. One approach involves the statistical application of data from nondestructive inservice examinations of welds. Lance et al. (1992) and Rosinski et al. (1997) described the use of data from inservice inspections (ISI) along with statistically based software (the SAVER code) to develop flaw size and density distributions. Another approach developed by Rolls Royce and Associates in the United Kingdom simulates the population of flaws in multi-pass welds by application of an expert system model based on input from experts in the areas of welding and vessel fabrication (Chapman 1993; Chapman, Khaleel and Simonen 1996).

Both approaches have the objective of using the best available data and knowledge to estimate fabrication flaw occurrence rates. Other objectives have been to develop a basis for extrapolating flaw occurrence rates to other vessels which have not been subject to detailed examinations, and to estimate the occurrence rates for flaws that are much larger than the flaws which are observed within the limited volumes of examined vessel material.

The present paper describes work performed by PNNL on two research projects. One project has involved nondestructive and destructive examinations of welds taken from reactor pressure vessels, which were manufactured for cancelled nuclear power plants (Schuster, Doctor and Simonen 1996; Schuster, Doctor and Pardini 1997). One such vessel is the PVRUF vessel. The objective has been to determine the number, locations, and sizes of flaws in the vessel welds, and to develop empirical estimates of fabrication flaw rates for use in fracture mechanics structural assessments.

A second PNNL project has involved collaboration with Rolls Royce and Associates to develop an expert system model (RR-PRODIGAL) for predicting the number and sizes of flaws in the welds of reactor pressure vessels as manufactured in the United States (Chapman and Simonen 1998). Simulations have been performed with the RR-PRODIGAL code to predict the types, number, and sizes of flaws in the welds of the PVRUF vessel. The objectives of these calculations have been 1) to provide a basis for generalizing the observed flaw rates and for estimating rates for flaws having depths significantly greater than those observed, and 2) to compare the observed flaw distributions with those predicted by the computer modeling code in order to validate the code, which can then be used as an engineering tool to extrapolate the observed flaw data to the entire population of U.S. reactor vessels.

This paper describes the PVRUF vessel and the methods used for the nondestructive and destructive examinations of the vessel. Data describing the numbers and sizes of the flaws that were detected and sized for the various regions of the vessel are documented. The paper then describes the modeling of the PVRUF welds by the RR-PRODIGAL computer code, and compares the predicted flaw distributions with the measured data from the vessel. Conclusions are then presented along with a discussion of some implications of the data in terms of fracture mechanics evaluations of reactor pressure vessel integrity.

Description of PVRUF Vessel

The PVRUF vessel (Figure 1) was assembled by Combustion Engineering in the late 1970s and early '80s for a nuclear power plant that was not completed. The vessel has a diameter of 4.39-meter (173-inch), a height of approximately 13.34-meter (525-inch), and is made out of A533B steel. The wall thickness varies from one region to the next, but within 25-cm (10-inch) of the beltline welds it is 22-cm (8.6-inch) thick. Subsequently, the vessel was moved to the Oak Ridge National Laboratory to be used for research purposes. One such study of the PVRUF vessel is described in the present paper.

The axial and circumferential welds were of a single-V configuration as indicated by Figure 2. Welding was by the submerged arc process with the root passes at the inner surface removed by back gouging and then rewelded with a process that produced an inner region characterized by smaller weld beads. Stainless steel cladding was strip welded circumferentially to the ring subassemblies that made up the vessel beltline using two staggered and overlapping layers of clad. The region over the circumferential welds was clad after these welds were completed using a single welding wire process.

There was evidence of weld repairs both from PNNL examinations and from documentation prepared by the vessel fabricator and made available to PNNL by ORNL. Some of these weld repairs involved the removal of substantial regions of the main seam welds to significant depths and were followed by rewelding by a manual metal arc process.

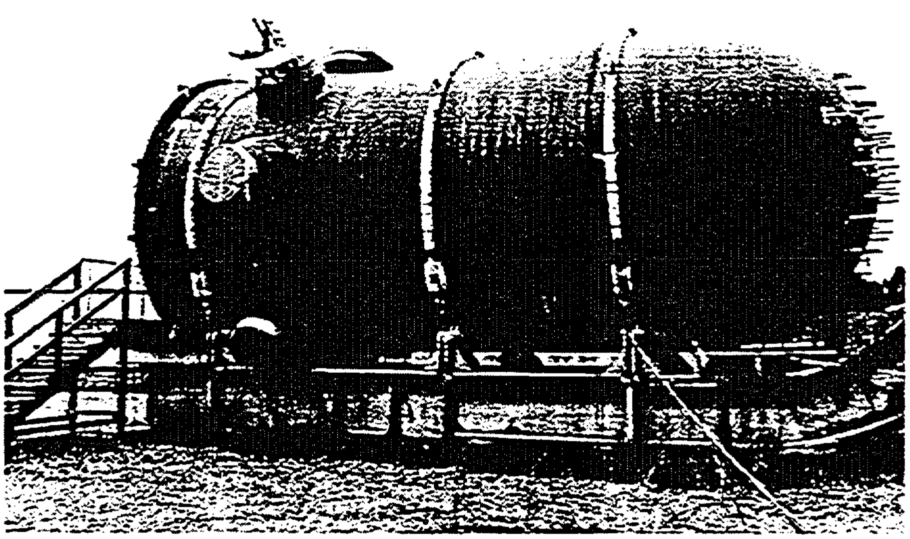

Figure 1 PVRUF Vessel

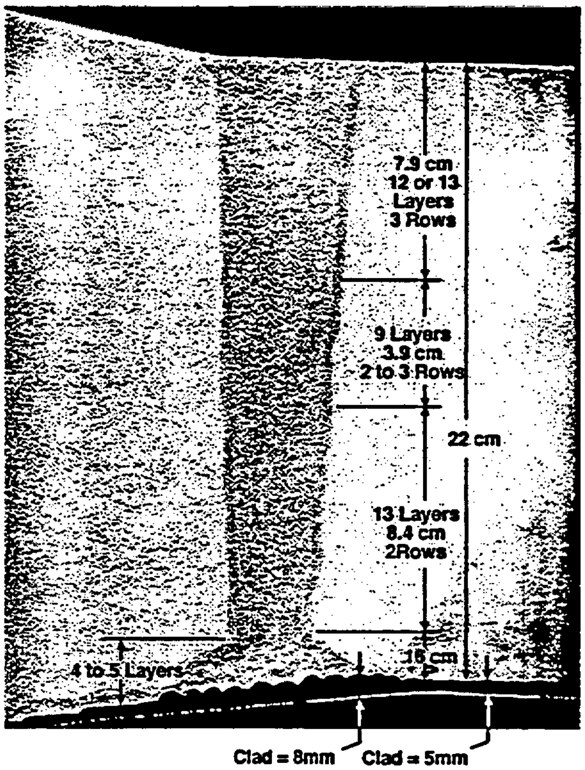

Figure 2 Cross Section of Weld Removed from PVRUF Vessel

Examination of PVRUF Vessel

Examinations of the welds and adjacent regions of base metal were performed as described in Table 1 to determine the number and sizes of fabrication flaws within the examined material. The very sensitive SAFT-UT (Synthetic Aperture Focusing Technique for Ultrasonic Examination) was used to detect and characterize flaws (Doctor et al. 1995). Following the initial SAFT-UT examinations at the ORNL site, weld segments were removed from the vessels for more detailed examinations at PNNL by SAFT-UT, radiography, and destructive metallography.

Table 1 Amount of Material Inspected by SAFT-UT in the PVRUF Vessel

Near Surface Zone	
Clad	0.027m^3 (0.95ft^3)
Clad to Base Metal Interface	4.6m^2 (50ft^2)
Weld Metal and HAZ	0.014m^3 (0.49ft^3)
Base Metal	0.073m^3 (2.58ft^3)
Remainder of Vessel Wall	
Weld Metal	0.20m^3 (7.1ft^3)
Base Metal	1.0m^3 (35.3ft^3)

Method of Examinations

The procedures used in the various steps of the examinations are described below.

SAFT-UT Examinations Performed Onsite – PNNL staff moved the SAFT-UT system to ORNL, and trained ORNL staff in operation of the system. ORNL staff examined 100 percent of the circumferential welds of the beltline region and 50% of the intermediate to upper shell course welds. Access for these examinations was from the vessel inner surface. In over 18 months of operation, the SAFT-UT system examined 20 linear meters of weld. The data collected at ORNL were sent to PNNL for detailed analyses to identify all zones that could contain the larger fabrication flaws.

SAFT-UT Examinations Performed at PNNL - Sections of welds that contained all potentially large flaws were remove from the PVRUF vessel and shipped from ORNL to PNNL for detailed examinations (Figure 3). With access from the various sectioned surfaces of the segments, it was possible to achieve an enhanced level of sensitivity and resolution. In a number of cases some flaws, which had been conservatively characterized by the preliminary examinations to be a single large flaw, were more accurately determined to be two or more closely spaced smaller flaws.

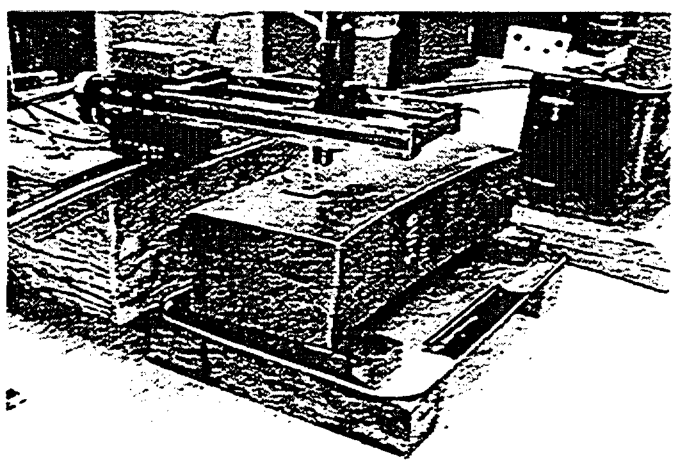

Figure 3 Scanning of Weld from PVRUF Vessel with SAFT-UT System

Destructive and Radiographic Examinations - Small cubes (one-inch size) containing larger indications were cut from the weld segments. The sizes and locations of the defects within the cubes were confirmed by radiography. The final step was to section cubes to obtain further confirmation of selected defects by metallography. The effort to confirm the number and sizes of flaws was limited to flaws in the weld metal and heat affected zone (HAZ) since these regions contained all of the largest flaws.

SAFT-UT Technique

The SAFT-UT field system is an effective method for evaluating fabrication flaws in nuclear pressure vessel material due to its validated high probability of detection of small flaws and its proven sizing accuracy for flaws in thick-section steel. The high performance of this system in this application is due to its focal properties.

In SAFT-UT, data is collected over a large area using a small transducer with a diverging sound field. High resolution and high signal-to-noise ratio images are created through computer processing of the data with SAFT algorithms. This technique, synthetic aperture focusing, has an advantage over physical focusing techniques in that the resulting image is full-volume focused over the entire inspection area. Traditional physical focusing techniques provide focused images only over a limited zone at the depth of the focus lens. In SAFT-UT, digital signal processing of the data reproduces the focal properties of a large focus transducer.

The inspection procedures were designed to cope with the clad surface roughness and the necessity for off-line analysis of the SAFT-UT data. Prior experience with vessel examinations indicated that small flaws (less than 2-mm) were expected and that 10 different ultrasonic inspection modes as indicated in Table 2 would be needed to characterize the ultrasonic indications. High spatial sampling rates produced smooth images that were able to separate small flaws that were relatively close together. Figure 4 is an example of an image constructed from the SAFT-UT data.

Table 2 SAFT-UT Inspection Plan for PVRUF

Inspection No./Type	Beam (Skew) Direction	Frequency, MHz	Y Length, cm (in.)	X Length, cm (in.)	File Size, MB
Near-Surface Zone Inspections					
1/Normal beam	NA	4.0	23 (9)	23 (9)	22
2/70°, L-Wave	+X	2.0	23 (9)	23 (9)	8
3/70°, L-Wave	+Y	2.0	23 (9)	23 (9)	8
4/70°, L-Wave	-X	2.0	23 (9)	23 (9)	8
5/70°, L-Wave	-Y	2.0	23 (9)	23 (9)	8
Inspection of the Base-Metal Weld					
6/45°, S-Wave	+X	1.5	23 (9)	28 (11)	80
7/45°, S-Wave	+Y	1.5	23 (9)	23 (9)	75
8/45°, S-Wave	-X	1.5	23 (9)	28 (11)	80
9/45°, S-Wave	-Y	1.5	23 (9)	23 (9)	75
10/Normal beam	NA	5.0	23 (9)	12.7 (5)	157

Five of the modes were used for inspection of the inner 25-mm of the vessel wall. A 4-MHz dual element normal-beam transducer was used for inspection mode #1 to provide sensitivity to volumetric flaws (porosity and slag inclusions) near the inner surface of the vessel. A 2-MHz dual element 70° angle-beam transducer was used in modes 2 through 5 to provide sensitivity to planar flaws (cracks or lack of fusion). In a similar fashion, the additional five inspection modes were used to detect and characterize indications in the vessel material more than 12-mm below the vessel inner surface.

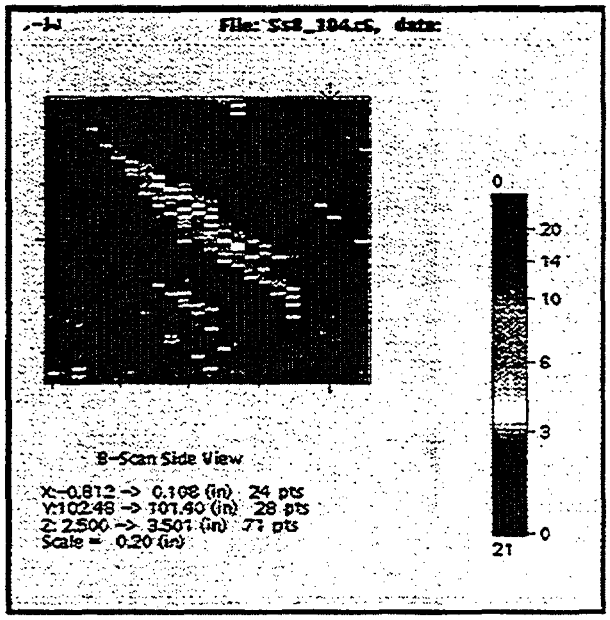

Figure 4 Embedded, Compact Time-of-Flight Shape from SAFT-UT Examination

The evaluation of the SAFT-UT data as collected at ORNL was performed using a two step procedure, which consisted of a detection process followed by a characterization process. Details of the procedure will be documented in a future report (NUREG/CR-6471). In detecting about 2500 flaw indications, the evaluation started by finding collections of adjacent pixels that made up recognizable shapes. The detection rules considered the brightness and shape of the objects in combination with the location of the object, proximity of other objects, and consideration of surrounding random noise. Estimates of the through-wall extent of the indications were recorded during the detection process. This identified the most significant flaws in the SAFT-UT data that warranted more complete characterization and measurements of through-wall extent. A significant portion of indications identified by the initial evaluation of the SAFT-UT data were multiple detections of the same indication (up to 10 times corresponding to the 10 inspection modes). Procedures and rules were developed to account for multiple detections.

SAFT-UT sizing rules were developed to conservatively size indication zones to ensure that all potentially larger flaws would be included in the validation plan. The validation effort applied additional SAFT-UT, radiography, and destructive examinations to confirm the existence and to accurately measure the size of the indications. The sizing procedure for the ORNL data characterized flaws in terms of through-wall extent, length, location relative to the inner surface of the vessel, planar or volumetric, and flaw location relative to the weld volume (weld metal, base metal, heat affected zone or cladding). The approach was to be conservative in the initial sizing and characterization in order to ensure that all potentially significant flaws would be addressed by the subsequent steps of validation and more exact sizing.

Weld segments that had more significant indications were subject to additional SAFT-UT examinations at the NDE laboratory located at PNNL. The various surfaces of these segments could be scanned, which enabled the SAFT-UT examination to access the flaws in an optimal manner for detection and sizing. A set of procedures and rules was developed for the laboratory examinations.

These procedures were similar to those described above for evaluation of the SAFT-UT data collected at ORNL from the intact vessel.

Data from Examinations

The total number of detected flaws was about 2500 with most of these flaws having through-wall dimensions of less than 4-mm. Larger flaws were confirmed and characterized on the basis of the detailed SAFT-UT, radiography, and destructive examinations. The sizes of the flaws in the material which remained at Oak Ridge were reestablished by using sizing rules adjusted on the basis of experience gained from the enhanced SAFT-UT, RT, and destructive examinations.

Categorization of Flaws

Data on the number and sizes of flaws detected from the examinations of the PVRUF vessel performed at ORNL are listed in Table 3 for flaws inside the near surface zone (inner 25-mm). Table 4 gives data for the remainder of the vessel wall (outside the near surface zone). Data in both tables have been corrected on the basis of validation and sizing measurements from PNNL's inspections of vessel segments. The zones described as clad, weld, and base metal regions correspond to the volumes listed in Table 1.

Table 3 PVRUF: Flaw Frequency in the Near Surface Zone

	Through-Wall Extent of Flaw (DZ)														
	<3 mm	3 mm		4 mm		5 mm		6 mm		7 mm		8 mm		Total >2 mm	
Zone		V	P	V	P	V	P	V	P	V	P	V	P	V	P
Clad	1148	3	0	1	0	0	0	0	0	0	0	0	0	4	0
Weld	191	1	6	0	3	1	2	0	0	0	0	0	0	2	11
Base	180	4	6	0	3	0	0	0	0	0	0	0	0	4	9
Total	1519	8	12	1	6	1	2	0	0	0	0	0	0	10	20
Total Number Characterized >2 mm															30
V = Volumetric	Total Number <3 mm														1519
P = Planar	Total Number														1549

Table 4 PVRUF: Flaw Frequency in Remainder of Vessel Wall (All confirmed flaws and unconfirmed flaws)

	Through-Wall Extent of Flaw (DZ)																
	<5 mm	5-6 mm		7-8 mm		9-10 mm		11-12 mm		13-14 mm		15-16 mm		17-18 mm		Total >5 mm	
Zone		V	P	V	P	V	P	V	P	V	P	V	P	V	P	V	P
Repair	--	0	9	0	2	0	0	0	1	0	0	0	0	0	1	0	13
Weld	653	2	17	0	7	0	1	0	0	0	0	0	0	0	0	2	25
Base	365	3	8	0	2	0	0	1	1	0	0	0	0	0	0	4	11
Total	1018	5	34	0	11	0	1	1	2	0	0	0	0	0	1	6	49
Total Number Characterized >5 mm																	55
V = Volumetric	Total Number <5 mm																1018
P = Planar	Total Number																1073

Flaws in the heat-affected zones (HAZ) were combined with those in the weld zones to construct Tables 3 and 4. It was not possible to locate the flaws relative to the weld fusion line with adequate precision to define a separate category of HAZ flaws. Base metal flaws include only those flaws which were clearly outside the weld and HAZ regions. This approach was taken to prevent the true base metal flaw density from being masked by a small fraction of the weld related flaws, which might be incorrectly located outside the weld region due to small errors in the measurements of flaw location. Clad flaws include both flaws within the clad itself and a large number of small flaws laminar (due to lack of fusion) at the clad to ferritic steel interface.

The through-wall extents (DZ) of the detected flaws were placed in discrete categories of depth ranges, with the magnitudes of these ranges being consistent with the sizing accuracy of the SAFT-UT measurements. Most of the total of the 2622 flaws are in the smallest of the depth categories (< 3-mm for near surface flaws and < 5-mm for outer region flaws). There was no observed trend of a bounding or minimum size of flaw within this smallest category of flaw depths. The observations suggested that the distribution of flaw depths were skewed towards the lower end of the depth range.

Each flaw was classified as being either volumetric or planar based solely on the SAFT-UT data. A flaw was classified as volumetric whenever the UT response provided evidence that a flaw had a volume or relatively larger separation between defect surfaces. Such flaws could range from porosity (or voids) to relatively thick slag. The classification of a flaw as volumetric was not based on detailed considerations of defect morphology or evaluations of whether a given defect had sharp geometric features, which would result in crack-like behavior for purposes of fracture mechanics evaluations. Therefore, the volumetric and planar categories of flaws were combined in the somewhat conservative applications of the PVRUF data for estimating flaw rates to be used as input to probabilistic fracture mechanics calculations.

Summary of Data

The data from the examinations provides a large amount of information on the number and characteristics of flaws in the PVRUF vessel. The characteristics include through-wall extent, length, orientation, and location relative to the inner surface of the vessel, location relative to weld, base metal and clad regions. The discussion below summarizes an evaluation of trends from the data. Future evaluations will expand on these characterizations.

Figures 5 and 6 show micrographs of two of the many flaws removed from the PVRUF vessel during the validation phase of the study. The small flaw of Figure 5 has a through-wall dimension of 1-mm and is typical of the vast majority of the detected flaws. The 17-mm flaw of Figure 6 shows the largest flaw found in the examined material of the vessel. The micrograph showed that this unusually large flaw was associated with a repair. The skewed orientation of this flaw would have made it difficult to detect during a radiographic examination performed after the repair.

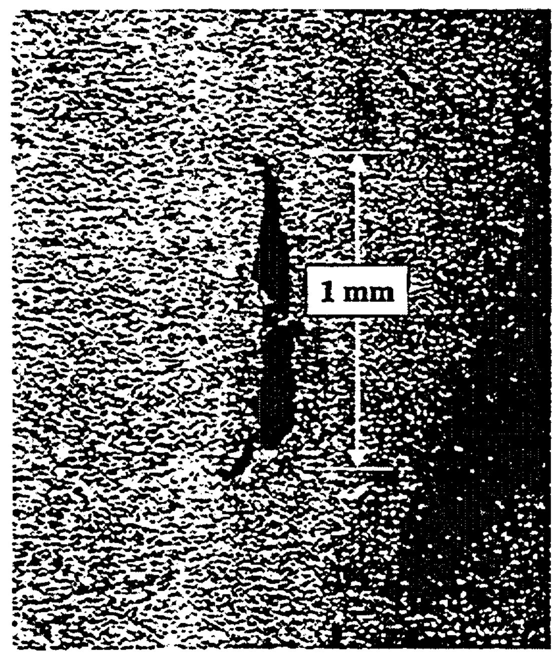

Figure 5 Micrograph of 1-mm Flaw

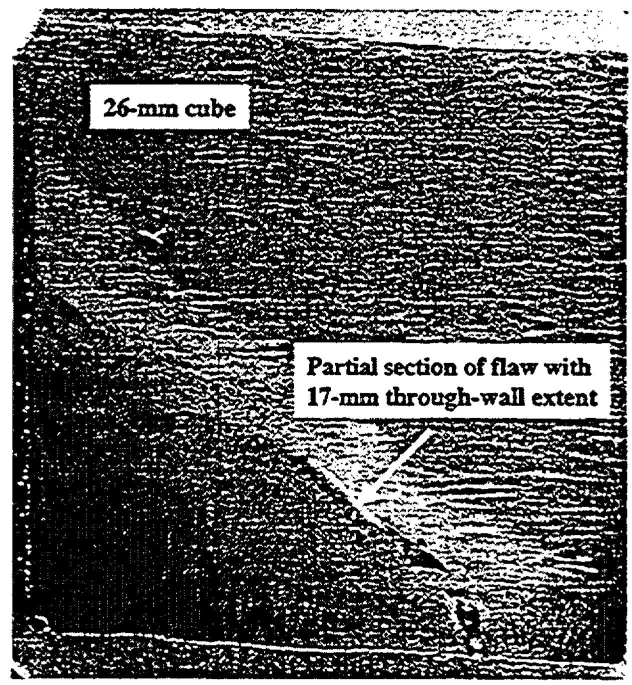

Figure 6 Micrograph of 17-mm Flaw in Repair Weld

Figure 7 presents the frequency of flaws in three different regions of the vessel as a function of their depth or through-wall extent. These three regions correspond to the inner 25-mm of the weld, the weld region outside of the 25-mm near surface zone, and finally the examined base metal region adjacent to the welds. The number of flaws in each region has been normalized in Figure 7 relative to the volume of examined material in each respective region. All types of flaws, including planar flaws, volumetric flaws, and flaws associated with weld repairs, have been combined into a single population. Cladding related flaws have been excluded.

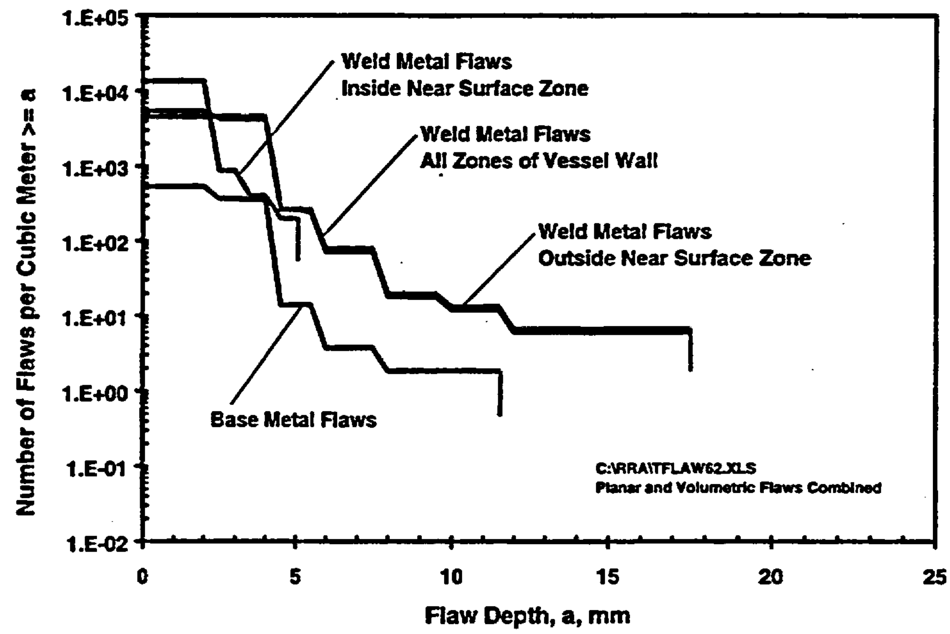

Figure 7 Flaw Frequencies from PVRUF Data

It can be seen in Figure 7 that the inner region has more flaws per cubic meter, if flaws of all sizes are included, but has fewer larger flaws on a per cubic meter basis. The flaw density for the base metal region is about a factor of ten less than the density for the weld region. Because detailed metallographic characterization of the flaws in base metal regions was beyond the scope of the study, it is not possible to estimate separate flaw densities for flaws inherent to the actual base metal material versus flaws associated with repair welds made to the base metal.

The SAFT-UT examinations also established length dimension of flaws and the locations of the flaws relative to the inner surface of the vessel. Figure 8 displays both the depths and lengths of the larger flaws in the examined weld metal showing that the largest flaws were all outside of the near surface zone. The flaw aspect ratios (ratio of flaw length to flaw depth) were randomly distributed between ratios of 2:1 to 10:1, with the deeper flaws tending to have somewhat smaller aspect ratios.

Figure 9 summarizes the data for the locations of the weld flaws relative to the clad inner surface of clad vessel. If the flaws were distributed through the vessel wall in a completely random fashion, the curve from the PVRUF data would coincide with the dashed line. For the inner half of the vessel wall, the measured flaw locations are indeed nearly randomly distributed. However, within the outer half of the vessel, the flaws tended to be concentrated in the outer regions of the wall. The reasons for this trend are not entirely clear, but can in part be explained statistically in terms of a random region within the wall that is relatively free of flaws. This is also the region of the wall for which the SAFT-UT has the least sensitivity. There may also be reasons why the welding process tends to produce the most consistent and flaw free weld deposits within this region of the wall, but these have not been evaluated.

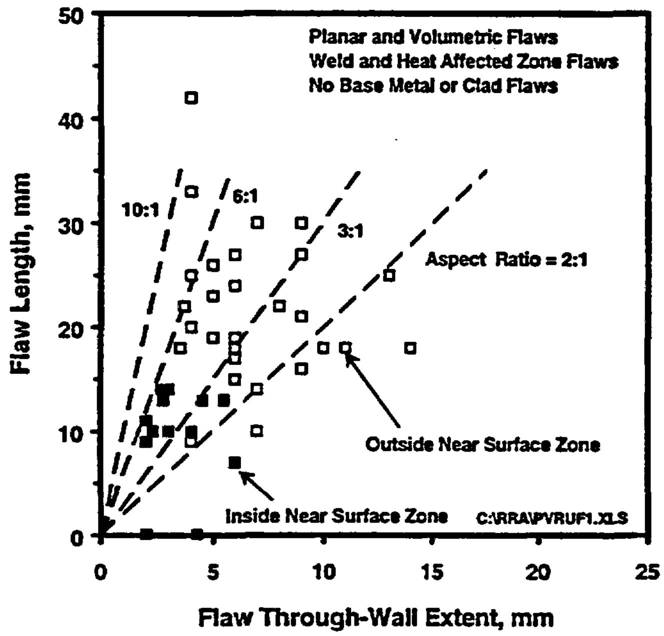

Figure 8 Flaw Lengths from PVRUF Data

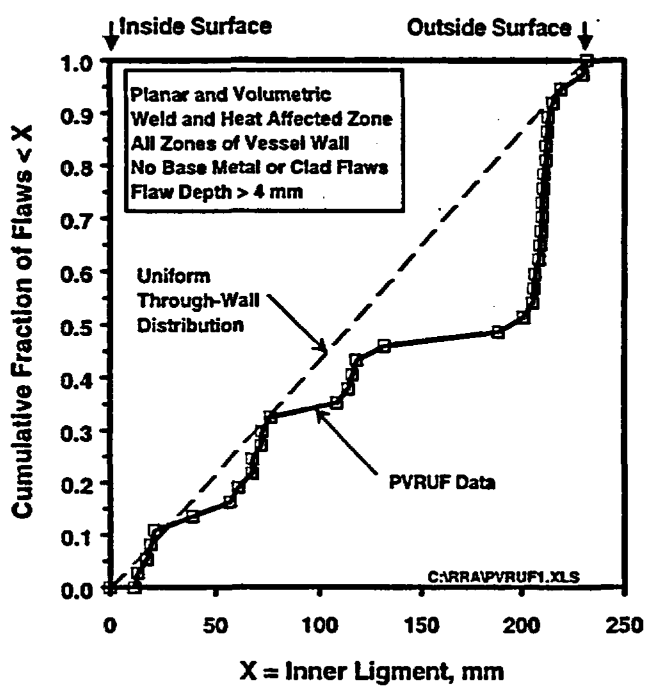

Figure 9 Flaw Locations from PVRUF Data

Simulated Flaw Distribution from RR-PRODIGAL Code

The methodology of the RR-PRODIGAL code (Chapman and Simonen 1998) uses expert elicitation and mathematical modeling to simulate the steps in manufacturing a weld and the errors that lead to the different types of welding defects as shown in Figure 10.

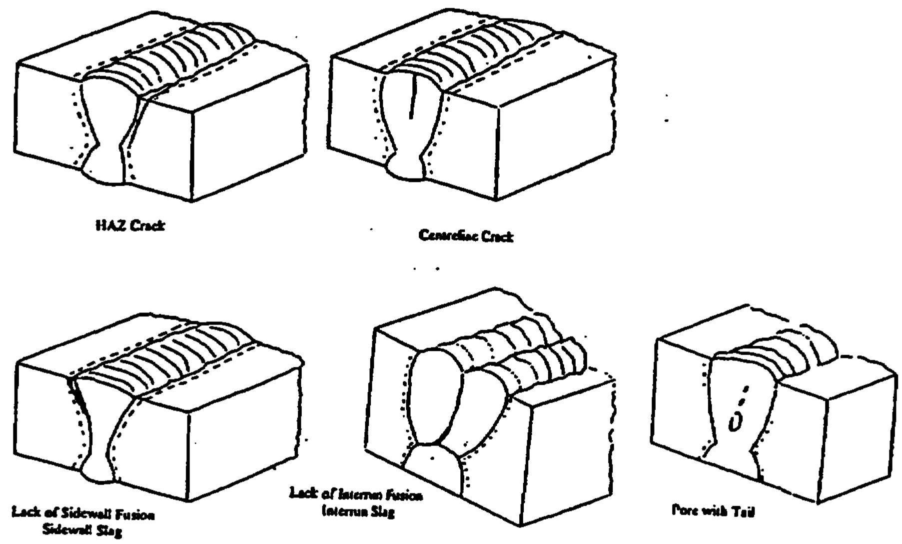

Figure 10 Types of Crack-Like Defects Simulated by RR-PRODIGAL

Description of Model

The original model was developed in the United Kingdom by Rolls Royce and Associates (RRA) based on knowledge and data provided by welding metallurgists and inspection engineers. This model was later reviewed and revised as a result of two meetings with U.S. experts to ensure that the model properly addressed flaws in U.S. reactor vessels. These experts were individuals from the major U.S. vessel manufacturing facilities, who were involved on a first-hand basis during the 1960s and 1970s with the welding and inspections of the vessels that are currently in use at operating nuclear power plants.

The types of defects that may initiate within weld beads include center cracks, lack of fusion, slag, pores with tails and cracks in heat affected zones. Various welding processes are addressed by the model including submerged metal arc and manual metal arc welding. The model includes the effects of radiographic and dye penetrant surface examinations. The probability of flaws extending from one weld bead to the next bead is also simulated by the model. Output from the simulation gives occurrence frequencies for defects as a function of both flaw size and flaw location (surface connected and buried flaws).

The expert system model uses a Monte Carlo simulation procedure to address the various activities in completing a multi-pass weld. The modeling of a weld in the PVRUF vessel began with a cross section description of the weld as shown in Figure 2. This weld is then represented as the series of layers and weld beads of Figure 11. Table 5 lists the input parameters used to specify inputs to RR-PRODIGAL for the known attributes of the weld.

Table 5 Input Parameters Used with RR-PRODIGAL to Simulate Flaws in Welds of PVRUF Vessel

Parameter	Uniform Thickness Weld	Transition Thickness Weld
Material type	A533B	A533B
Welding process	Submerged arc	Submerged arc
Weld angle	4.10 degree	3.18 mm
Upper (outer) weld width	50.8 mm	52.1 mm
Lower (inner) weld width	48.8 mm	64.7 mm
Weld passes	Layers 0-14 6.01-mm thick 2 runs per layer Layers 15-29 7.53-mm thick 3 runs per layer Layers 30-34 3.56-mm thick 2 runs per layer	Layers 0-16 6.09-mm thick Layers 17-32 5.97-mm thick 3 runs per layer Layer 33 4.0-mm thick 3 runs Layer 34 4.0-mm thick 4 runs Layer 35 4.0-mm thick 5 runs Layer 36 4.0-mm thick 6 runs
Clad material	Stainless steel Fe controlled	Stainless steel Fe controlled
Clad weld	In-line orientation 2 layers each 5.5-mm thick 5 runs per layer Each pass 0.74-mm wide	In-line orientation 2 layers each 4.0 mm thick 7 runs per layer Each pass 0.98-mm wide
X-ray	Energy level = 2.5 Mev Source diameter = 4 mm Source to film distance = 2 m Inspection mode = SWSIROOT Access factor = 1.0	Energy level = 2.5 Mev Source diameter = 4 mm Source to film distance = 2 m Inspection mode = SWSIROOT Access factor = 1.0

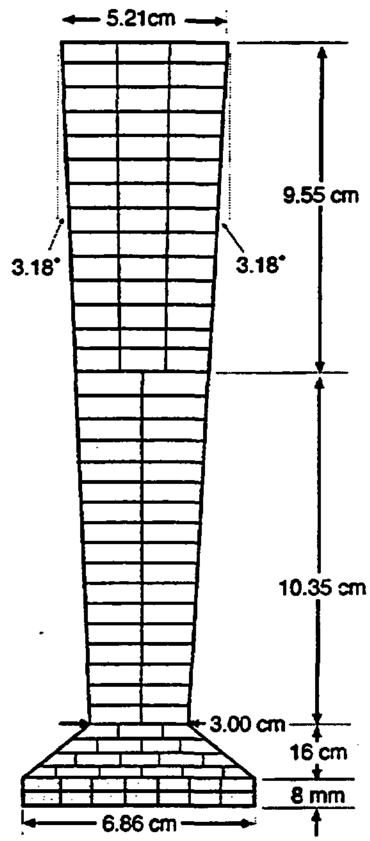

Figure 11 Model of Weld Used for Simulation of Flaws by RR-PRODIGAL

After simulating the completion of the weld, the vessel was given a reheat treatment. The inner and outer surfaces were then machined. Surface examinations of the inner and outer surfaces were performed, followed by an X-ray examination of the completed weld. In the RR-PRODIGAL simulation, all material with flaws detected by the surface and X-ray examinations was assumed to be replaced by defect-free material. The final operation after inspection was the application of cladding to the inner surface of the weld, with no inspection or repair of the cladding.

The selected parameters for the X-ray examination corresponded to typical or standard practice for the fabrication of U.S. vessels. An energy level of 2.5 Mev and a source diameter of 4-mm were specified. An access factor of 1.0 was specified to indicate that 100 percent of the weld length was radiographed. The film was assumed to be at the inner surface of the vessel, with an X-ray source-to-film distance of 2-m. A second simulation was performed as a sensitivity study, which assumed that no X-ray examination or repairs to the vessel were conducted.

Predicted Flaw Distributions

Figure 12 shows observed flaw size distributions from PVRUF vessel examinations along with predictions from the RR-PRODIGAL simulation model. For purposes of these plots, flaws within the inner and outer regions of the weld have been combined into a single population. Similarly, planar and volumetric flaws have been combined. The large number of small flaws associated with the vessel cladding have been excluded. The depths for the large number of small flaws ranging from zero- to 5-

mm in depth have been assigned values based on a set of rules which were consistent with the observation that the distribution of actual depths appeared to be skewed towards the lower end of the depth range.

Figure 12 has two curves from the RR-PRODIGAL model. The upper curve takes credit for the detection of flaws by the radiographic examinations performed in the shop by the vessel fabricator. The simulation optimistically assumes that all such detected flaws are repaired and replaced by weld metal which is free of flaws. Therefore, a second (lower) curve corresponds to predictions with no credit taken for radiographic examination (or equivalently assuming that material associated with each repair weld has just as many flaws as the material replaced by the repair).

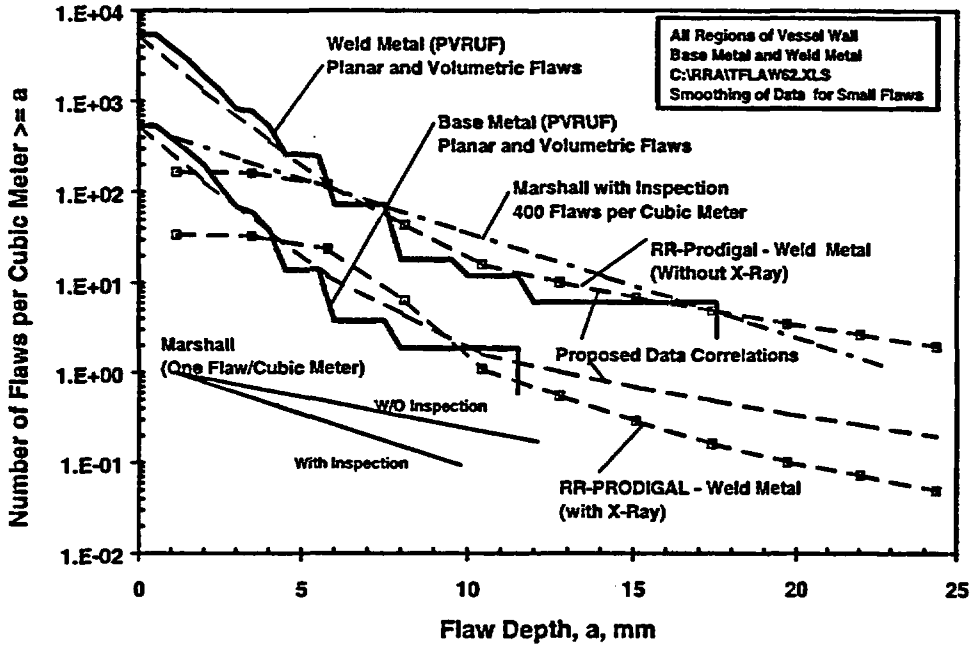

Figure 12 Observed versus Predicted Flaws for Welds of PVRUF Vessel

It is seen that the flaw frequencies from the PVRUF data are consistent with the predictions of the RR-PRODIGAL model. The correlation is best for the case which neglects the potential benefits of radiographic examinations. Shifting of this RR-PRODIGAL curve downward by a factor of ten gives a curve which is consistent with the PVRUF data from base metal regions.

The largest flaws detected in the weld and base metal regions of the PVRUF vessels had depths of about 17- and 12-mm respectively, and the flaw distribution curves for the PVRUF data are therefore truncated at these values. However, the RR-PRODIGAL model simulates the factors that can result in flaws of much greater depths, and the predicted curves are not truncated in Figure 12. Using the RR-PRODIGAL results can then be used as a basis to extrapolate the measured data to larger flaw depths.

For example, the estimated number of flaws with depths greater than 25-mm (1.0 inch) in depth is about two flaws per cubic meter. With about 0.15 cubic meters of examined weld for the PVRUF vessel, Figure 12 suggests that about one out of three vessels would have a 25-mm deep flaw somewhere in the beltline region (very likely a buried flaw). This same method of extrapolation can be applied to estimate the probability for the quarter thickness reference flaw of the ASME Code (about 60-mm deep for the PVRUF vessel). The predicted frequency for this flaw depth is 0.4 flaws per cubic meter, which implies that a quarter thickness flaw in the beltline region will occur for one vessel out of 200. Given the large uncertainty in this estimate (an order of magnitude or more), the probability for the quarter thickness flaw could be as low as one vessel out of every 2000.

Figure 12 also displays curves for the Marshall flaw size distribution. It is noted that there has been a large range of flaw densities (flaws per cubic meter) used with the Marshall distribution. This flaw density has been taken to be as small as one flaw per cubic meter (clearly inconsistent with the PVRUF data), but has more typically been assigned a value on the order of 30 flaws per cubic meter. Figure 12 shows that Marshall depth distribution with a flaw density of about 400 flaws per cubic meter predicts flaw frequencies consistent with the PVRUF data over the flaw depth range of 5 to 25-mm which is the range most important to pressurized thermal shock evaluations. It is noted that the PVRUF data shows that flaws are randomly distributed through the thickness of the vessel wall. Therefore, one should use the flaw density of 400 flaws per cubic meter in combination with a fracture mechanics model that assumes that flaws occur at random locations through the vessel wall (rather than always located at the vessel inner surface).

Summary and Conclusions

In summary, examinations of the PVRUF vessel produced a large amount of data on the number, sizes, and locations of flaws. The results of these studies and of the simulations of flaws in vessel welds with the RR-PRODIGAL code can be summarize by the following conclusions:

- The flaws in the PVRUF vessel welds have been characterized with a high level of accuracy by the SAFT-UT technique, and the validation effort has established the number and sizes of the larger flaws with a high level of confidence.

- If flaws of all sizes are considered, including flaws with depths of 1-mm or less, the measured flaw densities will be as large as 10,000 flaws per cubic meter.

- If only structurally significant flaws with depths of 5-mm or greater are included, the flaw densities decrease to 200-300 flaws per cubic meter.

- The PVRUF data also show a large number of small and structurally unimportant flaws in the cladding and at the interface between the clad and ferritic material of the vessel.

- Flaw densities in base metal regions are about an order of magnitude less than the densities for weld metal.

- A significant fraction of the larger flaws in weld material are likely to be associated with repair welded regions of the welds. Repair welding may have a similar impact on flaws in base metal regions.

- Flaws are located in a rather random manner relative to the inner surface of the vessel, with no trend of higher densities at the critical inner surface of the vessel.

- Application of the RR-PRODIGAL code to the simulation of flaws in the PVRUF vessel supports the validity of the assumptions and data on which the model is based.

- The agreement of the measured data with the predictions by the RR-PRODIGAL expert system model indicates that number and sizes of flaws in the PVRUF vessel are consistent with estimates made by welding experts based on their extensive knowledge and experience gained from the manufacture of a large number of welds similar to those in the PVRUF vessel.

- The number and sizes of flaws in the PVRUF vessel are believed to be representative of the expected flaws that exist in the reactor pressure vessels manufactured for nuclear power plants in the U.S. However, vessel-to-vessel variations in flaw distributions should be expected. Future examinations of welds from the River Bend and Shoreham plants will provide an indication of such vessel-to-vessel differences.

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NDE of Shoreham Reactor Pressure Vessel for Flaw Distribution Development

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Abstract

A joint program between EPRI and the US Nuclear Regulatory Commission (NRC) is underway to determine the distribution (size and number density) of fabrication defects in reactor pressure vessel (RPV) welds. A series of nondestructive examinations (ultrasonic – UT) are being performed on material removed from the decommissioned, unirradiated, Shoreham RPV. EPRI-sponsored UT inspections are being performed at the EPRI Nondestructive Evaluation (NDE) Center. Results obtained from the ultrasonic examinations will be used to determine appropriate areas for further evaluation through destructive sectioning and metallographic analysis. The destructive sectioning will be performed to characterize the defect type and determine the defect distribution for the material evaluated. In addition, it is anticipated that results from the NDE and destructive analyses will provide information regarding ultrasonic inspection performance and reliability for consideration in RPV operating criteria. The inspection activities being performed at the EPRI NDE Center are discussed.

Introduction

The ability to utilize RPV inspection results in plant operating criteria is a continuing goal in the nuclear industry and codes and standards organizations. The understanding of RPV defect distributions (obtained through NDE and destructive evaluations) and inspection system performance can impact plant operating criteria through the following areas:

- A revision in the assumed defect distribution used in probabilistic fracture mechanics analyses for RPV failure under postulated pressurized thermal shock (PTS) conditions
- A reduction in the assumed reference flaw size utilized in determination of RPV heatup and cooldown pressure-temperature limits

- A reduction in the inspection requirements for RPV beltline welds (including inspection frequency and material volume requirements)

A common factor in the three areas identified above is an accurate characterization of the distribution of fabrication defects (size and number density) remaining in the RPV weld materials when the vessel is initially placed in service. Several attempts at characterizing RPV defect distributions have been performed. The most notable of these, and that generally considered for PTS analyses under Regulatory Guide 1.154, is the Marshall distribution [1]. In addition, methodologies have been developed to utilize in-service inspection results in determination of a vessel-specific flaw distribution [2-3]. The application of these to RPV integrity assessment has been previously described [4].

Recent efforts by the USNRC have involved ultrasonic examination of a Combustion Engineering-fabricated RPV located at Oak Ridge National Laboratory. Preliminary results have been reported elsewhere that indicate a higher number density of defects as well as a larger fraction of small near-surface defects when compared to the Marshall distribution [5]. Additional work is underway to provide destructive verification of the indications recorded from the UT exams.

As NDE techniques continue to advance and the ability to detect smaller imperfections in RPV materials improves, care must be exercised regarding the relative subjective classification of UT-detected defects as specific initiation sites for subsequent crack propagation during plant operation. The categorization of all UT indications as actual RPV defects for consideration in RPV integrity analyses can lead to an unrealistically conservative assessment of vessel condition.

The utilization of inspection results in plant operating criteria requires that accurate information be available regarding characterization of the type of defects detected and that performance characteristics of the inspection system be quantified. This will help avoid including benign metallurgical microstructural features as defects in flaw distribution development and result in a more realistic assessment of flaw condition.

Characterization of the flaw distribution of the Midland vessel, based on UT examinations and destructive analysis, has been reported [6]. Efforts to quantify inspection system performance have also been reported [3]. These studies provided insight into the application of inspection results and a more realistic flaw distribution to RPV integrity assessment. However, additional work was recommended to provide further information necessary to reduce the present over conservatism associated with RPV flaw distribution assumptions.

Program Description

To provide additional information regarding RPV flaw distribution development, a joint program has been developed between EPRI and the USNRC to investigate the distribution (size and number density) and characterization of post-fabrication defects in RPV weld materials. The program involves inspection and destructive analysis of material removed from the unirradiated Shoreham RPV.

The Shoreham RPV is a boiling water reactor vessel fabricated by Combustion Engineering from SA533, Grade B, Class 1 plate material. Since plate material was utilized during fabrication, the Shoreham RPV contains both circumferential and axial welds. Plate sections were first welded together to form individual shell courses. The shell courses were then welded to form the vessel shell. Nominal thickness of the vessel is 6.2 inches with a nominal inside radius of 110.15 inches.

Following decommissioning of the Shoreham facility, three separate shell "rings", as well as the top and bottom heads, were purchased by Baltimore Gas & Electric (BG&E) for material evaluation studies. Results of these evaluations have been reported elsewhere [7-8]. The three shell course "rings" obtained by BG&E are described in Table 1.

Table 1. Shoreham Reactor Vessel Shell Rings

Ring	Weld ID	Weld Wire Heat	Flux Type	Flux Lot	Weld Elevation (inches)	Azimuth (degrees)	Weld Length (inches)	Weld Description
A	13-308	305414	0091	3947	723		731	flange to upper shell girth
	1-308-A	20291/12008	1092	3854	665-723	30	34[1]	upper shell axial
	1-308-B					150	34[2]	
	1-308-C					270	58	
B	1-308-A	20291/12008	1092	3854	581-665	30	84	upper shell axial
	1-308-B					150	84	
	1-308-C					270	84	
C	1-308-A	20291/12008	1092	3854	553-581	30	28	upper shell axial
	1-308-B					150	28	
	1-308-C					270	28	
	4-308-A	33A277	0091	3922/3977	553	–	693[3,4]	upper to upper-intermediate girth
	1-308-D	IP2809/IP2815	1092	3854	505-553	0	48	upper-intermediate axial
	1-308-E					120	48	
	1-308-F					240	0[4]	

1. 6T x 12W x 24L-inch section previously removed. Original weld length = 58 inches
2. 6T x 12W x 24L-inch section previously removed. Original weld length = 58 inches
3. 6T x 22W x 24L-inch section previously removed. Original weld length = 731 inches
4. Axial weld 1-308-F (including 13L-inch section from 4-308-A) previously removed from Ring C.

Ring A contains the vessel flange, the flange to upper shell course girth weld, and the upper shell course axial welds. The girth weld is a Linde 0091 flux, single arc weldment made with weld wire heat number 305414, which contains approximately 0.35 weight percent copper and 0.60 weight percent nickel. The axial welds are tandem Linde 0092 weldments made with weld wire heats 20291/12008, which contain approximately 0.20 weight percent copper and 0.88 weight percent nickel.

Ring B contains four manually welded steam outlet nozzles and contains the same axial welds as ring A.

Ring C contains the circumferential weld joining the upper and upper-intermediate shell courses. Ring C also contains two distinct sets of axial welds: (1) a continuation of the upper shell course axial welds contained in rings A and B, and (2) portions of the upper-intermediate axial welds. The girth seam is a Linde 0091 single arc weldment made using weld wire hear 33A277 which contains approximately 0.2 weight percent copper and 0.2 weight percent nickel. The upper-intermediate shell axial welds are Linde

1092 tandem arc weldments made using weld wire heats 1P2809 and 1P2815. These heats contain approximately 0.25 weight percent copper and 0.8 weight percent nickel.

A schematic of the upper portion of the Shoreham RPV showing the locations of the axial and circumferential welds and their relative azimuthal position, is shown in Figure 1. Rings A, B, and C, as delivered to BG&E, are indicated and labeled accordingly. Note that each individual ring was sectioned into smaller pieces, as indicated by dotted lines in Figure 1, to facilitate shipment to BG&E. Ring C was sectioned into 3 equal, 120°, arc length segments as follows: 90° - 210°, 210° - 330°, and 330° - 90°. Ring B was initially forwarded to Pacific Northwest National Laboratory (PNNL) for UT inspection using the PNNL-developed Synthetic Aperture Focusing Technique (SAFT) inspection system. Ring C was initially forwarded to the EPRI NDE Center for UT inspections. Results from both inspections will be used to govern subsequent destructive analysis to be performed by the USNRC in fiscal year 1999.

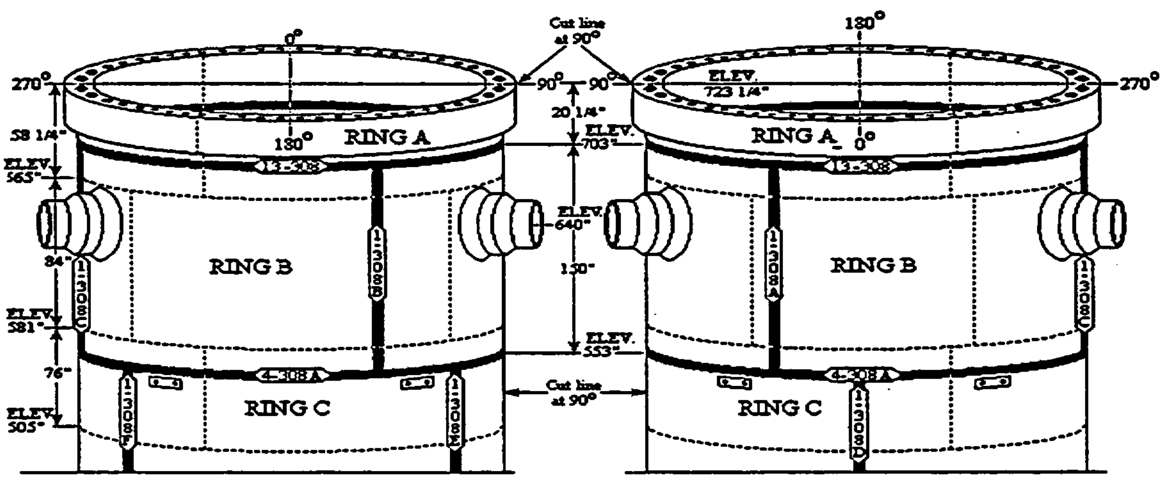

Figure 1. Schematic of Shoreham RPV - Rings A, B, and C

Prior to delivery of Ring C to EPRI, the three individual sections were further divided to facilitate shipping and to accommodate size restrictions regarding inspection equipment at the NDE Center. The resulting segments are shown schematically in Figure 2.

EPRI UT Inspections

Plates Examined

Three of the six vessel plates have been examined. These plates are designated 90° - 164°, 255° - 330°, and 330° - 45° to correspond with their approximate cuts, as provided by BG&E. All of the following measurements are from the OD surface.

The plate designated 90° - 164° contains 146" (3705 mm) of the circumferential weld #4-308A. The circumferential weld is a straight butt joint with a backing ring which has been removed as shown in Figure 3. Approximately 15" of the weld in this plate was only inspected from one side of the weld due to a welded lug on the OD. The remainder of the weld was accessible. There are two axial welds in this plate with a double U joint preparation (see Figure 3). Axial weld #1-308B is located at 150° on the upper shell and is 27.5" (700 mm) in length. Approximately 7" of this weld was not inspected due to a welded lug on the OD centerline of the weld. The remainder of the weld was accessible. Axial weld #1-308E is located at 120° on the lower shell and is 48.25" (700 mm) in length (see Figure 4).

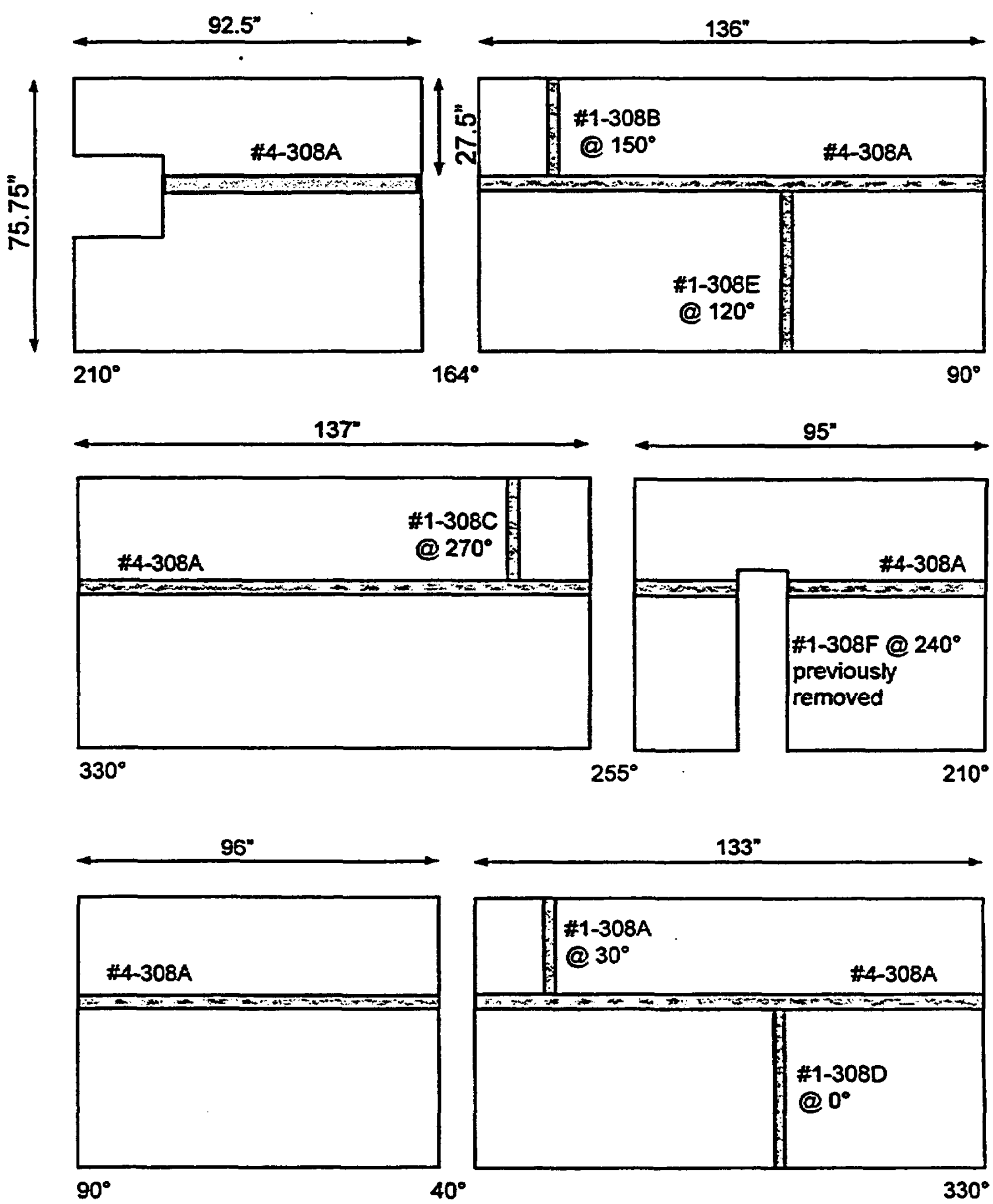

Figure 2. Schematic of Shoreham Ring C Segments

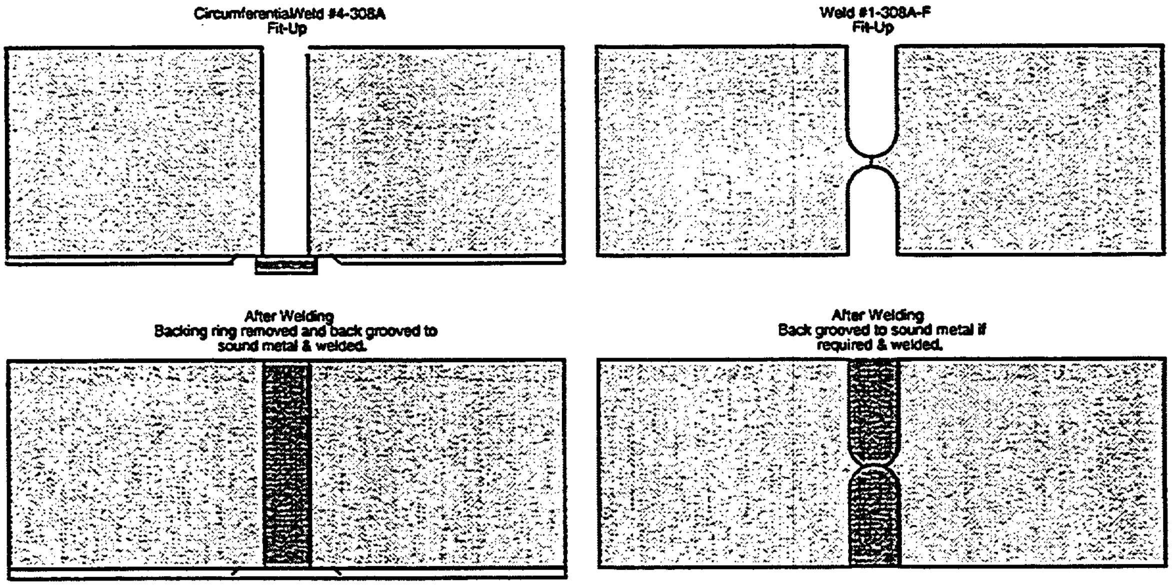

Figure 3. Butt-Joint Weld Profile and Double U-Joint Weld Profile

The plate designated 255° - 330° contains 147" (3730 mm) of the circumferential weld #4-308A. Approximately 15" of the weld in this plate was only inspected from one side of the weld due to a welded lug on the OD. The remainder of the weld was accessible. The axial weld in this plate was a double U joint preparation. Axial weld #1-308C is located at 270° on the upper shell and is 27.5" (700 mm) in length. Approximately 7" of this weld was not inspected due to a welded lug on the OD centerline of the weld. The remainder of the weld was accessible (see Figure 5).

The plate designated 330° - 45° contains 143" (3625 mm) of the circumferential weld #4-308A. There were no obstructions for inspection of the welds in this plate. The two axial welds in this plate were of a double U joint preparation. Axial weld #1-308A is located at 30° on the upper shell and is 27.5" (700 mm) in length. Approximately 7" of this weld was not inspected due to a welded lug on the OD centerline of the weld. The remainder of the weld was accessible. Axial weld #1-308D is located at 0° on the lower shell and is 48.25" (700 mm) in length (see Figure 6).

The total length of circumferential weld #4-308A inspected was 435.5" (11,060 mm). There are five axial weld to circumferential weld intersections. The total length of the five axial welds inspected was 179" (4545 mm).

The three remaining plates were not examined at this time. The plate designations are 45° - 90°, 164° - 210°, and 210° - 255°.

The plate designated 45° - 90° contains 100.5" (2550 mm) of the circumferential weld #4-308A. Approximately 15" of the weld in this plate was only inspected from one side of the weld due to a welded lug on the OD. The remainder of the weld was accessible. There were no axial welds in this plate (see Figure 7).

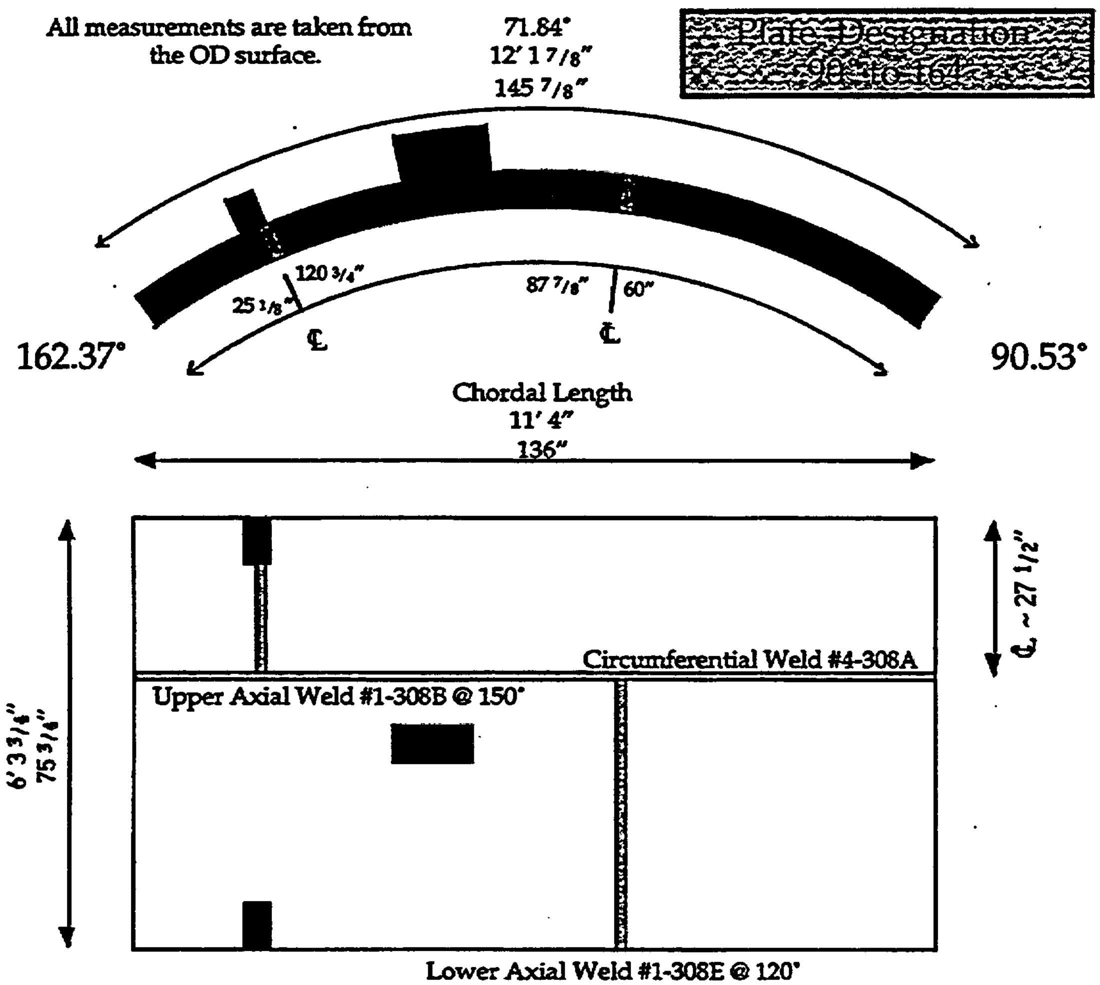

Figure 4. Plate 90° - 164°

The plate designated 164° - 210° contains 96" (2435 mm) of the circumferential weld #4-308A. Approximately 24" (610 mm) of the weld in this plate was previously removed, leaving 72" (1825 mm) of inspectable circumferential weld. There were no axial welds in this plate (see Figure 8).

The plate designated 210° - 255° contains 98" (2490 mm) of the circumferential weld #4-308A. The intersection of this weld along with the lower axial weld #1-308F was previously removed. Approximately 13" of the circumferential weld was removed, leaving 85" (2160 mm) of inspectable weld (see Figure 9).

The total length of circumferential weld #4-308A located in the three plates that was not inspected at this time is 294.5" (7,480 mm), minus the 37" (940 mm) that had been previously removed, leaving 257.5" (6540 mm) of inspectable weld material. There was no axial weld material remaining in these three plates.

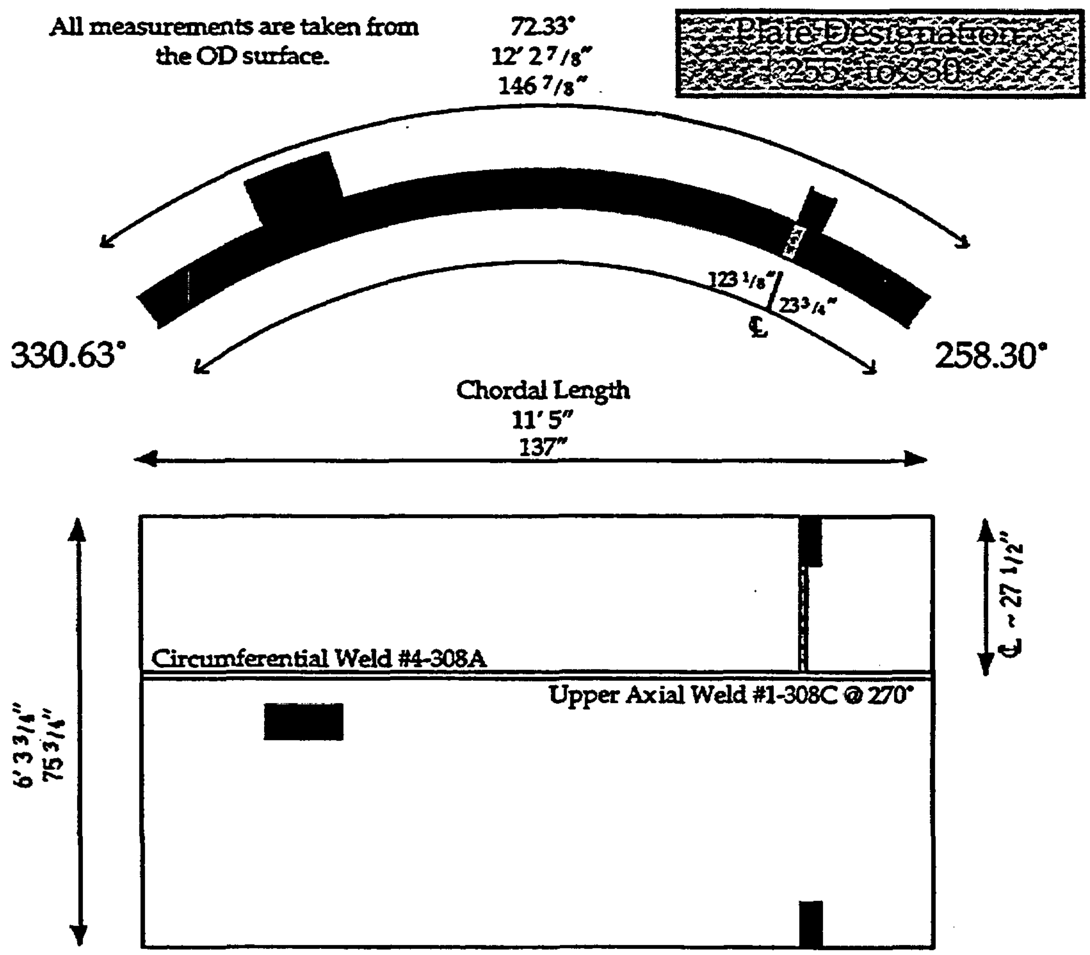

Figure 5. Plate 255° - 330°

A summary of the delivered plate material follows:

- Radius
  - Outside = 116.34" (2,955 mm)
  - Inside = 110.15" (2,800 mm)
- Diameter
  - Outside (OD) = 232.68" (5,910 mm)
  - Inside (ID) = 220.31" (5,595 mm)
- Circumference
  - OD = 731.00" (18,565 mm)
  - ID = 692.12" (17,580 mm)
- Nominal thickness = 6 3/16" (157 mm)
- OD Inch/Degree = 2.031" (51.5 mm/Degree)
- OD Degree/Inch = 0.492° (0.0194 Degree/mm)
- Weld Seams
  - Circumferential Weld #4-308A (Measurements from the OD)

Current Inspection = 435.5" (11,060 mm)
Future Inspection = 257.5" (6,540 mm)
Previously Removed = 37" (940 mm)
Total Inspectable = 693" (17,600 mm)
Inspectable & Removed = 730" (18,540 mm)
Cut Loss ~ 1" (25 mm)
Total Length = 731" (18,565 mm)

- Axial Welds
  Upper Axial Weld #1-308A, Current Inspection = 27.5" (700 mm)
  Upper Axial Weld #1-308B, Current Inspection = 27.5" (700 mm)
  Upper Axial Weld #1-308C, Current Inspection = 27.5" (700 mm)
  Lower Axial Weld #1-308D, Current Inspection = 48.25" (1225 mm)
  Lower Axial Weld #1-308E, Current Inspection = 48.25" (1225 mm)
  Lower Axial Weld #1-308F, Previously Removed = 48.25" (1225 mm)

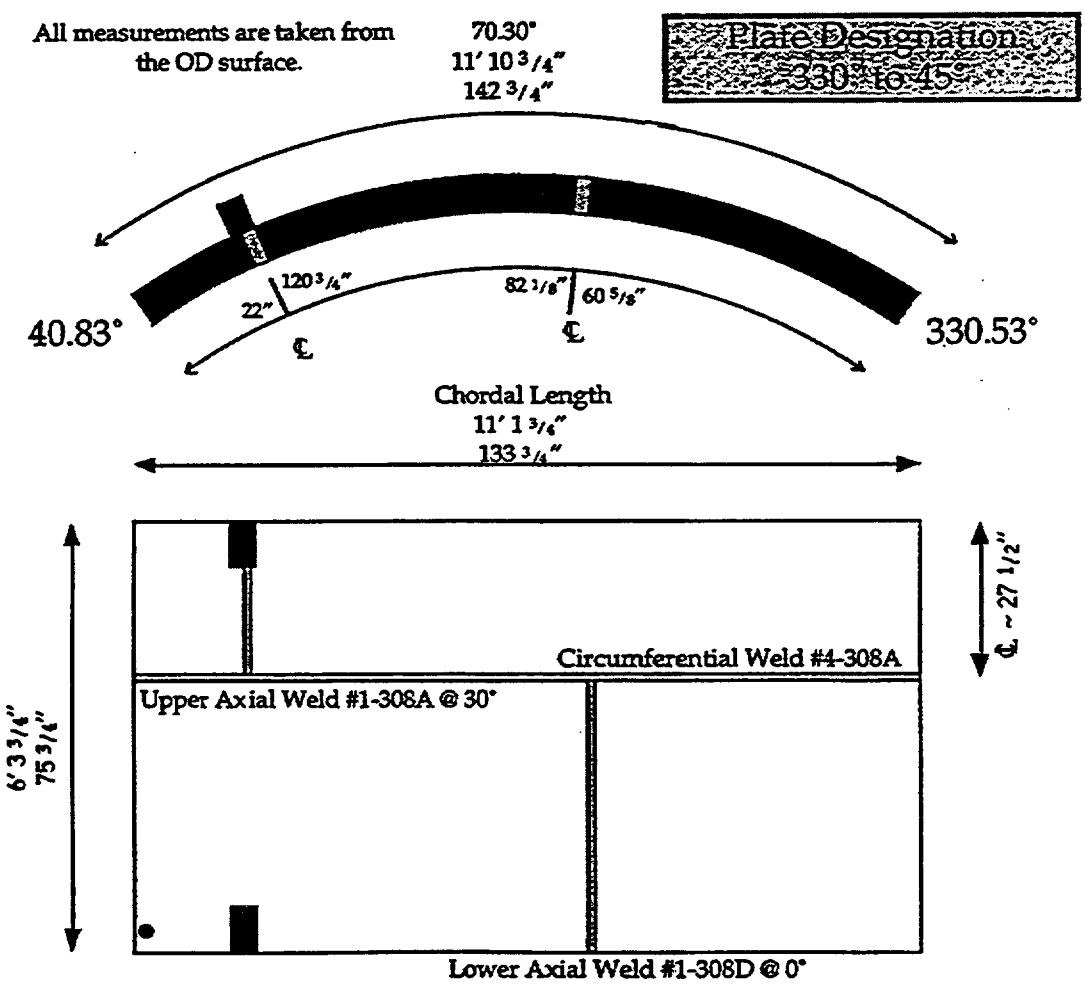

Figure 6. Plate 330° - 45°

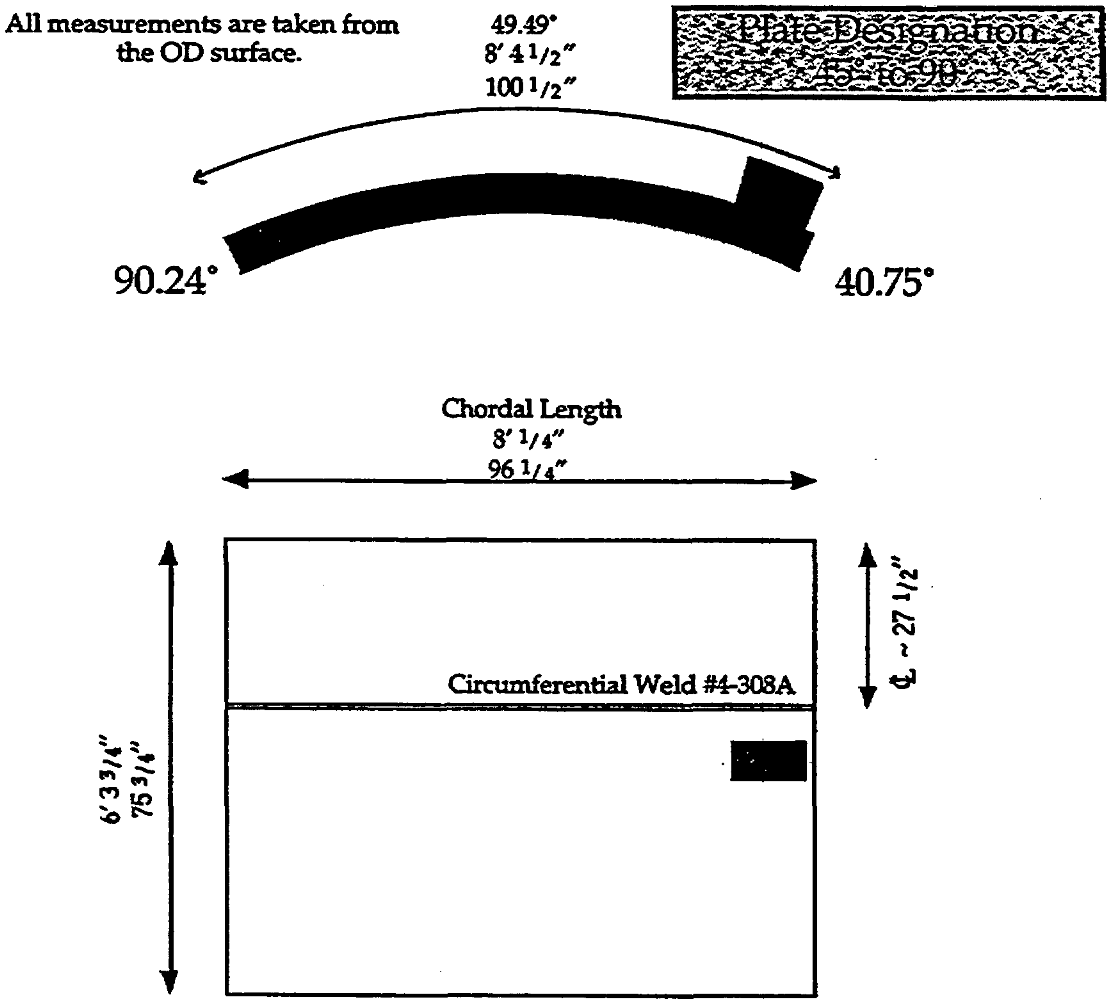

Figure 7. Plate 45° - 90°

Plate Preparation

The four large weld lugs located at 45°, 135°, 225°, and 315° were trimmed by flame cutting to allow for a greater inspection area. The plates were sanded to remove rough oxidation over the inspection surface. The weld profile was smoothed and chemically etched to identify the weld center. The circumferential and axial intersections were also chemically etched. Once the centerline was determined it was scribed onto the surface to aid in the alignment of the UT scanner.

UT Examinations Performed

An automated UT system manufactured by R/D Tech was used to conduct the inspection. A combination of flatbed and crawler type scanners was utilized. A contact UT technique with a water flood for coupling was used. An ASME type calibration block was used for setting up the electronic DAC and calibrating the transducers. An IIW calibration block was used to determine the transducers' induced angle.

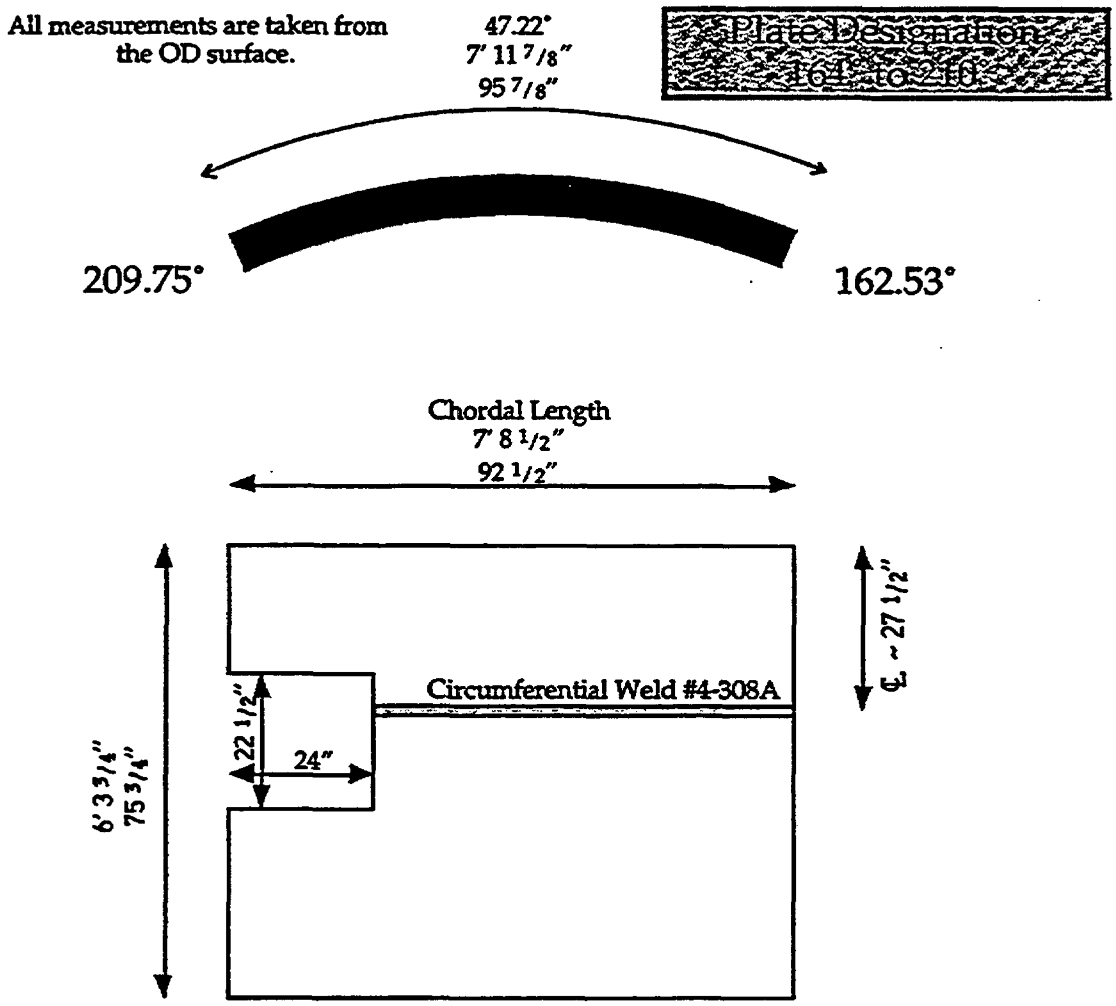

Figure 8. Plate 164° - 210°

The scan dimensions were as follows:

- An area sufficiently large to encompass the full cross section of the weld
- 0.1" (2.54 mm) interval in the scan direction
- 0.2" (5.08 mm) interval in the index direction
- 15 MHz digitization rate used for 45° shear and 60° longitudinal scans
- 30 MHz digitization rate used for 0° and 70° longitudinal scans

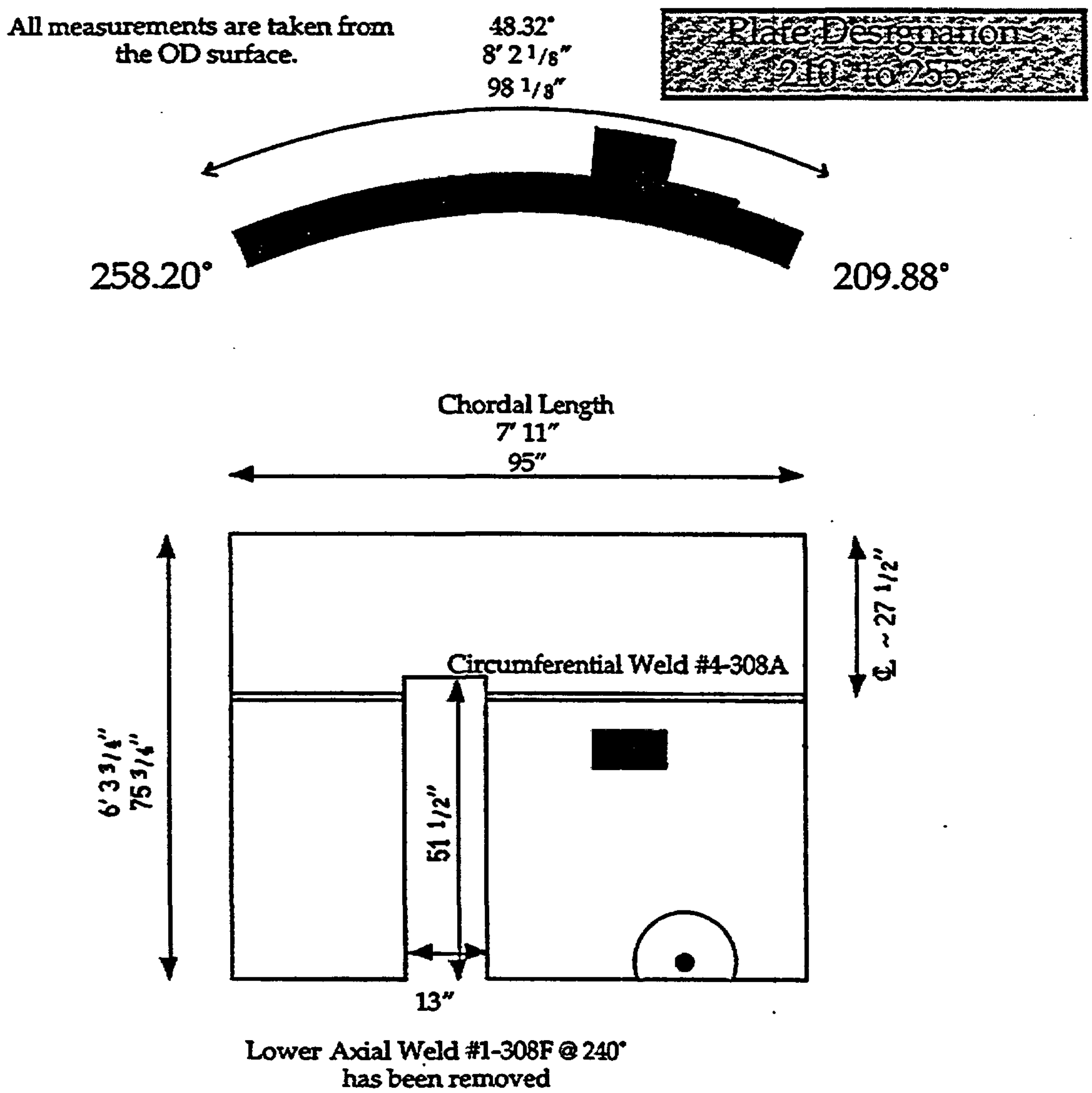

Figure 9. Plate 210° - 255°

The transducers used and the scans conducted are as follows:

- 0° longitudinal-wave (see Figure 10)
  - Single element
  - Single scan from the OD surface perpendicular to weld

- 45° shear-wave (see Figure 11)
  - Single element
  - Four scan directions from the OD
    - +/- axial scans
    - +/- circumferential scans

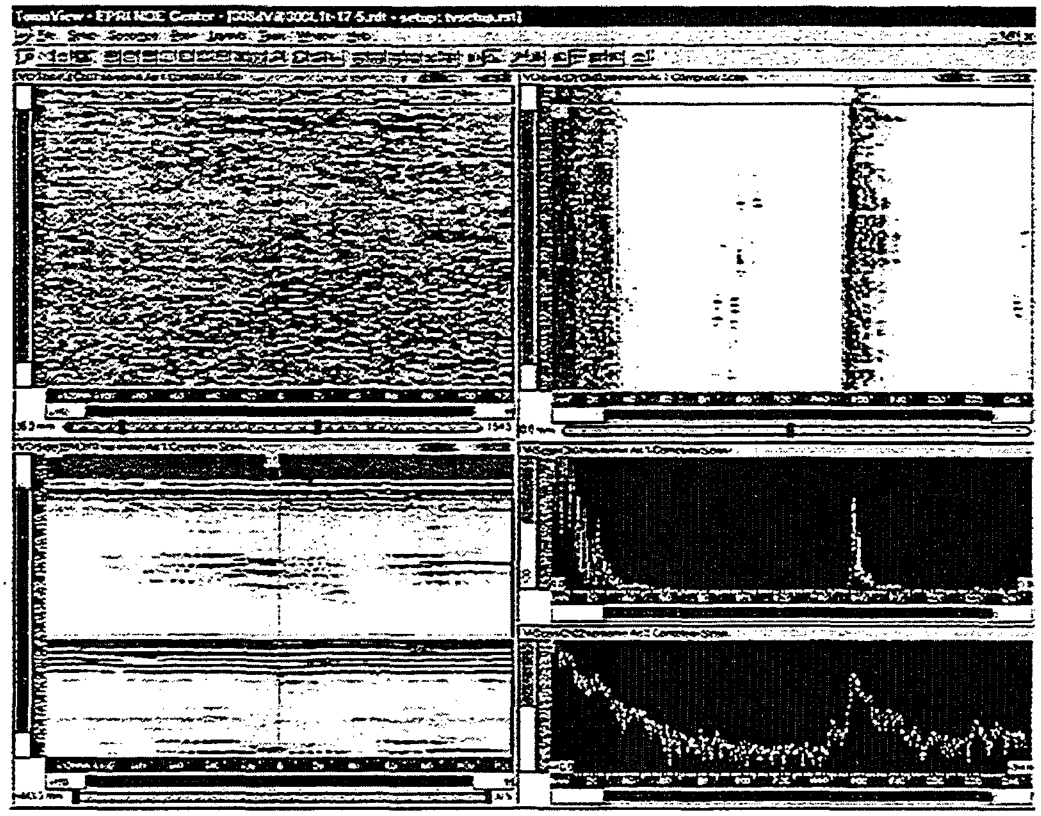

Figure 10. A-, B-, C-, and D-Scans of 0° Longitudinal-Wave Linear Data and Log-Amplified A-Scan

- 60° longitudinal-wave (see Figures 12 & 13)
  - Dual element
  - Four scan directions from the OD
    - +/- axial scans
    - +/- circumferential scans

- 70° longitudinal-wave
  - Dual element
  - Four scan directions from the ID
    - +/- axial scans
    - +/- circumferential scans

Two types of wave-forms were collected (see Figure 14):

- Linear data
  - The full RF-waveform saved
  - Gain setting adjusted to give a 25% full screen height from the clad roll
- Log amplified data
  - The full RF-waveform saved
  - Left at calibration gain setting

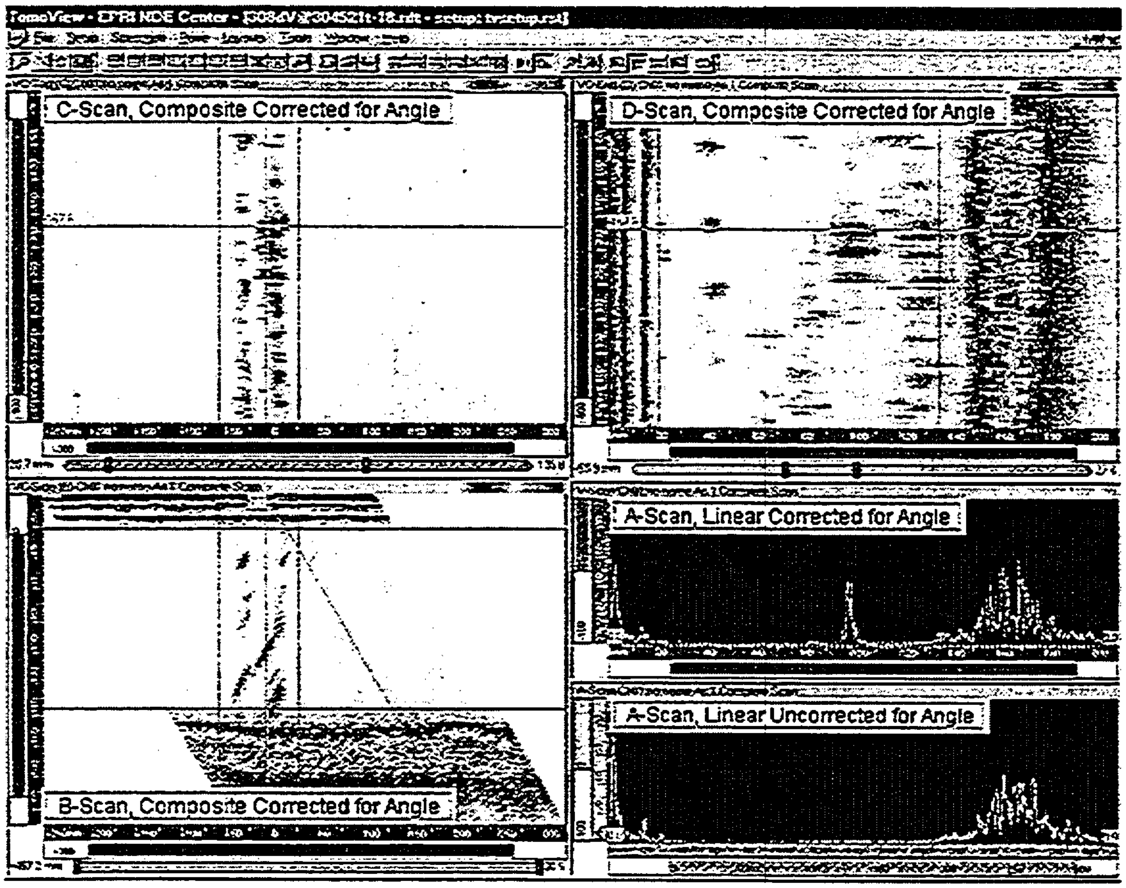

Figure 11. A-, B-, C-, and D-Scans of 45° Shear-Wave Linear Data

Linear data was collected since, historically, this has been the standard method to collect, review, and characterize indications. A disadvantage to this data type is that an indication can become saturated. It then is not possible to accurately characterize it. The indication would then need to be scanned again at a lower gain setting. The advantage of the log amplified data is that it does not become saturated. Therefore, an indication which would have been saturated in the linear data can still be characterized with the log amplified data.

After the data was collected it was saved to CD-ROM. Ultrasonic data collection has been completed on the third plate.

Analysis

A very limited amount of analysis was performed at the time of collection. The full analysis was initiated after completion of data collection and is presently ongoing. The analysis is being conducted in accordance with the ASME code. The analysis and manipulation of the data will not affect the raw data file.

While conducting the analysis the data can be presented in a number of ways, as follows:

• RF or A-scan presentations
• B-scan, C-scan, D-scan presentations
  - Single B, C, D-scan or
  - Composite B, C, D-scan
• A, B, C, D-scan
  - Presented in as-scanned (uncorrected for angle) or
  - Presented in a corrected-for-angle presentation (Figures 10, 11, 12, & 13)
• Adjustable color scale
• Overlays can be employed, saved as a separate associated file
• Other features

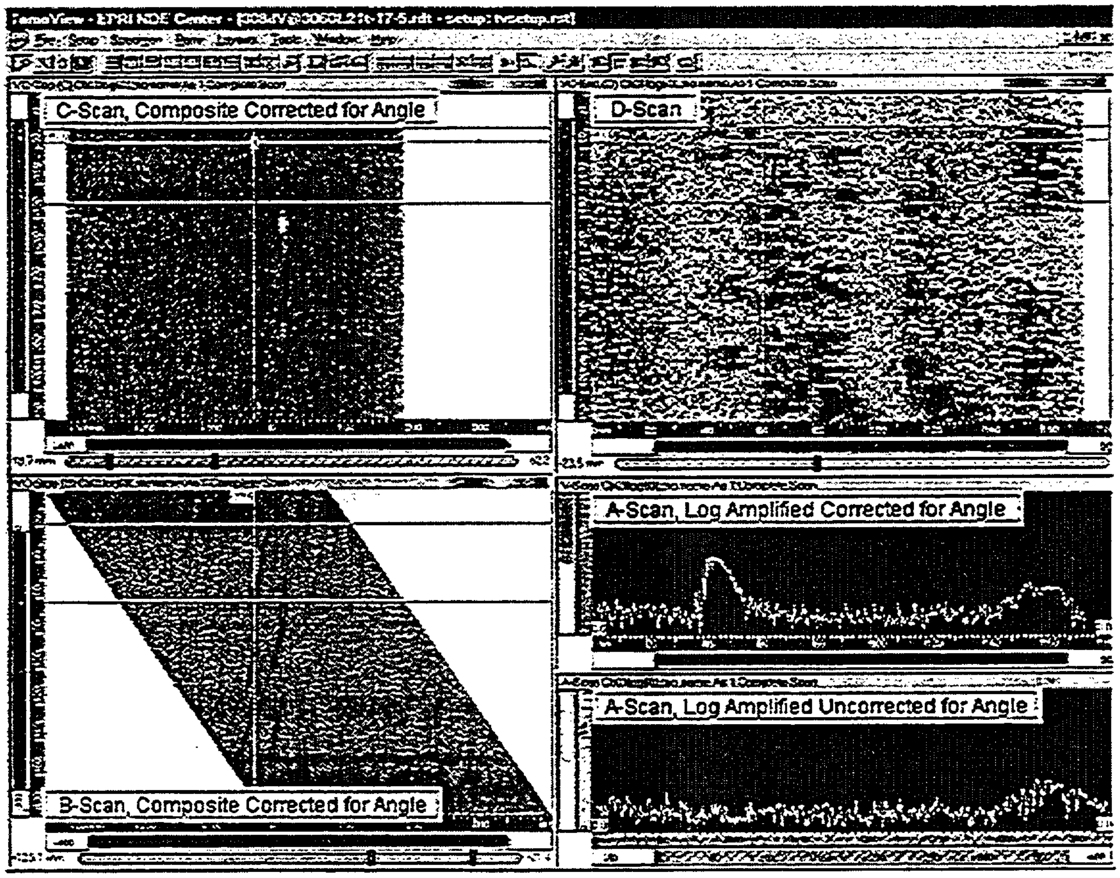

Figure 12. A-, B-, C-, and D-Scans of 60° Longitudinal-Wave Log-Amplified Data

The original radiographs and reader sheets from the radiographic inspection (RT) conducted during the manufacturing of the Shoreham Vessel were also located. The RT coordinate system and the vessel coordinate system were correlated. Radiographs are presently being digitized and will be used to complement results from the UT analysis. Indications noted on the RT reader sheets will be compared with the UT results. The digitized radiographs will also be scrutinized if no indication was reported on the reader sheet but was found during the UT analysis. The digitized radiographs will serve as an additional source of information to aid in the plate evaluation.

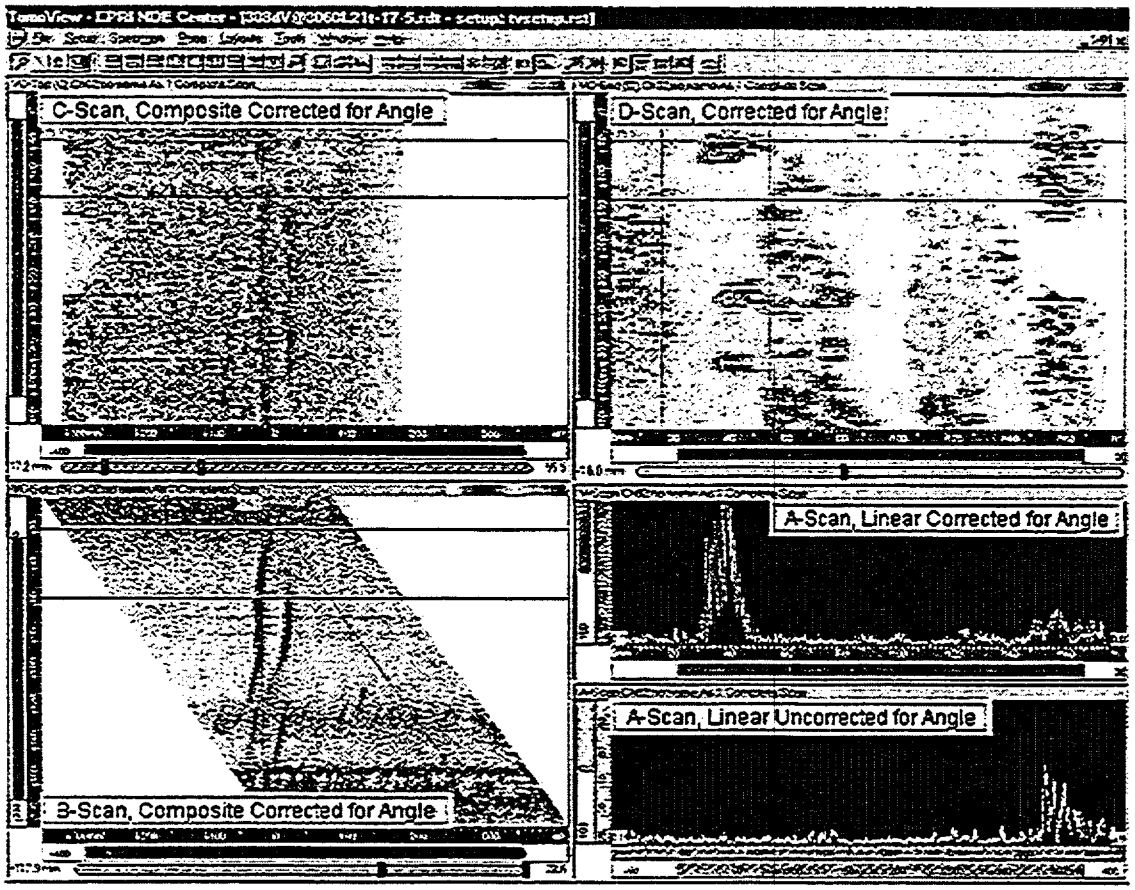

Figure 13. A-, B-, C-, and D-Scans of 60° Longitudinal-Wave Linear Data

Future Efforts

After all UT inspections have been completed the plates will be destructively sectioned. The proposed cut lines can be seen in Figures 15, 16, and 17. A cut that is perpendicular to the weld will be determined to be free of relevant indications by referring to the UT data. If relevant indications are found the cut line shall be adjusted as required. The resulting sectioned pieces will be ~11" (280 mm) in width to provide sufficient material for proposed fracture mechanics specimens (see Figure 18). The sectioned pieces will be in various lengths and will be kept to a manageable size to accommodate ease of machining, shipping, and further UT. The axial and circumferential "T" intersections will be sectioned intact as this volume of material is considered to have a greater chance of containing manufacturing defects. After sectioning, the faces parallel to the weld seams will be machined for better ultrasonic coupling.

Further ultrasonic inspections will be conducted after sectioning of the plates has been completed. The UT technique will be a focused 0° immersion from the machined face (see Figure 18). A fine resolution scan will be conducted. This will allow for a more precise placement of indications throughout the thickness of the weld and an accurate characterization of indications.

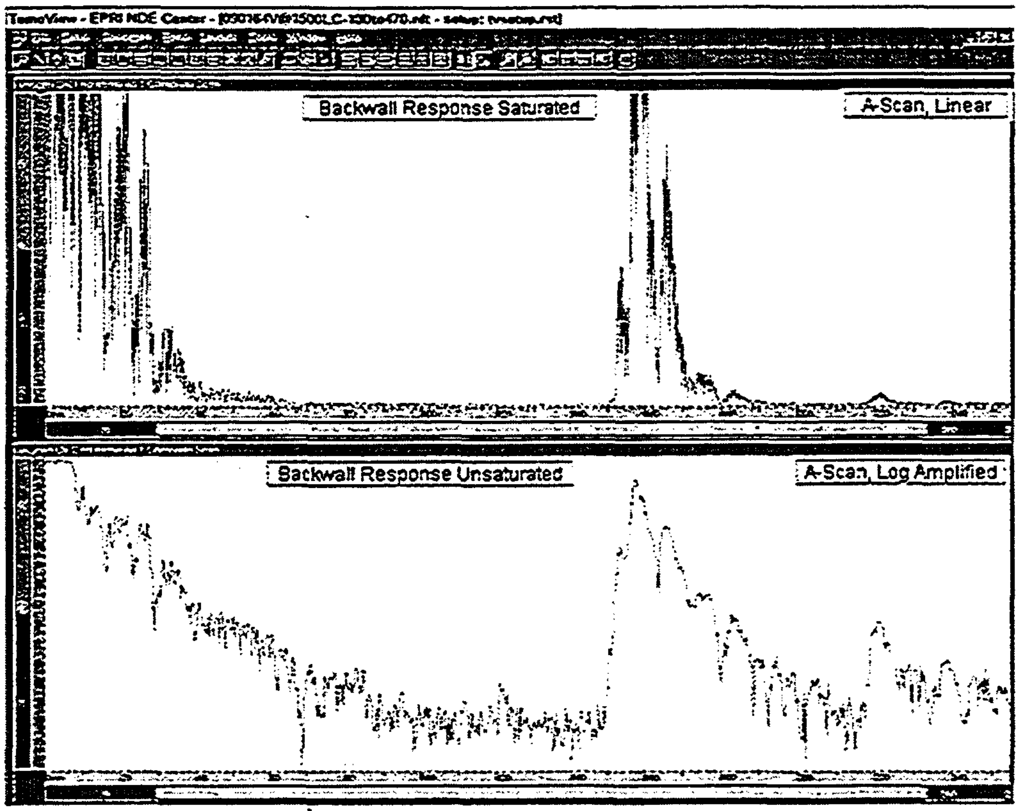

Figure 14. Linear and Log-Amplified A-Scans with Data from the Same Location

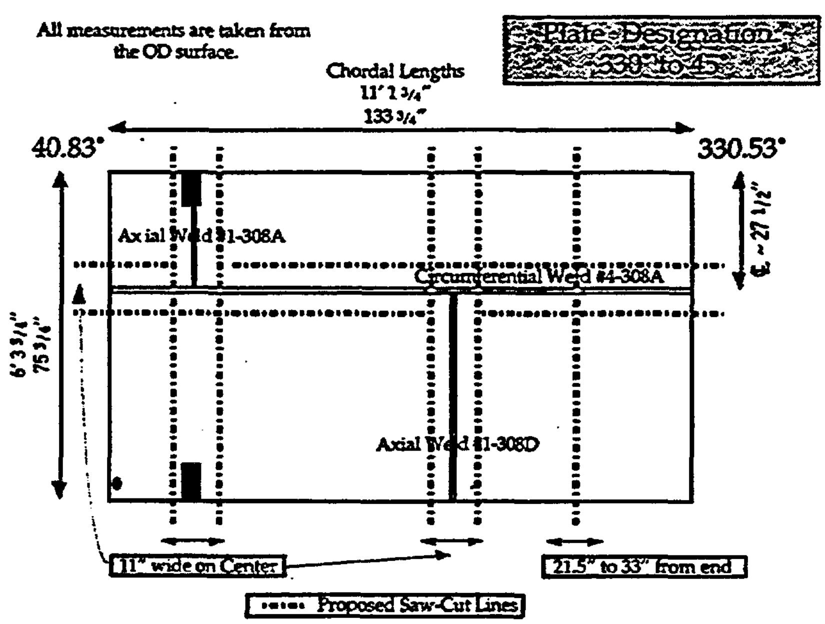

Figure 15. Proposed Sectioning of Plate 330° - 45°

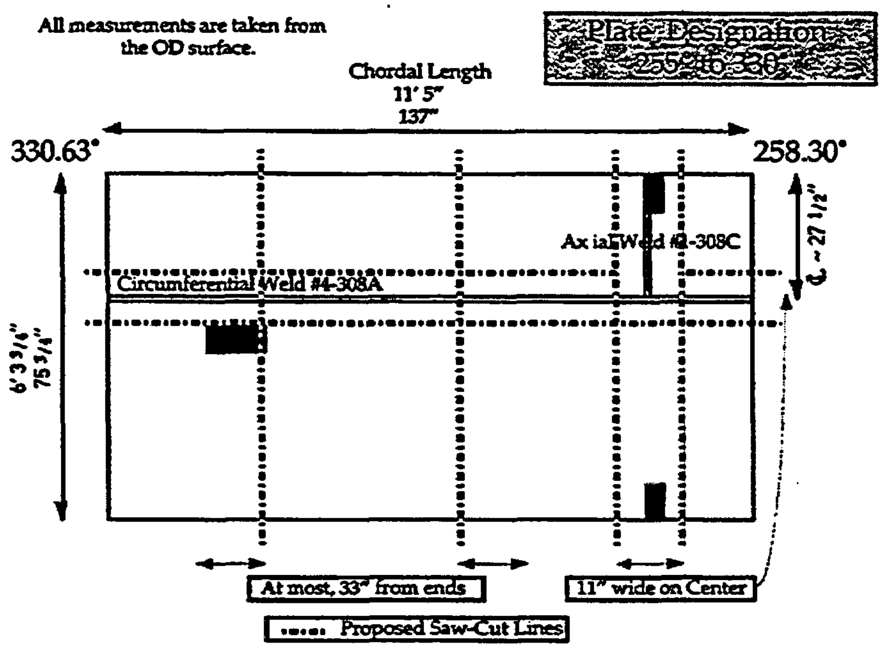

Figure 16. Proposed Sectioning of Plate 255° - 330°

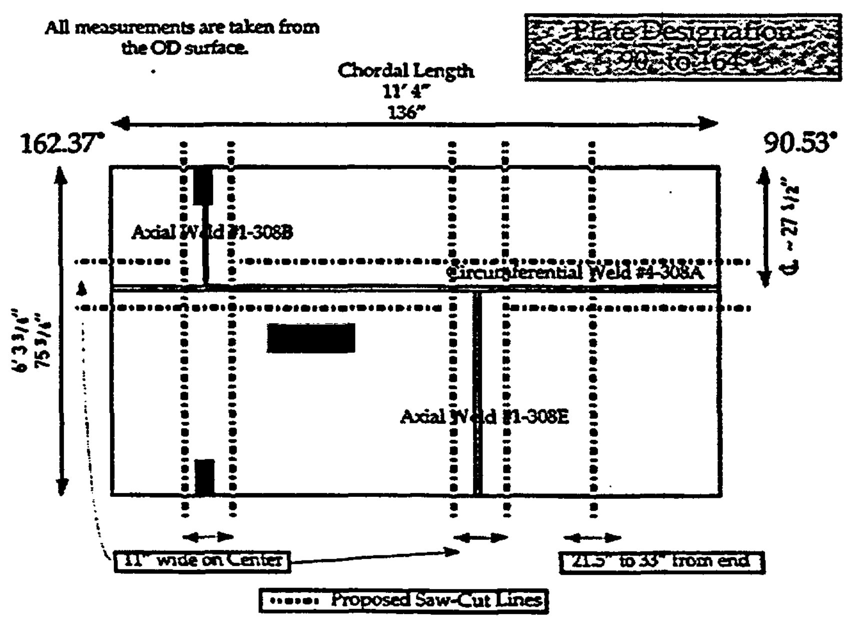

Figure 17. Proposed Sectioning of Plate 90° - 164°

The sectioned pieces will then be forwarded to the NRC. Additional and complementary inspections will be conducted by PNNL using the SAFT system. Inspections will be performed from the machined surface. The NRC will then use information derived from the EPRI and NRC inspection analyses for conducting the destructive examinations. Additional RT and UT examinations will be performed as needed to confirm flaw placement during the final destructive sectioning process.

The remaining three plates which only contain circumferential welds will be inspected and sectioned at a later date in a similar fashion.

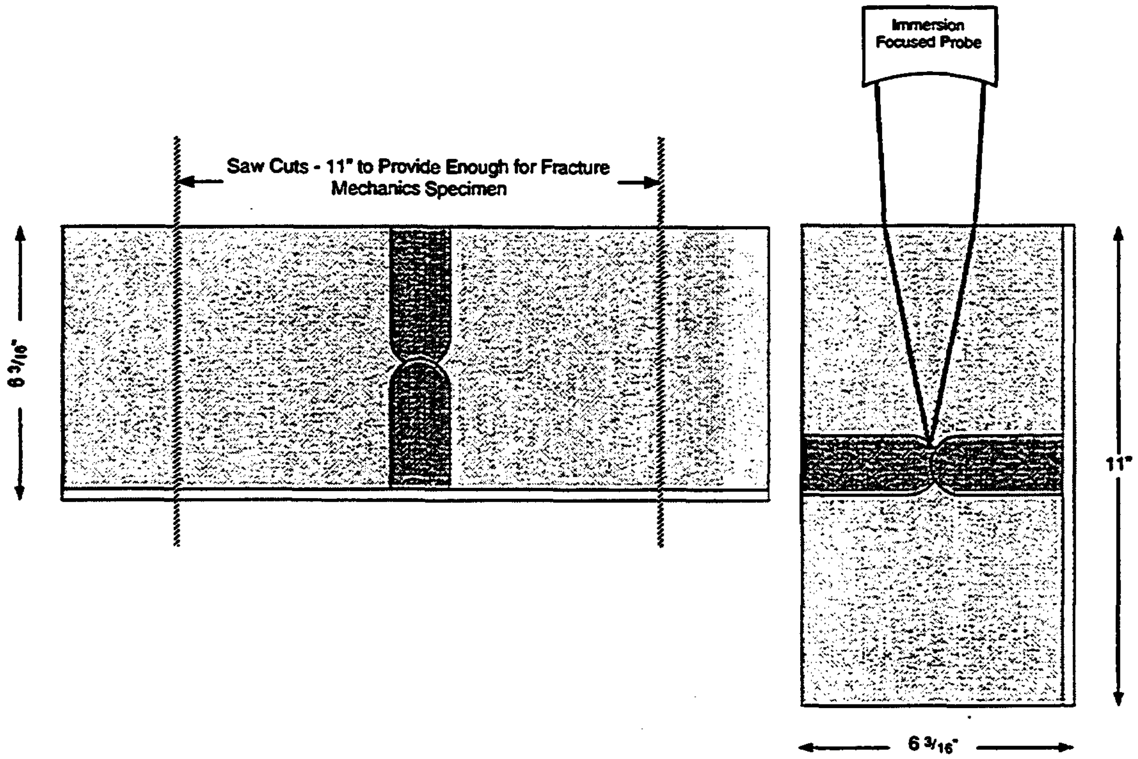

Figure 18. Proposed Fracture Mechanics Sectioning and Focused Immersion UT

Conclusions

A joint program between EPRI and the NRC has been established to investigate the distribution (size and number density) and characterization of post-fabrication defects in RPV weld materials. The program involves inspection and destructive analysis of material removed from the unirradiated Shoreham RPV. It is anticipated that results from the NDE and destructive analyses will provide information regarding ultrasonic inspection performance and reliability for consideration in RPV operating criteria and result in a more realistic assessment of RPV integrity.

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RESULTS OF REVISED (NUREG-1465) SOURCE TERM REBASELINING FOR OPERATING REACTORS

Presentation at the Water Reactor Safety Meeting

Jason H. Schaperow
Chester C. Gingrich
Office of Nuclear Regulatory Research

October 26, 1998

<u>Rebaselining</u>

Objective was to develop a better understanding of the impacts of implementing the revised source term for operating reactors.

- Effect on calculation of individual offsite and control room dose
- Effect on calculation of dose for equipment qualification
- Effect of potential plant modifications, including severe accident risk impacts

Rebaselining was an assessment of the likely dominant issues as revealed by analysis of representative plants and served as "test bed" for developing regulatory criteria

Rebaselining results provided in SECY-98-154

COMPARISON OF TID-14844 and NUREG-1465 SOURCE TERMS (PWR)

TID-14844	NUREG-1465
Instantaneous Release	Release over 1.8 Hrs
100% Noble Gases	100% Noble Gases
50% Iodines (with 50% Plateout)	40% Iodines
1% Solids	30% Cesium
	5% Tellurium
	2% Barium
	.02%-.2% Others

Iodine

TID-14844	NUREG-1465
91% inorganic vapor 4% organic vapor 5% aerosol	4.85% inorganic vapor .15% organic vapor 95% aerosol

Solids

TID-14844	NUREG-1465
Solids normally ignored for offsite dose calculation	Solids treated as aerosol

Rebaselining Tasks

Phase I	DBA Dose Calculations (SER modelling)
Phase II	DBA Dose Calculations (FSAR modelling)
Phase III	DBA Dose Calculations ("updated" fission product removal models) Thermal-Hydraulic Considerations Assessment of Margins Sump pH and Iodine Revolatilization
Phase IV	DBA Dose Calculations (plant changes) Risk Impacts of Plant Changes.

Phase I

DBA Dose Calculations (SER modelling)

Objectives were to determine the effect on individual dose of substituting NUREG-1465 for TID-14844 source term and the effect of the new dose acceptance criteria using the SER modelling.

Analyses performed for Grand Gulf and Surry.

- Accidents analyzed include LOCA, Fuel Handling, MSLB, Rod Drop
- Calculations performed for EAB, LPZ and control room

Phase II

DBA Dose Calculations (FSAR Modelling)

Objectives

Determine the effect on individual dose of substituting NUREG-1465 for TID-14844 source term and the effect of the new dose acceptance criteria utilizing the FSAR modelling.

Determine the effect on equipment qualification dose of substituting NUREG-1465 for TID-14844 source term.

Analyses performed for Grand Gulf and Surry.

Additional calculations performed for Zion, typical of a large, dry containment.

Grand Gulf LOCA - Doses (rem) Phase II

Source Term	EAB Thyroid	EAB Whole Body	EAB TEDE	LPZ Thyroid	LPZ Whole Body	LPZ TEDE
TID	23.1	9.5 11.5(1.0h)	NA	40.1	7.57	NA
NUREG	10.6 22.6(1.5h)*	1.5 5.9 (2.3h)	2.0 6.8 (2.2h)	19.5	4.05	4.73

* Worst two hours dose and start of worst two hours

Findings

Substantial difference between 1st 2 hr. dose and worst 2 hr. dose.

LPZ thyroid dose is reduced due to lower organic iodine and smaller (iodine) source term for ECCS leakage.

Major contribution to TEDE from noble gases.

Phase III

Objective: Assessment of detailed analysis assumptions and technical issues associated with implementation of the revised source term and evaluation of conservatism in regulatory analysis.

Tasks

1. Perform dose calculations for Surry, Grand Gulf, and Zion using NUREG-1465 and updated removal mechanism modeling: SRP models and models described in NUREG/CR-5966 (sprays), NUREG/CR-6153 (suppression pools).

2. Perform best estimate analysis using MELCOR for Surry, Grand Gulf, and Zion to assess margins in DBA LOCA analysis

Assumptions	EAB			LPZ		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
SER (TID-14844)	232	3.09	NA	20.3	.270	NA
FSAR (TID-14844)	225	3.36	NA	12	.15	NA
Updated Models (NUREG-1465)	76	.46	3.55	4.41	.021	.19
MELCOR (NUREG-1465)	1.55	.006	.055	1.16	.001	.037

Comparison of Phases I, II, and III Surry LOCA Doses (rem)

EQ Dose Analysis

EQ doses calculated for containment atmosphere and sump using TID and NUREG-1465 source terms

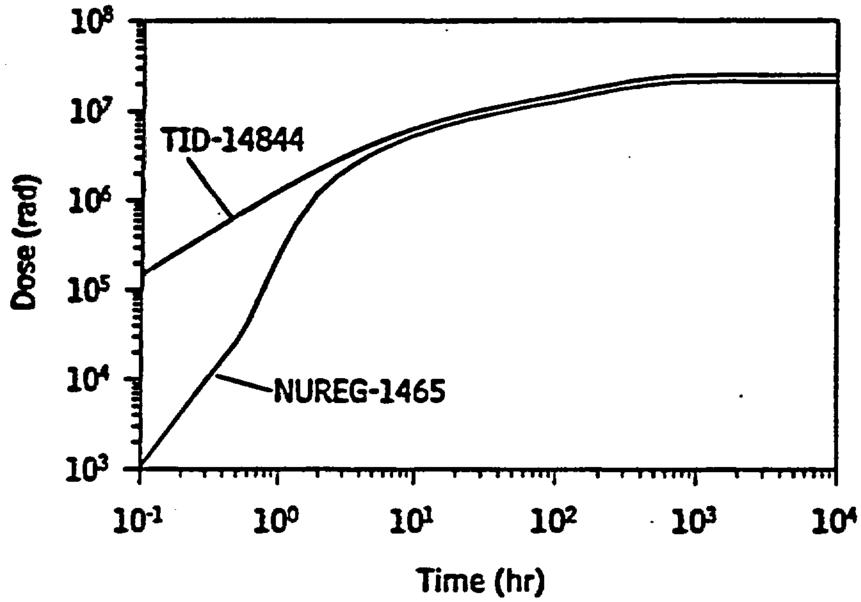
Gamma and beta doses in containment atmosphere

> Similar doses between TID and NUREG-1465 source terms, because the dose is from noble gases and iodine

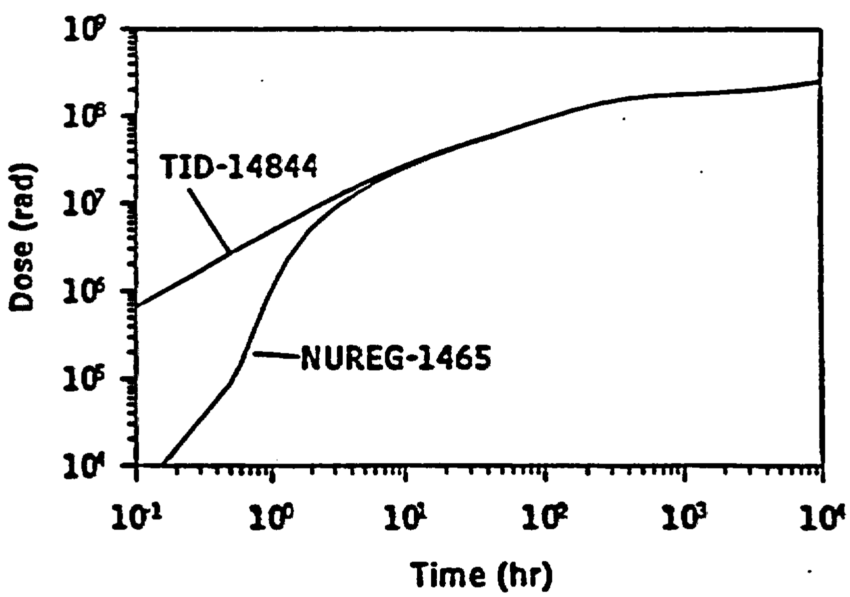
Gamma dose for equipment exposed to sump water

> Higher at later times for the NUREG-1465 source term, because of the large amount of cesium in the NUREG-1465 source term.
>
> TID-14844 includes 1% of the core inventory of cesium, NUREG-1465 includes 30% of the core inventory of cesium.

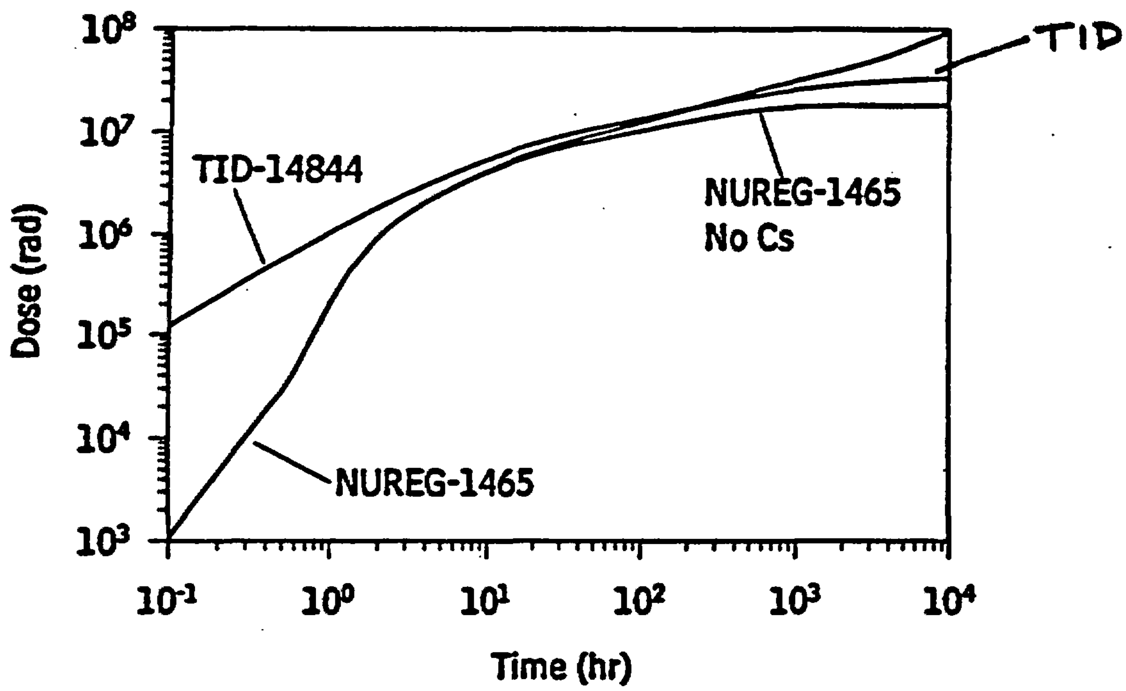
Surry Gamma Dose
At Containment Center



Surry Beta Dose
At Containment Center



Surry Sump Gamma Dose



Phase IV

DBA Dose Calculations (Plant Changes)

Performed offsite and control room dose calculations for LOCA for Surry, Zion, and Grand Gulf

Used FSAR plant models and updated removal mechanism models.

Changes considered to 1) MSIV leakage control system, 2) containment recirculation filters, 3) charcoal filters, 4) containment leak rate, 5) spray startup time, 6) enclosure building drawdown time, 7) changing from subatmospheric to atmospheric containment

Compared results to earlier results without plant changes and to current and proposed dose limits.

Elimination of MSIV Leakage Control System (LCS) and Increase in Allowable MSIV Leakage

Perry pilot plant proposed eliminating the MSIV leakage control system and increasing the MSIV leak rate from 100 to 250 scfh.

Assessed changes using Grand Gulf plant model.

Removal of the MSIV leakage control system resulted in doses less than the dose limits.

Some increase in the allowable MSIV leak rate was possible without exceeding the dose limits if credit for deposition in the main steam line is permitted. Not as much increase as proposed by Perry was acceptable.

Elimination of MSIV Leakage Control System and Increase in Allowable MSIV Leakage

Case	EAB			LPZ			Control Room		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
Phase III	24.4 (1.6h)	5.90 (2.3 h)	6.98 (2.2h)	19.9	4.06	4.76	4.06	.39	.54
No LCS	226 (.3h)	5.72 (2.5h)	14.5 (.5h)	54.4	4.22	6.82	9.34	.40	.85
No LCS, 2.2%/ day	2169 (.3h)	14.2 (.5h)	119 (.4h)	492	6.72	30.4	81.2	.57	4.49
No LCS, 2.2%/ day, steam line deposition	852 (.3h)	10 (.5h)	50 (.4h)	196	5.73	15.0	32.6	.51	2.05

Note: 2.2%/day corresponds roughly to 250 scfh
Based on Grand Gulf plant model.

Phase IV

Risk Impacts from Implementation of the Revised Source Term

Objective

The objective of the study was to evaluate the change in plant risk from modifying ESFs or their operation

Approach

Modify PRAs to model potential change in component performance for the various features including: 1) Containment Leak Rate, 2) Containment Spray Operation, 3) Reactor Building Drawdown Time, 4) Subatmospheric Containment at Atmospheric Pressure, and 5) Filtration Systems.

The NUREG-1150 Plants (Peach Bottom, Grand Gulf, Surry, Sequoyah, and Zion) and LaSalle were evaluated.

Results - Change in Plant Risk

1) *Containment Leak Rate*→Small (risk dominant sequences involve containment failure or containment bypass).

2) *Containment Spray Operation*→Small (delay of sprays until fission product release has no effect on risk and preserves stored water inventory)

3) *Reactor Building Drawdown Time*→Small (effect for sequences that do not fail containment <1% of total plant risk)

4) *Subatmospheric Containment at Atmospheric Pressure*→Small (major contributors to risk are accident sequences that involve containment bypass).

5) *Filtration Systems*→Small (filters generally not credited for risk dominant sequences).

Conclusions

Offsite and Control Room DBA Doses

Impact of NUREG-1465 vs TID-14844 is to generally produce lower calculated doses

Extent of the reduction is influenced by several factors

- Influence of safety features which are timing sensitive (e.g., SGTS, subatmospheric design)
- Analysis assumptions used in SER and FSAR calculations

Use of updated dose conversion factors will, by itself, produce lower calculated doses

Many of the types of plant changes being contemplated could be made and offsite and control room DBA doses would remain within acceptance limits.

Conclusions (2)

Equipment Qualification Doses

Similar doses for equipment exposed to containment atmosphere.

Higher doses later in time for equipment exposed to sump water, due to higher cesium inventory in NUREG-1465 source term.

Significance of higher sump water doses will be considered in the pilot plant reviews.

Conclusions (3)

Margin and Risk

Analysis performed with MELCOR severe accident code indicated that the offsite DBA doses still have substantial margin (a factor of 2 or greater) even though the dose may be well below the earlier TID analysis.

Potential plant changes being contemplated with the NUREG-1465 source term are not likely to have substantial risk impacts, because most of the systems being changed are not involved in risk significant sequences.

Implementation of NUREG-1465 Source Term

The staff did not identify any issues that would prevent implementation of the revised source term at operating reactors.

The rebaselining activities have provided a technical basis for rulemaking and the associated regulatory guides.

Recent MELCOR and VICTORIA Fission Product Research at the NRC*

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ABSTRACT

The MELCOR and VICTORIA severe accident analysis codes, which were developed at Sandia National Laboratories for the U. S. Nuclear Regulatory Commission, are designed to estimate fission product releases during nuclear reactor accidents in light water reactors. MELCOR is an integrated plant-assessment code that models the key phenomena in adequate detail for risk-assessment purposes. VICTORIA is a more specialized fission-product code that provides detailed modeling of chemical reactions and aerosol processes under the high-temperature conditions encountered in the reactor coolant system during a severe reactor accident. This paper focuses on recent enhancements and assessments of the two codes in the area of fission product chemistry modeling.

Recently, a model for iodine chemistry in aqueous pools in the containment building was incorporated into the MELCOR code. The model calculates dissolution of iodine into the pool and releases of organic and inorganic iodine vapors from the pool into the containment atmosphere. The main purpose of this model is to evaluate the effect of long-term revolatilization of dissolved iodine. Inputs to the model include dose rate in the pool, the amount of chloride-containing polymer, such as Hypalon, and the amount of buffering agents in the containment. Model predictions are compared against the Radioiodine Test Facility (RTF) experiments conducted by Atomic Energy of Canada Limited (AECL), specifically International Standard Problem 41.

Improvements to VICTORIA's chemical reactions models were implemented as a result of recommendations from a peer review of VICTORIA that was completed last year. Specifically, an option is now included to model aerosols and deposited fission products as three condensed phases in addition to the original option of a single condensed phase. The three-condensed-phase model results in somewhat higher predicted fission product volatilities than does the single-condensed-phase model. Modeling of UO_2 thermochemistry was also improved, and results in better prediction of vaporization of uranium from fuel, which can react with released fission products to affect their volatility. This model also improves the prediction of fission product release rates from fuel.

Finally, recent comparisons of MELCOR and VICTORIA with International Standard Problem 40 (STORM) data are presented. These comparisons focus on predicted thermophoretic deposition, which is the dominant deposition mechanism. Sensitivity studies were performed with the codes to examine experimental and modeling uncertainties.

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*This work was supported by the U. S. Nuclear Regulatory Commission and was performed at Sandia National Laboratories, which is a multiprogram laboratory operated by Sandia Corporation, a Lockheed Martin Company, for the U. S. Department of Energy under Contract DE-AC04-94AL85000.

1.0. Introduction

MELCOR [1] and VICTORIA [2] are computational tools that have been developed by Sandia National Laboratories for the U. S. Nuclear Regulatory Commission (USNRC). The two codes are similar in many respects but also have distinct differences. Both codes are intended for severe accident analysis of light water reactors and treat fission product release from fuel and transport through the reactor coolant system (RCS). However, MELCOR provides an integrated analysis while VICTORIA is more detailed but not fully integrated.

MELCOR includes models of thermal hydraulics, melt progression, fission product release from fuel, and fission product transport within the RCS and containment. These models are all fully integrated. VICTORIA, on the other hand, models fission product release and transport in a highly detailed fashion, melt progression in minimal detail, and thermal hydraulics not at all. MELCOR is primarily used to perform integrated analyses of reactor safety issues; VICTORIA is also used to perform analyses of reactor safety issues, but primarily to assess fission product behavior and to benchmark MELCOR.

2.0. Recent MELCOR Development, Testing, and Applications

Much has been accomplished since the release of MELCOR 1.8.4 in July 1997. In the area of model development, an iodine aqueous chemistry model has been developed and implemented. Also, the treatment of control and support structures has been improved. This latter accomplishment paves the way for future improvements in modeling of melt progression. In the area of testing, two International Standard Problems (ISPs) have been used to assess aerosol deposition models and the new aqueous chemistry model. These are described in this section and in Section 4. The primary USNRC application since the release of MELCOR 1.8.4 was to evaluate margin and containment thermal hydraulics for rebaselining with the revised source term [3,4].

The main function of the iodine aqueous chemistry model in MELCOR is to determine the partitioning of iodine within the reactor containment among an aqueous pool, the atmosphere, and surfaces. This partitioning is important because it affects potential fission product releases to the environment. The approach taken in developing this model is to use a relatively mechanistic treatment so that future refinements can easily be made. The model currently treats water radiolysis, atmosphere radiolysis (primarily formation of nitric and hydrochloric acids, which can acidify the aqueous pool), changes in pool pH, effects of buffering agents, mass transfer between pool and atmosphere and between atmosphere and surfaces, and formation of organic iodides. Iodine kinetics in the pool are assumed to be rapid (i.e., iodine species are assumed to be in equilibrium).

MELCOR was recently validated against International Standard Problem 41 (ISP-41), which is an iodine pool experiment conducted in the Atomic Energy of Canada, Ltd., (AECL) radioiodine test facility (RTF). The final report for ISP-41 is not yet available. The single test that was performed for this ISP was done in two phases, each with its own pH history. A schematic of the RTF is shown in Figure 1. The main vessel used in the ISP-41 test contained an aqueous iodine pool, an atmosphere, and surfaces on which iodine could deposit. An aqueous recirculation loop was used to keep the pool well mixed and to regulate pool pH. A gas recirculation loop was used to keep the atmosphere well mixed and to monitor H_2 production. An aqueous sampling loop was used to monitor pH and aqueous iodine concentrations. Participants in this ISP were provided with the test configuration, conditions such as temperature and total iodine content, and pH history. Participants were required to calculate the partitioning of iodine among the pool, atmosphere, and surfaces.

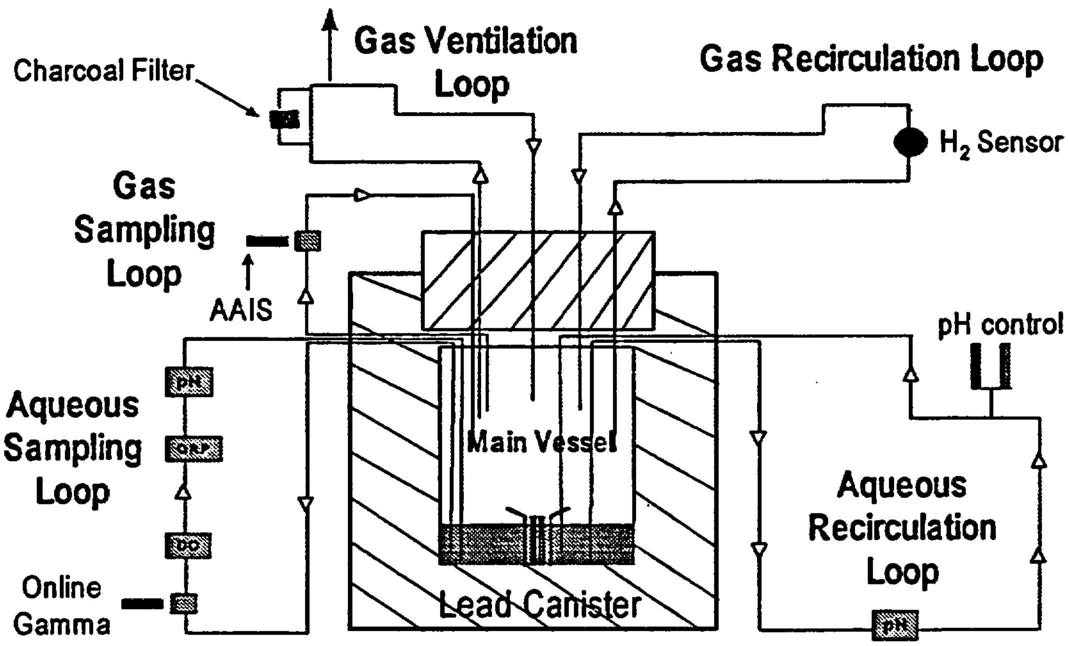

Figure 1. Schematic of the ISP-41 test configuration.

Phases 1 and 2 of ISP-41 were similar, so results for only one of these are shown here. Phase 2 is presented because pH was varied over a wider range and because the pH history is composed of a series of step functions separated by plateaus, as shown in Figure 2. Because of the simple nature of the pH history, the results are somewhat easier to describe than those for Phase 1.

MELCOR predictions for the number of moles of iodine on surfaces and for the concentrations of iodine in the atmosphere and pool are shown in Figures 3 through 5, respectively. A rapid decrease in pool pH results in a sudden increase in the concentration of iodine in the atmosphere because iodine is less soluble in the pool at lower pH, as seen by comparing Figures 2 and 4. This rapid change is an equilibrium chemistry effect. The sudden increase is followed by a gradual decrease in atmospheric iodine concentration. During the gradual decrease, iodine partitions out of the aqueous pool and onto surfaces. The time scale for the gradual decrease is controlled not by chemistry but by mass transfer. At the end of the test the pH is suddenly increased back to the initial value of about 10. This results in a very gradual redistribution of iodine from the surfaces back to the pool. The time scale for this process is also controlled by mass transfer.

Figures 3 through 5 do not show the experimental data, but the MELCOR predictions are within a factor of three of the data. While a factor of three sounds large, it is considered good agreement for this type of model and is as good as other participants in this ISP were able to obtain.

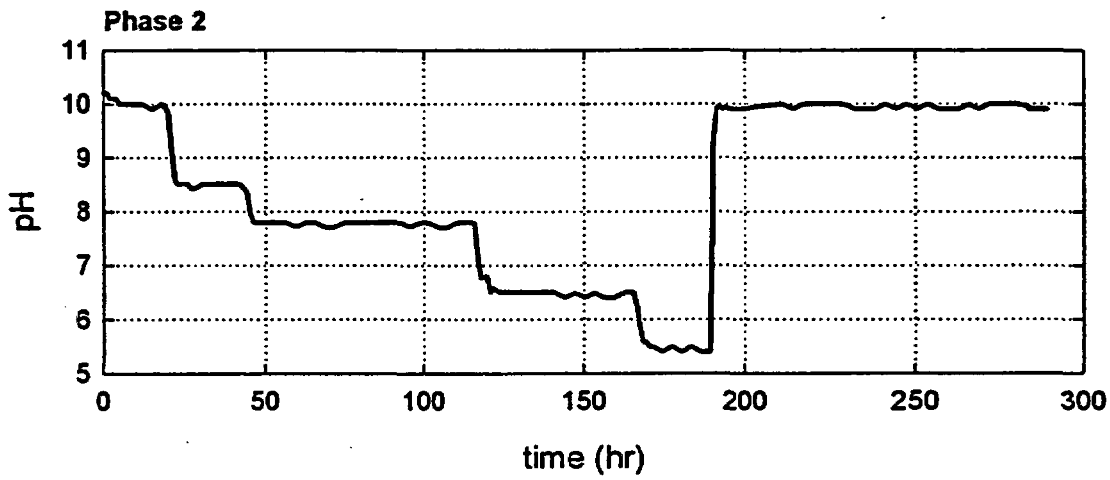

Figure 2. pH history for Phase 2 of the ISP-41 test.

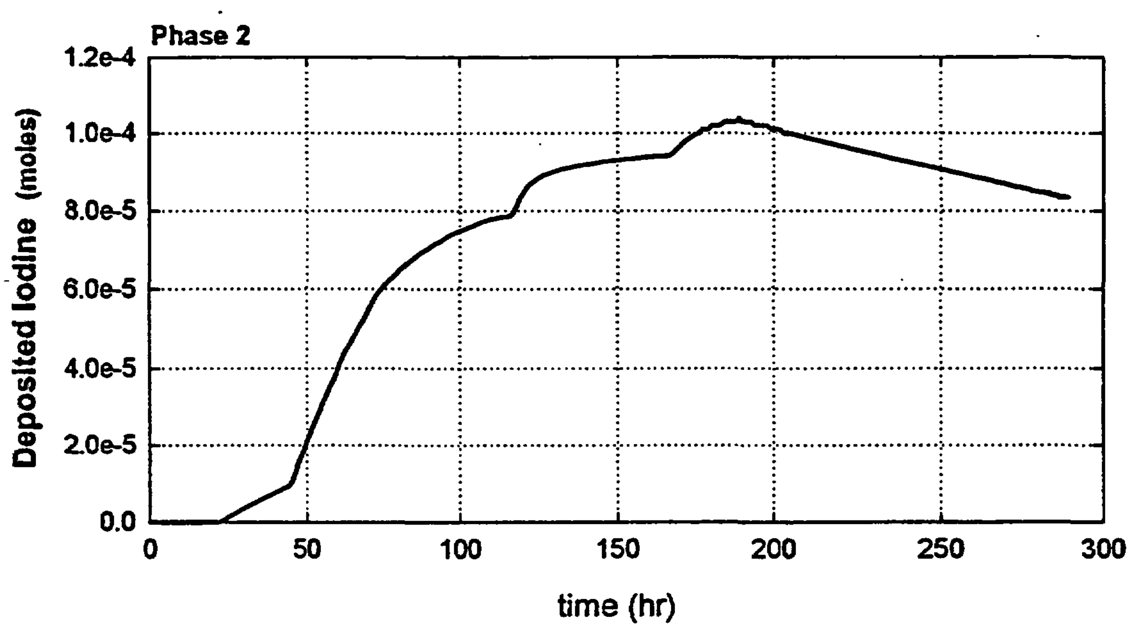

Figure 3. MELCOR-predicted moles of iodine on surfaces for Phase 2 of the ISP-41 test.

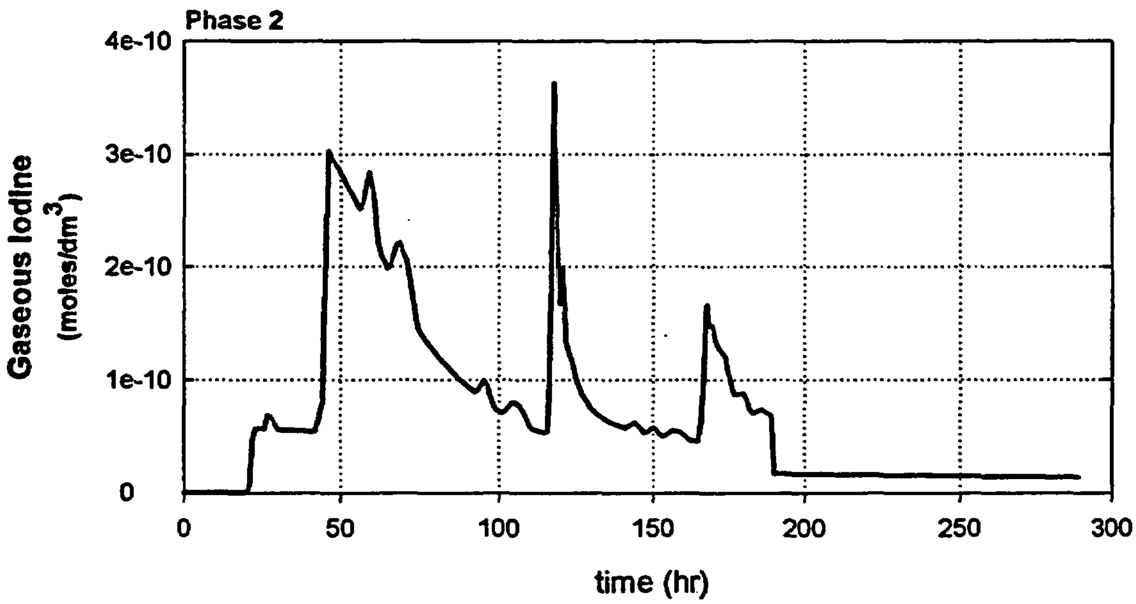

Figure 4. MELCOR-predicted gaseous iodine concentration for Phase 2 of the ISP-41 test.

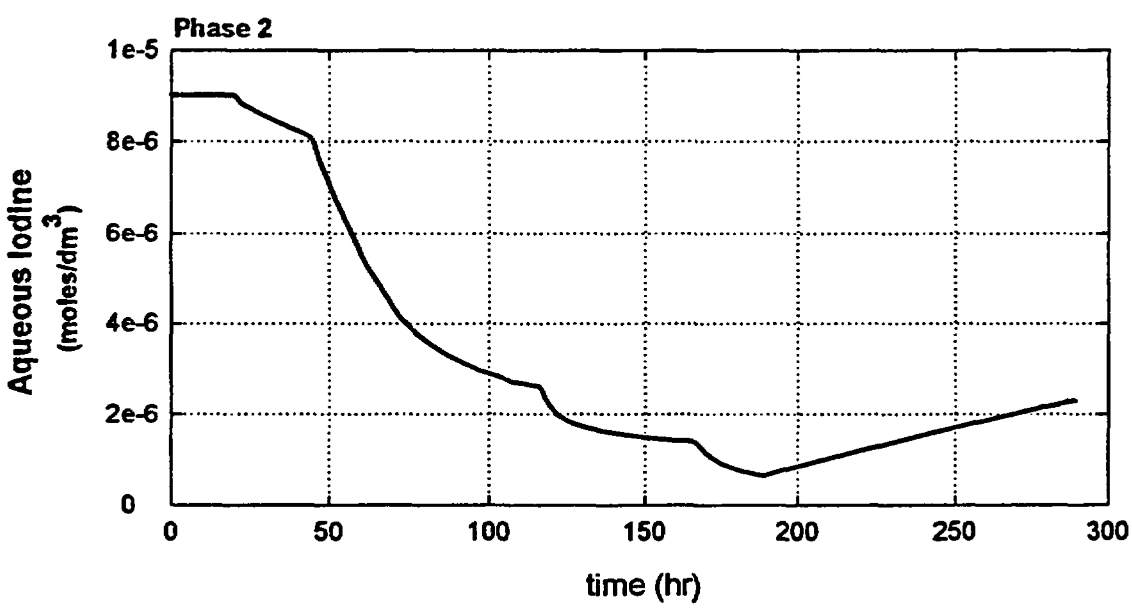

Figure 5. MELCOR-predicted aqueous iodine concentration for Phase 2 of the ISP-41 test.

3.0. Recent Improvements in VICTORIA

A peer review of the VICTORIA code was completed in 1997 [5]. Recommendations from the peer review committee were categorized as findings and high-, medium-, and low-priority concerns. To date, all of the findings and high-priority concerns have been addressed. Plans are to address the medium-priority concerns during 1999.

A new code version, VICTORIA 2.0 [1], has resulted from implementation of the peer review findings and high-priority concerns, as well as improvements that had been made prior to the peer review. The improvements are in two major areas: chemistry and user friendliness. Chemistry improvements include an option to treat three condensed phases as opposed to a single condensed phase and an improved treatment of fuel thermochemistry, which incorporates Blackburn's analysis of the thermochemistry of $UO_{2\pm x}$ [6]. Implementation of this latter model modifies predictions of releases of fission products from fuel and especially predictions of uranium release from fuel. At this point, Blackburn's model is implemented only for UO_{2+x}, where x is greater than 0. Improvements in user friendliness include warnings when thermal-hydraulic inputs have been chosen in an inconsistent manner and when the time-step size is larger than the Courant limit.

Other improvements in VICTORIA 2.0 include a treatment of fission product release from rubble beds based on the Booth approach, chemisorption models for HI and I_2, a simple model for chemical kinetics at low temperatures, a model for aerosol deposition in a vena contracta, simplified input of bulk gas flow rates, and a method for representing a domain (mathematical representation of a physical region) as coupled subdomains. That last improvement is especially useful for complex geometries and for sensitivity studies.

4.0. Comparisons of MELCOR and VICTORIA with ISP-40 Deposition Data

International Standard Problem 40 (ISP-40) was performed at the STORM facility in Ispra, Italy. This ISP test consisted of two phases: In the first, aerosols were deposited in a 5-m test section; in the second, part of the aerosols were mechanically resuspended. The ISP-40 test configuration is shown schematically in Figure 6. Aerosols and vapors were injected into a large chamber upstream of the test section. Flow through the test section was from left to right. Aerosol size distributions were measured upstream and downstream of the test section through sampling ports not shown in the schematic.

Conditions during the deposition phase of the ISP-40 test were nearly steady state. The test configuration, surface and gas temperatures, and mass injection rates were provided to participants of the ISP. Participants were required to calculate the final deposition profile for comparison with estimated data. Data for the deposition phase of the test were estimated because direct measurement would have precluded conduct of the second phase of the test in which the deposited aerosols were mechanically resuspended. Greater confidence should be placed on the overall quantity of aerosol that deposited (which was determined by mass balance) than on the deposition profile (which was estimated using results from previous tests).

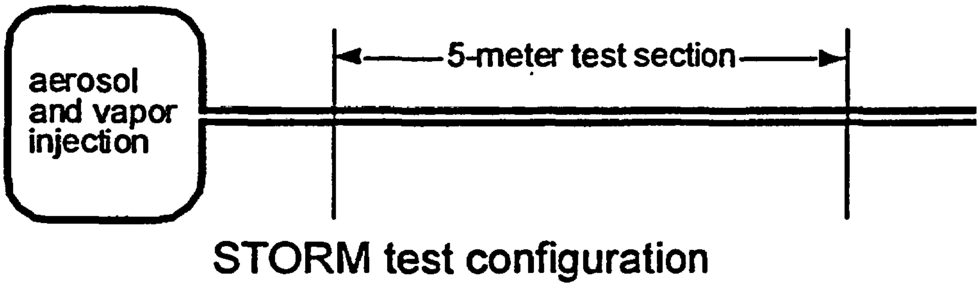

Figure 6. Schematic of the ISP-40 test configuration.

4.1. Modeling Considerations

Several considerations are central to modeling of aerosol deposition during an ISP-40 test. First, it is important to represent the aerosol size distribution well; this had a geometric mean of 0.43 μ and a standard deviation of 1.7. Furthermore, because only a small fraction of the aerosols deposit as they pass through the test section and agglomeration is relatively unimportant, the size distribution is nearly the same at the entrance and exit.

The second consideration is the appropriate calculation of the Reynolds number in each computational node. Many codes, including MELCOR and VICTORIA, compute the Reynolds number based on an average of inflow and outflow velocities for a node. If vapors are injected into the first computational node as a source, then the inflow velocity is zero and the Reynolds number in the first node is calculated to be one-half of the correct value. There are several ways to overcome this problem; the best is usually to create a dummy upstream node. If the Reynolds number is calculated incorrectly in the first node, then the predicted deposition profile will exhibit an unphysical depression at the upstream end of the test section.

All participants in ISP-40 determined that thermophoresis was the dominant deposition mechanism during the deposition phase of the test. This implies that the differences between gas and surface temperatures strongly influence predictions of the deposition profile. ISP-40 participants were provided with surface temperature data that were measured with thermocouples and gas temperature data that were calculated using a heat-transfer correlation. Some codes, like VICTORIA, were able to use the supplied data for surface and gas temperatures directly; other codes, like MELCOR, calculated their own gas temperatures given the surface temperatures. To match the ISP-supplied gas temperature profile, some participants modified the heat-transfer correlation used in MELCOR. However, modifying this correlation also modifies the calculated gas temperature gradient near the surface, which directly influences thermophoresis in the MELCOR treatment. This point is discussed further in the next subsection.

Finally, some codes, such as MELCOR, calculate the deposition and agglomeration integrals only at a few points in temperature and pressure, then interpolate to get values at intermediate temperatures and pressures. MELCOR calculates these integrals at four points representing two pressures and two temperatures. By default, the temperatures are 273 and 2000 K and the pressures are 1 and 200 bar. These broad ranges are appropriate for severe accidents but are not generally appropriate for small-scale tests like ISP-40. This point is also discussed further in the next subsection.

4.2. MELCOR Analyses

Three MELCOR calculations were performed, which represent a base case and two sensitivity cases. The first, or base, case was to specify the gas injection temperature according to the ISP recommended value but to let MELCOR calculate the gas temperature profile according to the default internal correlation for heat transfer. Adjusting the default heat-transfer correlation to better match the ISP-provided gas temperature profile was considered, but this idea was rejected because of the direct impact on thermophoretic deposition. Figure 7 compares the ISP-provided and the MELCOR-calculated gas temperature profiles. Because the MELCOR-calculated profile was somewhat lower than the ISP-provided profile, a second case was run with an inlet temperature that was 12 K higher than the recommended value. This resulted in about a 10% increase in the difference between gas and surface temperatures at the inlet. The MELCOR-calculated temperature profile for Case 2 is also shown in Figure 7. Case 3 was run with the same inlet temperature as Case 2, but used a narrower range of temperatures and pressures for calculating the agglomeration and deposition integrals: 550 and 650 K for temperature and 1 and 1.5 bar for pressure.

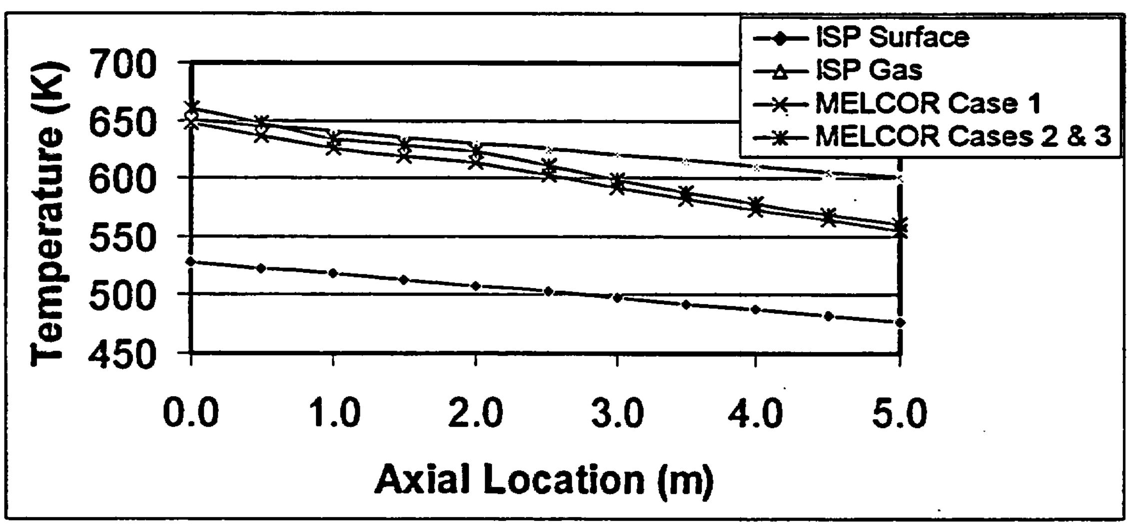

Figure 7. ISP-provided surface and gas temperature data and MELCOR-predicted gas temperatures for Case 1 and for Cases 2 and 3.

Figure 8 shows the ISP-estimated and MELCOR-predicted deposition profiles for the three cases. For all cases, the MELCOR predictions are in very good agreement with the estimated data. The higher gas inlet temperature used in Case 2 than in Case 1 results in a modest improvement in agreement with the data. The agreement between Case 3, which used a narrower range of temperatures and pressures for calculation of the agglomeration and deposition integrals, and the estimated data is excellent.

Predicted total masses deposited in the test section for Cases 1, 2, and 3 were 127, 137, and 163 g, respectively. The measured total deposition was 162 g [7]. This excellent agreement between predictions and data was achieved without any attempt to tune the standard correlations used in MELCOR. In fact, it appears that the MELCOR-calculated gas temperature profile may be a better representation of actual conditions than the one provided to the ISP participants. This point is discussed further in the next subsection.

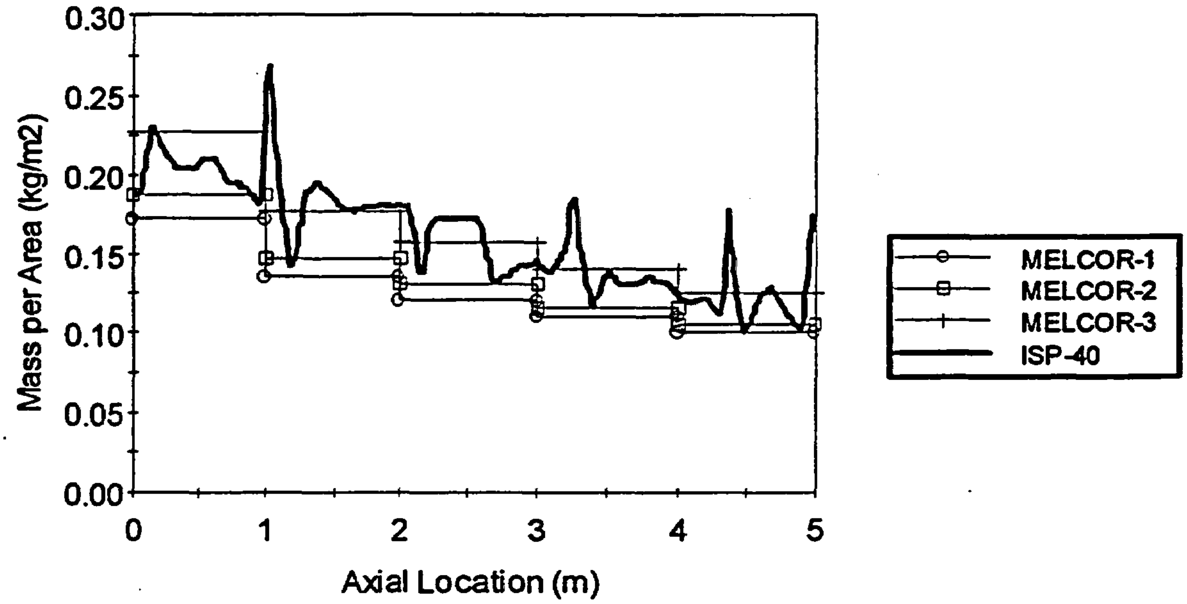

Figure 8. ISP-estimated deposition data and MELCOR-predicted deposition profiles for the three cases.

4.3. VICTORIA Analyses

Two VICTORIA cases are described. Other cases were run to demonstrate that predicted results were relatively independent of the number of nodes and of time-step size. It was determined that 5 nodes (plus an upstream dummy node) and a time step of 0.025 s were sufficient to give good predictions. With these choices, VICTORIA ran at approximately real time. (VICTORIA and MELCOR run times were similar.) The two cases, Cases 1 and 2, were performed with the ISP-supplied and the MELCOR-predicted (Cases 2 & 3) gas temperature profiles, respectively. As with the three MELCOR cases, the ISP-provided surface temperature data were used. Thus, the surface and gas temperatures used in VICTORIA are the ones shown in Figure 7.

Figure 9 shows the VICTORIA-predicted deposition profiles for the two cases. The equation used to represent thermophoretic deposition in VICTORIA is the same as the one used in MELCOR, namely the Brock equation [8]. However, VICTORIA uses as defaults the coefficients recommended by Talbot et al.

[9], while MELCOR uses coefficients similar to the original ones recommended by Brock. As seen by comparing Case 3 in Figure 8 and Case 2 in Figure 9, predicted thermophoretic deposition is greater using the Talbot et al. coefficients than using the Brock coefficients. The same trend is observed for other participants in the ISP who used the Talbot et al. coefficients in their analysis.

Many experts in aerosol science regard the Brock equation with the coefficients proposed by Talbot et al. as the best model available for thermophoretic deposition. It should not be argued based on this single comparison that the Talbot coefficients should be discarded in favor of the original coefficients proposed by Brock, because the preponderance of evidence goes the other way.

A second point to note from Figure 9 is that using the ISP-provided gas temperature profile (Case 1) results in nearly a flat deposition profile, which is because the temperature difference between the gas and the wall is nearly uniform. This result is not in agreement with the estimated trend, where mass per surface area decreases approximately linearly from inlet to outlet of the 5-m test section. The trend using the MELCOR-predicted temperature profile (Case 2) is in much better agreement with the estimated trend than that obtained using the ISP-provided profile. This indicates that the heat-transfer correlation in MELCOR is a better approximation than the one used in the ISP. This may explain, at least in part, why the deposition profiles predicted by ISP participants did not match the trend of the estimated data.

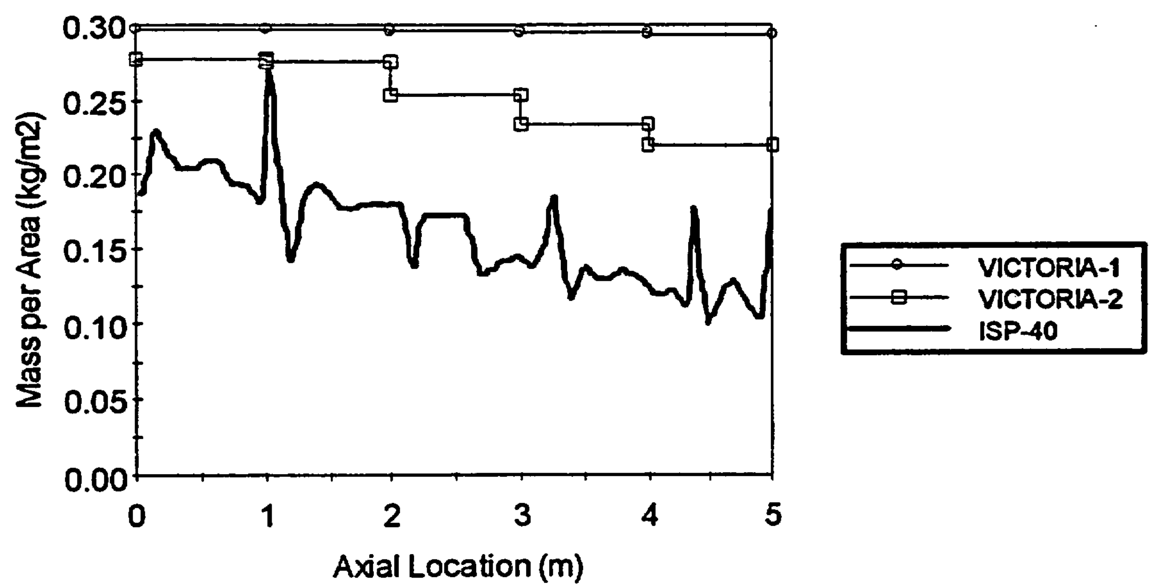

Figure 9. ISP-estimated deposition data and VICTORIA-predicted deposition profiles for the two cases.

The total deposited masses predicted by VICTORIA for Cases 1 and 2 were 293 and 249 g, respectively. These are both higher than the measured value of 162 g, but are consistent with other codes using the Brock/Talbot correlation. Furthermore, this level of agreement is considered by the authors to be acceptable.

5.0. Status and Future Directions

Development and application of both MELCOR and VICTORIA are continuing. Some of the ongoing and future activities are described in this section.

5.1. Status of MELCOR and VICTORIA

MELCOR model development continues in several areas, including reflood/quenching phenomena, core degradation modeling improvements, and iodine chemistry modeling. Currently, a comprehensive test matrix is being developed. The test matrix will include experimental data and International Standard Problems on core degradation, fission product release and transport, and containment phenomena. The matrix will also include plant applications. Future activities will include a strong emphasis on assessment analyses.

Peer review findings and high-priority concerns for VICTORIA have been addressed. The modified code version, VICTORIA 2.0, has been tested and is scheduled to be released, along with supporting documentation, by the end of 1998. Medium-priority concerns from the peer review will be addressed during 1999. Ongoing work with the VICTORIA code includes further applications to plant sequences, analyses of Phebus tests, and benchmarking of MELCOR. Experience with the VICTORIA code will be used as a basis for recommendations on improvements to the MELCOR code.

5.2. Future Directions for MELCOR and VICTORIA

There are several specific areas slated for model development for the MELCOR code: (1) improved modeling of failure of support structures, (2) improved geometrical representation of boiling water reactor (BWR) flow channels, (3) implementation of a BWR core spray model, (4) implementation of a core reflood model, (5) implementation of a passive autocatalytic recombiner model, (6) expanded control function capability, (7) upgrades in core degradation modeling, and (8) improved modeling of in-vessel/ RCS natural circulation phenomena.

Further work in the area of testing and assessment of MELCOR is also planned. This work will investigate accumulator water injection during a Surry 6-inch cold leg break sequence, evaluate natural circulation in the core for a Surry TMLB' sequence, and develop a comprehensive test matrix.

Future development of the VICTORIA code is being guided by recommendations made during the peer review [5]. The next phase of development will focus on the medium-priority concerns, which include the following: (1) revise the interpolation scheme for Gibbs free energy data; (2) add a treatment for hypostoichiometric fuel, i.e., for UO_{2-x}; (3) add carbon species to the thermochemical database so that boron control blades and rods can be modeled; (4) investigate the effects of porous media parameters that are used in modeling fission product release from fuel; and (5) modify the XMGR5 code to interface with VICTORIA graphics data to provide simplified postprocessing.

Several applications are planned for the VICTORIA code. These include evaluating the sensitivity of off-site releases to the level of detail of the chemistry modeling, benchmarking MELCOR, recommending improvements for the MELCOR code based on insights gained with VICTORIA, and analyzing some of the Phebus FPT series tests.

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THE PHEBUS F.P. INTERNATIONAL RESEARCH PROGRAM ON SEVERE ACCIDENT: STATUS AND MAIN FINDINGS

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ABSTRACT

The Phebus F.P. program is a wide international effort to investigate LWR severe accident phenomena, in particular core melt progression up to pool formation and the subsequent source term. The Program comprises 6 integral in-pile experiments, carried out in the Phebus nuclear facility operated by the French Institut de Protection et de Sûreté Nucléaire (IPSN).

Two tests have already been successfully performed, simulating a cold leg break under a low pressure, steam rich environment. They have provided experimental data of high interest, particularly on the hydrogen production rate and the core liquefaction temperature. Regarding the source term issue, the presence of significant amounts of gaseous iodine at the break and the fast trapping of iodine by silver in the sump of the simulated containment constitute certainly the most valuable observations for the analysts. The next two tests, expected in 1999, are being actively prepared. FPT-4 will provide data on fuel volatilisation from an irradiated debris bed and FPT-2 will investigate iodine volatility, under a steam poor environment, with boric acid and an alkaline sump. Possible interactions between catalytic H2 recombining surfaces and fission products will be also investigated in FPT-2.

INTRODUCTION

The international Phebus FP program [1] was launched in the late eighties by the French « Institut de Protection et de Sûreté Nucléaire » (IPSN) in collaboration with the European Commission and the French Utility (EDF). The ultimate objective was mainly to reduce the uncertainty on the evaluation of the amount and nature of radioactive products which could be released into the environment (Source Term) in case of a core melt-down accident in a Light Water Reactor Power Plant.

The international collaboration was rapidly extended to most of the countries which are using nuclear power: the USA (USNRC), Canada (COG), Japan (NUPEC and JAERI), South Korea (KAERI), and recently Switzerland (HSK and PSI).

The program involves a series of six integral experiments [2], using real fuel material and a scaled, well instrumented, primary circuit and containment models. Various thermal-hydraulics and physico-chemical conditions, typical of accident sequences, can be reproduced and the test fuel can be heated up to and beyond its melting point. The Phebus program does not intend to simulate typical accident scenarios but rather to reproduce well defined conditions to check that all the key phenomena controlling the transfer of radioactive products from the core into the reactor containment and their behaviour in the hours following the accident, are well understood and adequately modelled.

Two experiments were performed, FPT-0 in December 1993 and FPT-1 in July 1996. They were investigating low pressure highly oxidising conditions and involved about 10 kg of respectively trace irradiated (FPT-0) and irradiated (FPT-1) fuel. One of the major objectives was to investigate the behaviour of radioactive iodine under such conditions, as it constitutes a potentially high risk for health and the uncertainty on its volatility in the containment is large. The evaluation of the experimental data is now completed for FPT-0, whereas it is still in progress for FPT-1 at the time of writing.

Meanwhile, the preparation on the site of the next tests, FPT-4 and FPT-2, is very active. They are indeed currently planned to be performed in 1999, within a few months of each other. Discussions between safety experts around the definition of the last experiments, FPT-3 and FPT-5, are also progressing . Feasibility studies are expected to start by the end of this year.

THE PHEBUS FACILITY AND INSTRUMENTATION

As seen on Figure 1, the experimental circuit comprises three main sections: an in-pile section, a circuit simulating the main elements of the primary circuit of a LWR and a vessel simulating a reactor containment building.

In-Pile Section:

The reactor core is represented by a 20-rod, 1 m high, test fuel bundle surrounded by a ceramic shroud fitted inside a pressure tube. In FPT-0 and FPT-1, the central position is occupied by a silver-indium-cadmium rod with stainless steel cladding, inside a Zircaloy guide tube. This rod is intended to simulate the control rod system present in a typical 900 MWe PWR. Two Zircaloy grids are located at 220 and 740 mm from the bottom of the fuel column. The test package is inserted into a pressurised water loop, located at the centre of the 40 MW Phebus reactor core.

Prior to the experimental heat-up sequence, the test bundle is re-irradiated *in situ* during a few days of operation at nominal power (an average of 15 kW/m of rod) , cooled by a forced flow of pressurised water. The objective is to re-create in the test fuel the inventory in short-lived fission products as I131, for on-line gamma spectroscopy, and in order to induce radiation doses large enough to observe radio-chemistry effects in the containment.

A remotely operated foot valve below the fuel bundle is closed for the high-temperature transient phase. Steam is then injected at the base of the test bundle under a system pressure of 0.2 MPa.

The test fuel was 4.5% enriched fresh UO_2 for the first test. The irradiated fuel used for FPT-1 originates from the BR3 plant in Mol, Belgium. The mean burn-up was of the order of 23 GWd/tU.

Simulated Reactor Cooling System:

The upper plenum above the bundle is connected to an horizontal line made of Inconel and trace heated to 970 K. It conveys the gases and aerosol getting out of the test bundle to a U tube simulating a PWR steam generator. In FPT-0 and FPT-1, the wall of the steam generator model was maintained at 423 K, i.e. above the saturation condition of steam for the considered pressure. The outlet of the U tube is then connected to the containment model, thus simulating a cold leg break. The volume scaling ratio is close to 1/5000 as compared to a 900 MWe PWR, that is close to the overall scaling ratio for the core region.

Apart from the deposition of a low amount of FPs by chemisorption along the hot line, the main depositions are expected in the U tube, mainly by vapour condensation and aerosol thermophoresis.

Containment model:

A 10-m^3 cylindrical vessel collects the aerosol, vapour and steam/hydrogen effluents conveyed by the circuit during the test, simulating a reactor containment building in presence of a break in the primary circuit.

Particular design features of the containment vessel are a sump at the bottom and a group of three condensers in the upper part, which are designed to control steam condensation and which thus simulate the cold structures of a reactor building. The outer vessel wall is heated to avoid steam condensation and subsequent aerosol deposition, which would be out of scale and could not be quantified accurately.

Painted surfaces on the condensers and in the sump enable the investigation of organic iodine formation.

The overall scaling factor for the containment volume, the sump and the cooled structure areas is of the order of 1/5000 as compared to a 900 MWe Nuclear Plant.

Experimental circuit and containment model are located inside a 300 m^3 stainless steel housing, called the « FP caisson » which constitutes the first radiological barrier.

Instrumentation and diagnostic tools for core degradation observation:

The test train is instrumented with about 70 thermocouples, two ultrasonic thermometers, miniature fission chambers and a differential pressure transducer. The injected steam is determined by weighing and the hydrogen production is determined by means of a hydrogen sensor located in the circuit and the measurement of its partial pressure in the containment.

After destruction of the high temperature thermocouples located in the bundle (W/Re TC) above 2300 K, the heat-up of the bundle and the relocation of the fuel is mainly determined by means of the thermocouples located radially inside and outside of the cooler insulating layers surrounding the bundle.

After the experiment, the test train is removed from the reactor cavity and examined in a shielded cell. Non destructive examinations involved radiographs and transmission tomograms which are performed using a linear electron accelerator. The use of a digital camera makes it possible to treat the large number of information collected and to obtain high quality pictures with false colour representation of the different material densities. 400 tomograms were produced after FPT-1 with a spatial resolution of 0.5 mm. The numerical treatment of this data has provided information on the material redistribution, the extent of flow area blockages and the porosity of these blockages.

The test train is then cut in a hot cell laboratory and further examined (Macrographs, Electron Probe Micro Analysis) in order to investigate into material interactions during the core melt-down process. Melting point of the obtained corium is also measured.

Fission product and structure material measurements:

The evaluation of the amount of fission products and structure material products still present after the test in the fuel region or deposited in the vertical channel is obtained post-test by applying several techniques.

First, the association of γ emission tomograms, giving the precise 3-D location of the various sources, and of transmission tomograms, indicating the actual geometry, enables to derive for each specific γ emitter an attenuation coefficient. The axial γ scan can then be interpreted quantitatively with, in addition, indication of the fraction of fission products which is still in the fuel. The γ emission tomogram technique was only available at the time of FPT-1.

In addition, various instruments are used along the experimental circuit and around the containment model which give information on the masses of fission products and structure material transported and deposited . These instruments are located (see Figure 2) :

- at inlet of steam generator simulator (point C)
- along steam generator,
- at outlet of steam generator (point G),
- in the containment atmosphere (point H),
- in the sump.

They consist in:

- on-line γ spectrometers equipped with He cooled germanium diodes and automatically adjustable collimator windows, allowing to measure the activities of the numerous radionuclides released by the fuel; three are located along the circuit (inlet of SG, along the SG and outlet of the SG), one measures the activity of the containment atmosphere as well as of the deposits on the wall and condensers, whereas another one measures the activity of the water in the sump ; the latter was modified prior to FPT-1 to measure also the activity of the deposits on the bottom of the sump;
- sampling instruments adapted to, or developed for, Phebus FP, as inertial impactors, filters and sedimentation coupons for aerosols analysis, capsules for gas or liquid sampling, and selective iodine speciation samplers, called May-packs; iodine aerosols are retained in the quartz filter, molecular iodine is chemisorbed on the "knitmesh" filters Ag-plated Cu mesh and organic iodine on the zeolite filters; the different filter stages of one of this instrument were on-line γ scanned during FPT-1 as was a gas capsule equipped with zeolite;
- on-line photometer determining the concentration in aerosol transported in the circuit (OLAM at point C) or in suspension in the containment atmosphere;
- thermal gradient tubes; the different fission product vapours are deposited selectively at different locations along the axially imposed temperature gradient, depending on their condensation points.

After the test, the condensers are γ scanned and the inner faces of the containment wall are inspected using a photomultiplier in order to measure the axial distribution of the deposits.

All sampling instruments and sections of the circuit main component are recovered by remote handling as soon as the experimental installation is back to atmospheric pressure and room temperature. They are transferred to a hot cell under the « FP caisson » where first inspections and gamma scans are carried out, beginning with those samplers which have to be scanned for iodine 131 analysis.

After a first selection, specimens (filter stages, impactor plates, sub-section of circuit wall, ...) are shipped to a number of laboratories for Post-Test Analysis (PTA) with the following objectives :

- An overall fission product and structure material mass balance.
- Determination of elemental and isotopic composition, the solubility and the chemical speciation of the samples.
- Determination of aerosol granulometry and morphology in solid deposits.

The PTA plan involves the use of complementary radio-chemical techniques as γ scans of the as received samples for comparison with Phebus references, scanning electron microscopy, associated with energy-dispersive or wave length X-ray spectroscopy, X-ray diffraction and fluorescence, inductively coupled plasma optical emission and mass spectroscopy, electron microprobe analysis, and wet radiochemistry.

FPT-0 AND FPT-1 TEST CONDITIONS

The first test of the programme, FPT-0, using fresh fuel, was performed from December 2 to 6, 1993. The second, FPT-1, with irradiated fuel, from July 26 to 30, 1996. Both experiments were operated under similar thermal-hydraulic conditions, the main difference between them being the burn-up of the test fuel: trace irradiation in FPT-0 and 23 GWd/tU in FPT-1. Whereas the short life fission product inventories were about the same in both tests - similar durations of irradiation just prior to the test -, the overall fission product inventory was about 30 to 50 times larger in FPT-1, depending of the element, owing to the presence of long life isotopes.

The main objective was to investigate bundle degradation, fission product release and transport under low pressure (0.2 MPa) and oxidising (steam-rich) conditions. Initially, it was intended to reach the melting point of fuel and to obtain a melt fraction of roughly 20% of the bundle fuel inventory , i.e. 2 kg of liquid UO_2.

Boundary conditions

The bundle boundary conditions were set up after years of pre-calculations in order to reach the assigned objectives. Steam flow was varied from 3g /s (2g/s in FPT-1) during the cladding oxidation phase down to 1.5g/s during the high temperature phase, whereas the Phebus driver core power was increased stepwise, up to reaching the requested fuel melt fraction. The corresponding estimated bundle power was of 90 kW. The rather high steam flow rate during the cladding oxidation phase was chosen to ensure that the absence of steam starvation ($[H_2]/([H_2O]+[H_2])<1$).

The main component in both circuits was a steam generator regulated at 150°C to avoid any steam condensation under the pressure conditions of the test. The line connecting the outlet of the bundle to the inlet of the steam generator was heated to 700°C.

During the degradation phase, the containment vessel boundary conditions were adjusted in order to limit the relative humidity ratio to about 70 - 80%, avoiding uncontrolled steam condensation on the outer walls which could force fission product deposition on undesirable non instrumented spots. The condensers were cooled at 74°C, the outer wall heated to 110°C whereas the sump was at 90°C, colder than the overlying atmosphere. The 100 liters of sump water were at pH 5, buffered by a mixture of boric acid and soda, in order to maximise iodine re-volatilisation by radiolytic effects.

After the degradation phase was completed, wall and condenser temperatures were changed (see table 1), first to allow the washing of the vessel bottom without water vaporisation and then in order to promote the homogenisation of the containment atmosphere by free convection during the 4 days of containment chemistry observation.

Table 1 : Containment boundary conditions during FPT-0

Phases	Objectives	Outer wall T (°C)	Condenser T (°C)	Sump T (°C)
Bundle degradation (5 hours)	Fission product and structure material release	110	74	90
SCRAM (18138 s) - Containment isolation (20258 s)				
Aerosol phase (19 hours)	Aerosol deposition on walls, condensers, sump	110	74 to 110	90
Washing (6 hours of temperature transient + 15 minutes of rinsing)	Entrainment to the sump of aerosols deposited on bottom of vessel	110 to 100	40	40
Chemistry phase (4 days)	Long term iodine behaviour	130	110	90

Test scenarios

After an irradiation phase of about 9 days (FPT-0) and 6 days (FPT-1) at nominal power, the bundle degradation transient was performed in four main phases (see Figure 3.):

- A thermal calibration phase of 10000s (11000s in FPT-1) with three different steady state bundle temperature levels (730K, 870K and 1200K).

- A cladding oxidation phase which lasted less than 2000s and was followed by a power plateau for instrumentation check-out.

- A final heat-up phase of 5000s (3000s in FPT-1) ; fuel relocations occurred and resulted in the achievement of pre-set shroud temperature criteria requesting to terminate the bundle transient by manual reactor shutdown; it is worth noting that the power level reached at that time was, for both experiments, well below the level at which fuel melting was expected to take place.

- A bundle cooling-down phase which followed the reactor shutdown during which the steam flow rate was maintained for continued fission product and aerosol transport through the circuit to the containment; during this phase aerosols were being deposited on the surfaces and on the bottom of the containment

The total duration of the degradation and the fission product release phase of the two tests was about 5 hours. After isolation of the circuit, the experiments continued for four days during which long term chemistry effects were investigated in the vessel simulating the containment.

MAIN FINDINGS

Core degradation

In both experiments, the bundle degradation transient was characterised by the following successive events (see Figure 3) :

- failure of the fuel rod claddings at the expected temperature of 800°C,
- failure of the control rod around 1200°C,
- oxidation of the zircaloy claddings with large production of heat and hydrogen,
- material relocation and formation of a molten pool in the vicinity of the lower grid for a bundle power of the order of 35 kW.

The steam-zircaloy reaction started in the upper part of the bundle at a temperature of 1550°C and propagated first upwards and then downwards. Temperatures as high as 2600°C were measured in FPT-0 around the upper grid, with temperature ramps of the order of 10°C/s.

A large flow of hydrogen was produced (a few tenths of g/s) but, as planned, steam starvation was never reached at the outlet of the bundle. It is estimated that roughly 70% of the available zircaloy oxidised within the few minutes this phase lasted. Hydrogen was also produced later on during the transient, when fuel relocated and heated up the lower part of the bundle.

At the time of FPT-0, most codes underpredicted the amount of hydrogen produced during the exothermic reaction of fuel rod cladding oxidation by steam, some of them by nearly a factor 2. The temperature level reached during this event was also largely underestimated.

The main reason was identified as being the tendency for degradation codes to predict that cladding dislocated and relocated early during the transient, thereby stopping the reaction. Actually, cladding appeared to have remained in place up to complete oxidation. The explanation is certainly to be found in the fact that the experimental basis, which has supported the development of the dislocation models, was poor in tests under the steam rich conditions prevailing in the first Phebus experiments, although corresponding also to some severe accident situations. When transposed to reactor situation, the amount by which codes underestimate hydrogen production is less than a factor 2, owing to the large size of cores and thus the non homogeneous temperature distribution. Nevertheless, as a lesson learnt from FPT-0, the codes have been corrected and the predictions for FPT-1 were in good agreement with the measurements.

Material relocation events inside the bundle occurred shortly after heat-up was resumed and were detected by thermal and aerosol instrumentation signals. Post-test analysis using the ICARE 2 computer code [2] have led to the conclusion that in FPT-0, the fuel rods began to liquefy at a temperature of 2230°C, i.e. some 300°C below the minimum melting point of a UO_2-ZrO_2 mixture.

In the case of the FPT-1, the additional information from ultrasonic thermometers (not working in FPT-0) and preliminary calculations using ICARE 2 confirm that fuel rods began also to liquefy around 2200°C.

Post-test non destructive and destructive examinations indicated a very advanced stage of degradation (see Figure 4). Numerical treatments of FPT-0 and FPT-1 tomograms indicate that only 50% (FPT-0) to 70% (FPT-1) of the fuel remained in the shape of pellets or solid fragments. Between 2 and 3 kg of once liquid material accumulated on and below the lower grid in FPT-0, slightly less in FPT-1.

Destructive examinations of FPT-0 have indicated that the molten pool area mainly consists of $(U,Zr)O_2$ corium with 1 to 2% of iron and chromium oxides. An attempt to measure the melting point of the resulting corium was made by the Transuranium Institute JRC, using a calibrated laser flash technique. Three measurements were performed, indicating a liquidus - solidus transition of 2300°C within a few tens of °C, consistent with the rod liquefaction temperature deduced from the thermal analysis. This measurement is currently under evaluation.

Such a low melting point can explain why the melt fraction objective of 20% was achieved in both test with a lower bundle power than anticipated. The current understanding for this large difference is the combination of the main following factors:

- increase of fuel stoechiometry in steam rich environment,
- presence of zircaloy rich mixtures which can dissolve fuel, even when totally or partially oxidised,
- effect of iron and chromium oxides, originating from the control rod cladding and the springs in the fuel element; iron and chromium oxides significantly lower the melting point of the zircaloy - uranium oxides as already observed by P. Hofmann [3],
- inclusions of fission products in the fuel matrix.

The FPT-1 post test destructive examinations are in progress. From the tomograms taken after the test, it appears that for the same conditions of temperature, the fresh fuel pellets, of the two instrumented rods, were substantially less degraded than the irradiated fuel pellets. It suggests that burn-up can increase the kinetics of core melt progression, probably due to the fact that volatile fission products make irradiated fuel porous (fuel swelling) and zircaloy rich melt can then easily penetrate the fuel by capillarity, inducing grain separation and bulk dissolution (fuel foaming).

Thirteen organisations (AEA-T, CEA-DRN, CSN, ENEA, FZK, GRS, IPSN, JRC, KAERI, KEMA, NUPEC, SNL, UPM) using five different codes (ATHLET-CD, ICARE 2, MAAP-4, MELCOR, SCDAP/RELAP5) have analysed FPT-0 and try to re-calculate the degradation pattern observed.

Fission product and structure material behaviour

The analysis of the FPT-0 fission product measurements has demanded a tremendous amount of manpower and skills : analysis of 40 000 γ spectra, consistency studies between measurements and chemical analyses, as well as uncertainty evaluation. Because of the sampling nature of these measurements, extrapolations were necessary to obtain the values of concentrations in the steam along the circuit or of amounts of deposited materials in the steam generator and on the containment surfaces.

Thanks to the number of isotopes detected by γ spectrometry, mass balances have been closed for most of the elements, in the circuit, the containment atmosphere and the sump, using a multiple linear regression technique [4]. The estimated uncertainty is of the order of 20% for those radionuclides which are γ emitters and 30% for the others. Mass balances were closed within better than 10% for most of the relevant elements as shown on Figure 5.

i) Release from fuel region

The overall fractional releases of fission products and structure aerosols are given in table 2. In the case of FPT-0, they result from the integration of the measurements given by the samplings performed at the inlet of the steam generator and thus are lower bounds since they do not account for the deposits along the test channel between the bundle outlet and the steam generator inlet.

In the case of FPT-1, the values are deduced from γ scanning evaluation using the results of the tomograms and have to be consolidated by the results of the samplings. The orders of magnitude are quite comparable to those obtained in FPT-0, with a tendency to lower values, which can be explained by the less advanced stage of degradation achieved in the second test.

Table 2 : Overall fractional FP and structure material release

Elements	Measured in FPT-0 (% bundle inventory)	First estimate in FPT-1 (% bundle inventory)
I,Te (volatiles)	70 to 100	>50
Cs (volatiles)	50 to 70	>50
Ru (semi volatiles)	2	<6
Ba(semi-volatiles)	1	< 6
Ag (control rod)	10 to 15	not yet available
U (fuel)	0.1	not yet available

The measured values are not very different from what was expected, with maybe the exception of Ba, for which release of the order of 10% to 40% was expected from out of pile studies at ORNL (HI-VI) and Grenoble (VERCORS)[5]. This point is not yet quite elucidated. In the particular case of ruthenium, FPT-1 γ emission tomograms show that most of the Ru103 was released by the fuel but was then redeposited according to rather complex patterns, most of the released Ru being redeposited in the upper part of the bundle.

Despite a low fractional release, the mass of aerosol originating from the structure materials (control rod alloy, cladding, fuel) and released by the core is of the order of 170 g for FPT-0. This figure is of course to be consolidated by the results of FPT-1, still under evaluation.

One interesting result from FPT-0 is that the rate of fission product release is significantly reduced when the pool forms (see Figure 6). This can be correlated to the fact that a pool offers a more compact geometry but is not yet modelled in the computer codes. The net result would be reduced emission rates in semi-volatiles and low volatiles with possibly a larger decay heat in the pool. However, complementary data from coming Phebus-FP tests and more refined evaluation are of course necessary prior to confirming this trend.

ii) Transport and deposition in the circuit

When analysing the deposits on the filters sampled at the inlet of the FPT-0 steam generator, it becomes clear that all the fission products except iodine, and all the structure material, except cadmium, were in a condensed form at 700°C. This implies that caesium was not transported as the expected CsOH species but was present as a less volatile forms which is not yet identified. A candidate would be Cs_2MoO_4 according to thermodynamic calculations. Iodine was vapour as predicted and the position of its deposition peak in the thermal gradient tube, which was sampled just after scram, corresponds to the temperature range of 220 to 430°C. This is consistent with an iodine species of moderate volatility such as CsI, RbI or even AgI.

From impactor data, the aerosol mean mass aerodynamic diameter for all the nuclides, except iodine and cadmium, is rather small and of the same order of magnitude from 0.5 to 1.0 µm, with however some uncertainty due to the fact that sampling flows fluctuated. The aerosols are essentially composed of structure materials: 30 to 40% Ag, 20% Re, 13% Sn, 7% Ni, 6% Cd and U. Rhenium results from the degradation of the thermal instrumentation located in the bundle. From the investigations performed so far [6], it is concluded [7] that Re did not play a significant role on the fission product speciation and transport in the circuit. In particular, it could not explain why Cs was already condensed at 700°C in the FPT-0 conditions.

The deposition of aerosol in the steam generator, essentially by thermophoresis, was quite low (see Figures 5 and 7): 14.5 ± 1.5% of the mass which has entered. This is about twice as low as predicted. No firm explanation has been found so far except that there is an evidence that the deposited aerosols did not fully stick to the wall. Indeed, when later in the test, nitrogen was injected into the circuit to inert it, the activity in the containment suddenly increased, indicating mechanical resuspension had taken place.

Iodine deposition, mainly by vapour condensation (see Figure 7), is more important, 27 ± 5%, and in better agreement with the predictions of codes such as SOPHAEROS.

The most striking finding in FPT-0, not anticipated by the best estimate pre-calculations, was the presence of gaseous iodine in the gas capsules located at the outlet of the steam generator, i.e. at a temperature of 150°C. The capsules were located downstream of either filters or impactors. It turned out that the efficiency of filters was not perfect and corrections for the presence of iodine under aerosol forms were necessary. The efficiency of impactors was much better but it was found that the large surface of steel reacts with gaseous iodine. Thus the figure which came out is mostly qualitative, but the observations made in the containment confirmed that significant fractions of the transported iodine at 150°C may have been in a gaseous form.

Preliminary data from FPT-1 confirm the overall behaviour observed in FPT-0, but with this time typical material and fission product concentrations. In particular, gaseous iodine was again detected at 150°C with gas capsules which were significantly improved with respect to iodine trapping, as a lesson learnt from FPT-0 (zeolite filling structures).

A particularly interesting finding in FPT-1, still under evaluation, is the rather large flow of caesium detected at the inlet of the steam generator for the ten minutes which followed the scram. This could suggest that revaporisation phenomena had taken place in the very late stage of the test.

iii) Behaviour in the containment

In FPT-0, nearly 85% of the aerosols, mostly composed of silver, and 70% of the iodine which arrived at the inlet of the steam generator, reached the containment. It is important to note that silver was in large excess relative to iodine (molar ratio of the order of 5000). The same trend is observed in FPT-1, with a molar ratio lower by more than a decade.

- ***Behaviour in the containment atmosphere***

Figure 8 shows the evolution of the I131 activity in the atmosphere of the containment during FPT-0, as seen by the on-line spectrometer. The activity culminated some 20 minutes before the scram, indicating that aerosols were deposited faster than they were arriving. The analysis of the impactors revealed that the aerosols had a mean mass aerodynamic diameter of about 6μm during the degradation phase. About 5 hours after scram, the activity has decreased and reached a plateau, the remaining activity being attributed to the deposition on the outer wall of the containment.

Indeed, as seen on Figure 5, a large fraction of aerosols (11% of what was injected at point C for silver, 18% for iodine) was deposited on the outer wall, in contradiction with all pre-calculations (less than one percent). It is recalled that the outer wall was hotter than the containment atmosphere. Note the larger fraction of iodine trapped on the stainless steel wall and on the contrary, the much lower fraction of caesium. As seen later, the larger fraction of iodine may be correlated with the rather large amount of gaseous iodine which entered the containment during the degradation phase and which could be then adsorbed by the steel.

Concerning the caesium behaviour, it is in contradiction with the observation that aerosols were quite homogeneous, unless lixivation took place either on the suspended aerosols or after deposition on the wall. The poor level of detection of the caesium isotopes during the first days after the transient makes it difficult to arrive at a conclusion on this point.

The reason for the large deposition on the outer wall is today unexplained. Thermal-hydraulics simulations are in progress on a well instrumented replica of the Phebus containment vessel, SISYPHE, in order to investigate the flow patterns along the wall.

About 10% was deposited, as expected, on the condensing part of the condensers, by thermophoresis and diffusiophoresis. Note that iodine and silver remained stuck to or adsorbed on the condenser as evidenced in the final balance depicted on Figure 5. On the contrary, the amount of caesium found on the condensers is low, as it was dissolved in the condensing steam and entrained to the sump.

The analysis of the gas capsules located downstream of the filters and impactors, after correction for the fraction of iodine which reacted with the large surfaces of the impactors, revealed that the gaseous concentration of iodine was in fact greatest in FPT-0 during the early degradation phase, i.e. during a phase rich in hydrogen: a little bit less than 5% of the initial bundle inventory (with a precision of ± 1%) as measured by the first capsule attached to a filter sampled at 13520s, i.e. just after the cladding oxidation phase. The activity of iodine in suspension at that time in the vessel (see Figure 8), as measured by the on-line γ spectrometer, is consistent with the sum of masses of iodine sampled by the filter and the corresponding capsule. It turns out that nearly half of the iodine entering the containment vessel at that time was gaseous.

The gaseous iodine concentration then decreased regularly (see Figure 9) to stabilise after some 15 hours at a value of less of 0.1% of the initial bundle inventory (after correction for the radioactivity decay, i.e. assuming a typical iodine inventory constituted of long life isotopes). It is worth mentioning that the analysis of the Maypacks tends to indicate that this iodine in the long term was essentially organic iodine, with however the reservation that the knitmesh filters, which were supposed to trap the molecular fraction of iodine, did not work properly for a reason which is still unexplained after many out-of pile investigations.

The FPT-1 results are under evaluation. Typically, the same qualitative behaviour is observed regarding gaseous iodine behaviour in the circuit and the containment atmosphere, with however a trend to have a lower fraction of iodine transported in a volatile form in the cold leg during the degradation phase. Note that the actual concentrations are nevertheless greater owing to the largest inventory in fission products in FPT-1.

The X-ray diffractions and photoelectron spectroscopies performed on FPT-0 samples collected on the sedimentation coupons have shown that the aerosols had about the same composition: 30w% Ag, 30w% Re a few w% In, Cd, Sn, W, U, Mo). Silver was metal, Rhenium both metal and oxide whereas the other species detected were SnO_2, In_2O_3, mixture of UO_2 and UO_3. the amounts of fission products were too low in FPT-0 to unambiguously identify the species involved.

Solution analyses of the deposits in the containment have shown that a large fraction of the aerosols was soluble: 80 w% of Re, 15w% of Ag 70w% of Cd. In fact, their solubility increased over the first hours they were present in the containment owing to oxidation processes. On the contrary, In, U and Sn were not soluble.

- *Behaviour in the sump*

The evolution of the iodine activity in the sump for both tests (Figure 10) shows clearly that iodine does not remain in solution. Instead, it deposited as detected by the on-line measurement of the bottom in the case of FPT-1, or as evidenced by the increase of activity after the sump was emptied in both cases.

This observation is confirmed by the analysis of the liquid capsules which show negligible amounts of iodine in the water.

In the case of FPT-0, Figure 5 shows that the mass balances in the sump for Ag110m, I131 and Cs137, as , deduced from the on-line γ measurement and post-test examination using a photomultiplier, were nearly closed. It is interesting to note that silver and iodine were found in the same proportion on the vertical wall and on the bottom, indicating that silver and iodine reacted before deposition and that settling of precipitates on the bottom was not the only process of deposition.

Another important observation is the decrease of the sump pH from 5.0 to 4.0 just after the washing of the vessel bottom in FPT-0. This suggest an effect of a soluble structure material and a serious candidate is Re, as it may form rhenium peroxide Re_2O_7 under the FPT-0 atmosphere conditions. This species is known for reacting promptly with water to form a strong acid $HReO_4$. Nitric acid formed as a result of radiolysis in the atmosphere of the containment is an alternative explanation. Analyses are in progress in FPT-1, where again a strong pH decrease was observed, to resolve this issue.

- *Status of understanding on iodine behaviour [7] [8]*

The amount of gaseous iodine observed at 150°C in the circuit is not supported by thermodynamic calculations. One possible assumption is that iodine, which should leave the bundle as AgI, according to thermo-chemical calculations, may not have time enough to totally recombine in CsI (or RbI) at the steam generator inlet (700°C) and then CdI_2 in the steam generator. Indeed, fission products, when transported from the bundle to the outlet of the circuit, undergo a very rapid cooling (several hundred of °C/s). Therefore, molecular iodine or HI may coexist at the outlet of the circuit together with condensed iodine compounds.

Regarding the containment, current modelling assumed that the only source of production of gaseous iodine in the containment was the sump, as a result of complex radiolytic oxidation reactions involving iodine ions in solution. The acidic conditions prevailing in the FPT-0 and FPT-1 sump were specifically chosen to promote these reactions. Thus, a generation of gaseous iodine was only expected after sufficient aerosol had been collected in the sump.

In reality, volatile iodine was produced at the break and thus, already during the core degradation phase, there was a significant fraction of gaseous iodine in the atmosphere of the containment.

Secondly, it has been shown that iodine becomes insoluble in the sump, as a result of very fast reactions with the silver coming from the control rod and which is largely in excess and subsequently, the production of gaseous iodine by the sump in FPT-0 was quite negligible. This was attributed to 3 possible mechanisms:

- heterogeneous reaction of insoluble solid particles with dissolved iodine species (I^- and I_2),
- solution reaction of silver and iodine ions giving insoluble AgI possibly in colloidal suspension,
- heterogeneous reaction of I^- and I_2 with Ag/Ag+ colloidal particles.

Out-of-pile tests in support to the Phebus program, performed by SIEMENS and AECL (in the RTF facility) have indeed shown that it was a very efficient reaction (a few tens of minutes for the Ag/I molar ratio of FPT-0). The formed particles would settled only on the bottom and it could not explain the large deposition observed on the vertical wall of the sump.

The fact that 15w% of silver had become soluble tends to indicate that the two other mechanisms were possible and would explain the depositions seen on the vertical wall of the sump. Test are in progress within the 4th European Framework R&D Programme to study Ag/I chemistry under severe accident conditions.

The mid term behaviour of gaseous iodine can be explained as the result of an equilibrium with the exposed surfaces of the containment, in particular the painted ones which reacted with adsorbed iodine, desorbing it slowly as organic iodine (see Figure 9).

A first analysis of the FPT-1 results confirms this trend but work is in progress to confirm if the circuit was as in FPT-0 the only source of inorganic volatile iodine for the containment.

The implication for the reactor is pending on improvement of code modelling and revised evaluations. For those nuclear power plant having control rods made of silver alloy (typically the 900 MWe PWRs), it could be a larger iodine content in the fission products released in case of an early leakage in the reactor containment. On the contrary, in the mid term, no large amount of gaseous iodine should be produced even if there is a failure to inject soda in sump.

This lesson is of course also pending on complementary tests, in particular those supported by the European Commission, to check the stability of AgI under β radiations (AgI is stable under γ radiations) and on the outcome of coming Phebus tests.

PREPARATION OF THE NEXT EXPERIMENTS

The next experiment (FPT-4), to be performed in mid-1999, aims firstly at investigating the release of low volatile fission products and transuranium elements from a solid debris bed ; release during the transition from a solid debris bed to a molten pool and from a molten pool will also be measured. The experimental configuration will be rather different from the previous tests. A debris bed has been pre-fabricated using fuel fragments (size range 2-5 mm) coming from irradiated EDF fuel rods (about 33GWd/tU) and fully oxidised zircaloy cladding shards. The amount of fuel is 3.2 kg whereas the mass of oxidised cladding material is 0.8 kg. The debris bed is surrounded by a thermal shroud made of thoria and zirconia, as in previous tests. It is sitting on a non active debris bed, made of depleted urania, and surrounded by a hafnia neutronic shield. This design has been chosen in order to prevent too great an axial melt progression in the course of the experiment.

The experiment will consist of several temperature plateaux, the first ones being used for thermal calibration and the last three ones being experimental. The first experimental plateau will be realised at 2200 K, the objective being to compare volatile fission products release in rod-like and solid debris bed configurations. The second plateau at 2700 K will be devoted to studying the release of low volatile elements from solid debris. The power will then be increased up to the formation of a molten pool, containing about one half of the fuel inventory. The released material will be trapped in sequential filters located in the upper part of the in-pile test device (see Figure 11). Different filters will be used for the steady-state temperature plateaux and for the transients.

The development of the FPT-4 in-pile test package has required a good deal of technological research, from the design of the filtering and flow diverter system up to the meticulous operations needed to load the debris in a tiny canister and transport it from Chinon (EDF hot lab.) to Cadarache without any disturbances for the bed.

Non-destructive and destructive post-test analyses will provide information on the amount and composition of released material, the chemical activity of different species and the aerosol morphology. It is also planned to perform specific post test studies using samples from the released elements. These studies will address issues such as re-vaporisation under different conditions, using for instance transpiration or Knudsen cell experiments.

The preparation of the test through pre-calculations has involved a number of different teams, the main contributors being IPSN and Sandia National Laboratory. In the course of the preparation, the question of urania oxidation and subsequent volatilisation as uranium trioxide was raised. After an extensive bibliographic study [9], it turned out that the uncertainty on uranium release in such a configuration was very high (up to a factor 100). Though the study of fuel volatilisation was not included among the FPT-4 primary objectives, we expect to gather new information from that test.

The following test (FPT-2), scheduled about six months after FPT-4, is part of the FPT-0-1-2 series, starting from a fuel bundle geometry and including a silver-indium-cadmium control rod. It will be performed under steam poor conditions, so that part of the fission product release and transport will occur under reducing conditions. Boric acid will be introduced as an additive in the coolant flow, with a potential impact on fission product chemistry (e.g. caesium borate formation). Changes in the containment conditions (hot evaporating and alkaline sump), will allow to study new situations for iodine volatility. It is expected to gather enough information from the three integral experiments FPT-

0, 1 and 2 to understand the fission product behaviour, - in particular for iodine -, under severe accident conditions in presence of Ag-In-Cd.

FPT-2 will also address another issue : the possible poisoning of passive autocatalytic recombiners. Taking advantage of the Phebus « representative » fission product source, small recombiners coupons will be exposed during about half an hour to the Phebus containment atmosphere (see Figure 12). They are provided by different suppliers : NIS, AECL, SIEMENS, GRS and the French « Institut de Recherche sur la Catalyse ». Mainly through post-test analyses, information will be obtained on recombiner poisoning by fission products. The time of coupon exposure is chosen in order not to disturb the overall chemistry of the containment vessel. This means that another issue, that is the impact of recombiners on fission product chemistry (e.g. re-vaporisation of condensed species), is not a primary objective of the FPT-2 test. This point will probably need further investigations.

The definition of the last two tests (FPT-3 and FPT-5) is currently underway. Out-of-pile experiments such as CORA [10] have shown the impact of boron carbide absorbers on the early phase of fuel degradation. FPT-0 and FPT-1 have shown a major impact of Ag-In-Cd control rod material on iodine chemistry, whilst the degradation products of a boron carbide absorber might have a different impact [11]. It is therefore proposed to run the FPT-3 experiment in conditions similar to the FPT-2 (reducing conditions), replacing the Ag-In-Cd control rod by a boron carbide absorber.

The FPT-5 experiment should investigate the consequences of air ingress into a partially damaged core. Under very highly oxidising conditions, some elements, such as ruthenium which is of radiological importance, might become volatile and be largely released. The possibility of air ingress has been identified in two situations : shutdown accidents, with an open reactor pressure vessel , and after the reactor pressure vessel lower head failure. The occurrence probability of such sequences and the associated accident scenarios (e.g. the fuel temperature and degradation state for which air enters the pressure vessel) are plant-dependent and are being further assessed before designing the Phebus FPT-5 experiment.

CONCLUSION

The first two Phebus-FP experiments have proved the capability of the Programme to improve our understanding of the Source Term issue but also of the physics of core degradation and core melt progression.

The most striking findings for the source term are the presence of significant amounts of volatile iodine at the break in the cold leg and the insolubility of the iodine in the sump because of the silver released by the control rod. The main source of volatile iodine for the containment atmosphere is thus the circuit and not the sump by radiolytic effects, as previously thought.

As a lesson learnt from Phebus, a review of the models used in the iodine codes has been initiated, introducing in particular reactions with the silver released by the Ag-In-Cd control rods used in most western PWR plants.

These findings might have an impact on source term evaluation. They have to be consolidated by the coming Phebus-FP tests which will investigate complementary fields of parameter variation (presence of boric acid in circuit, alkaline sump, use of a B_4C rod instead of an Ag-In-Cd rod, absence of control rod material).

The first Phebus experiments have also provided valuable information which has a significant impact on our understanding of the physics involved in severe accidents, as the releases of semi volatile elements in steam rich environment and in the late phase, the composition of representative reactor aerosols, their lower than foreseen retention in the circuit and their larger deposition on heated walls in the containment, the indication of resuspension and revaporisation phenomena. Data on hydrogen production and core melt progression in steam rich environment have been also obtained which are being taken into account in the core degradation codes.

FPT-4, planned for mid 99, is the most challenging. It will help to reduce the uncertainty regarding the evaluation of the low volatile element release, which is presently of the order of some decades.

Phebus FP will also provide a quite unique occasion in FPT-2 to investigate the behaviour of hydrogen recombiner catalytic materials under prototypical severe accident conditions.

A complete understanding of the Phebus experiments and its transposition to the reactor case will continue to require intensive efforts for the years to come. It is therefore necessary to strengthen the collaboration already in place within the Phebus international community to better organise the interpretation works and encourage related research activities using the results from Phebus. Areas of high priority appear to be iodine chemistry in the circuit, AgI stability in the sump, and the volatilisation of iodine from deposited aerosols.

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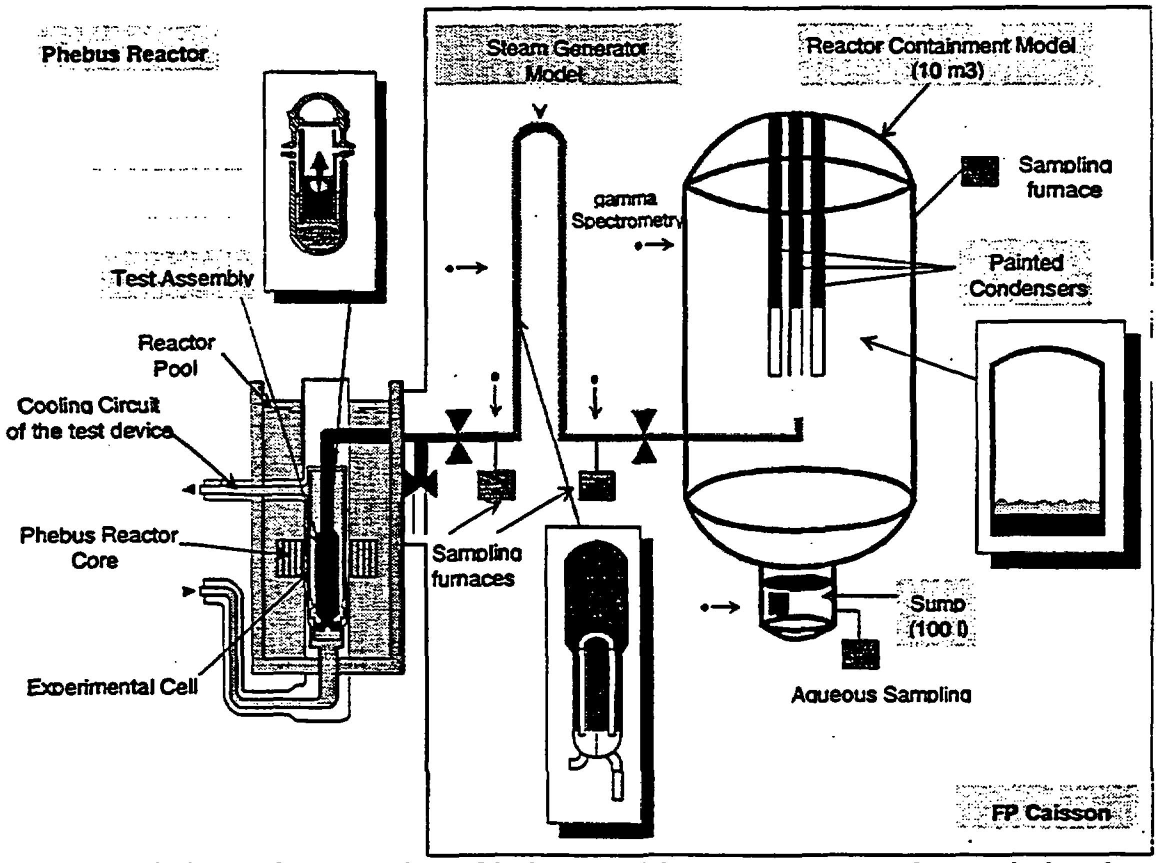

Fig. 1 : Schematic view of the Phebus FPT-0 and FPT-1 experimental circuit

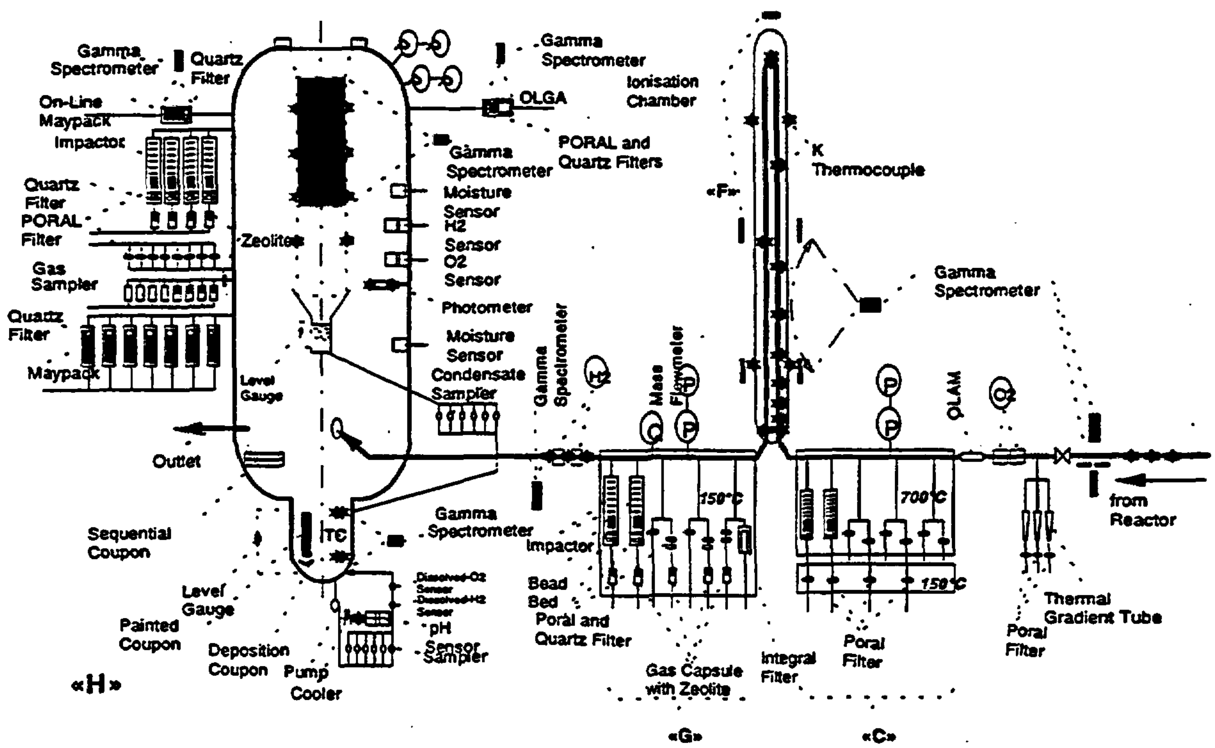

Fig. 2: The FPT-1 circuit instrumentation

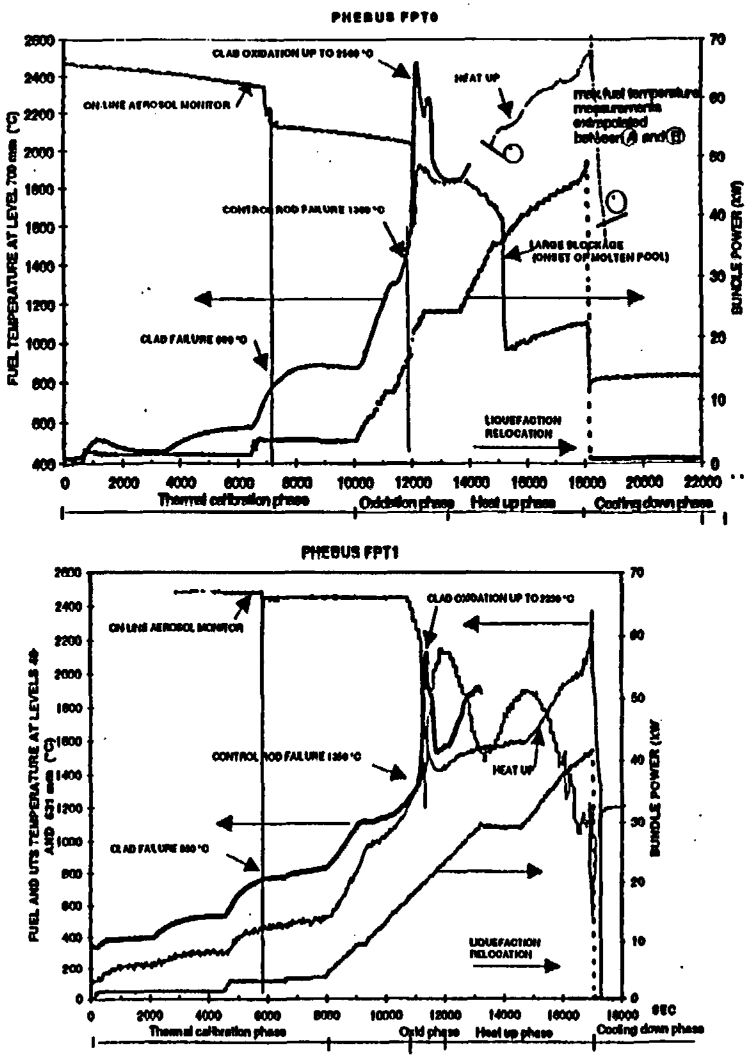

Fig.3 : FPT-0 and FPT-1 Degradation Scenarios

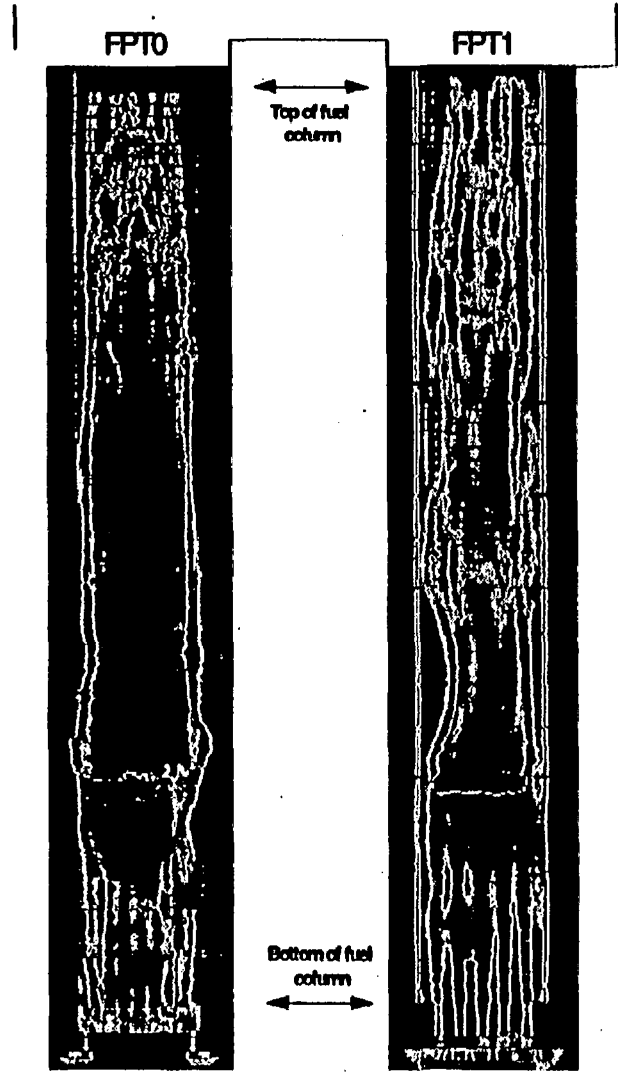

Fig. 4 : FPT-0 and FPT-1 post-test radiographies

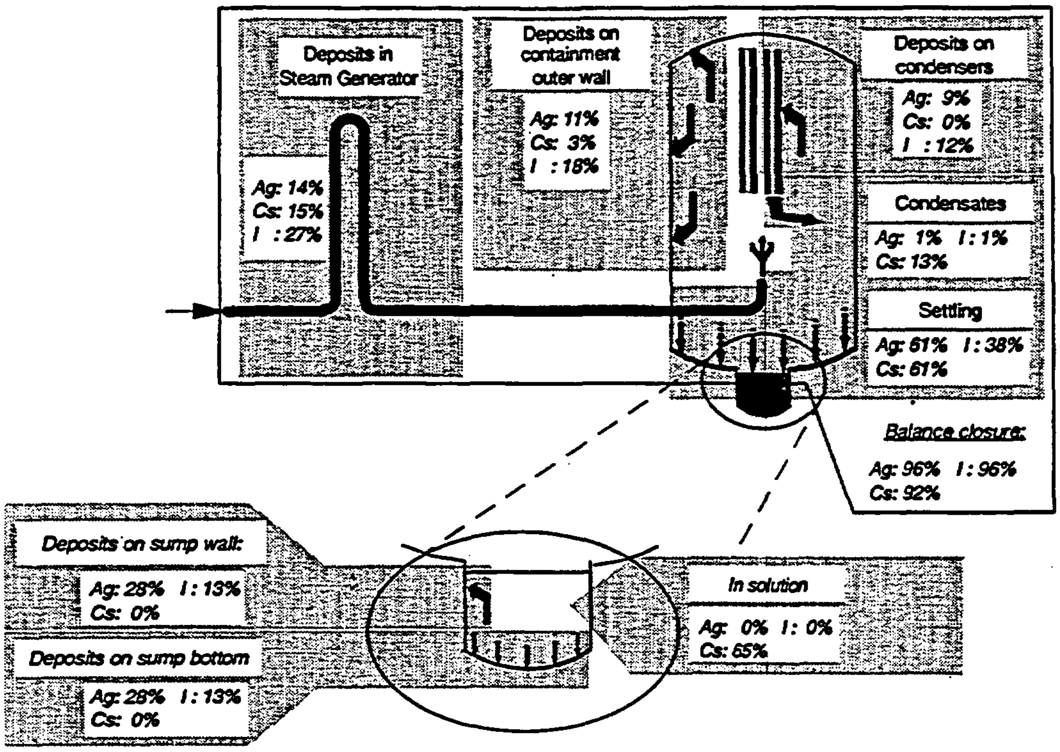

Fig. 5: FPT-0 overall mass balance for Ag, Cs and I in circuit and containment models (in % of masses injected at inlet of steam generator)

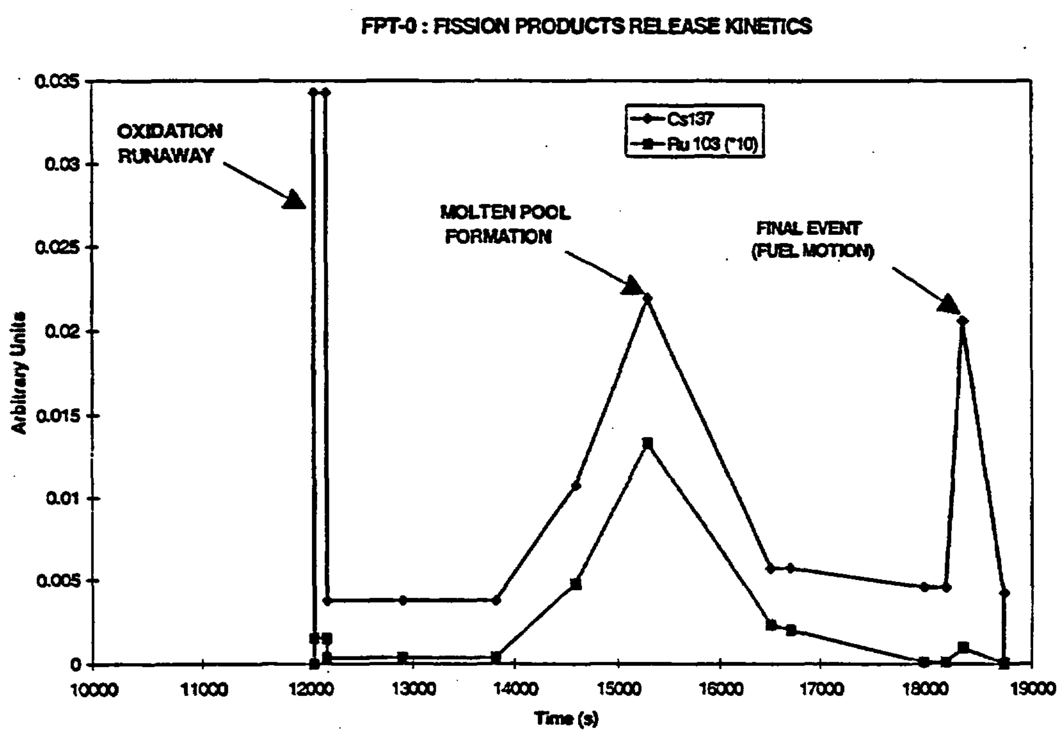

Fig. 6 : FPT-0 Emission Kinetics

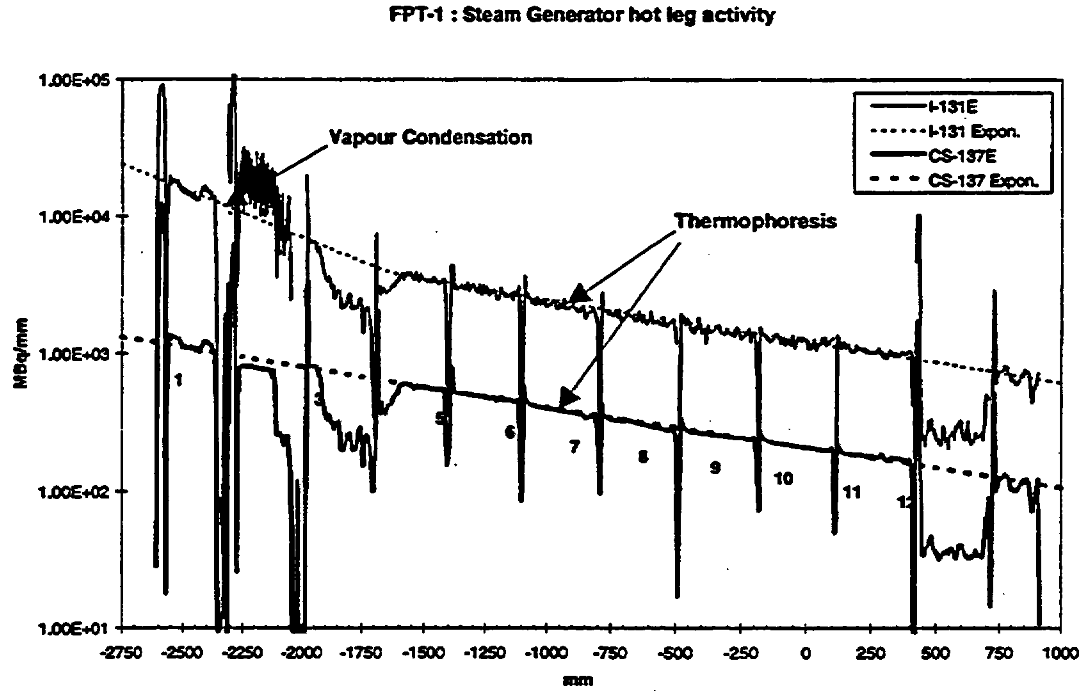

Fig. 7 : FPT-1 Deposition Profile in Steam Generator

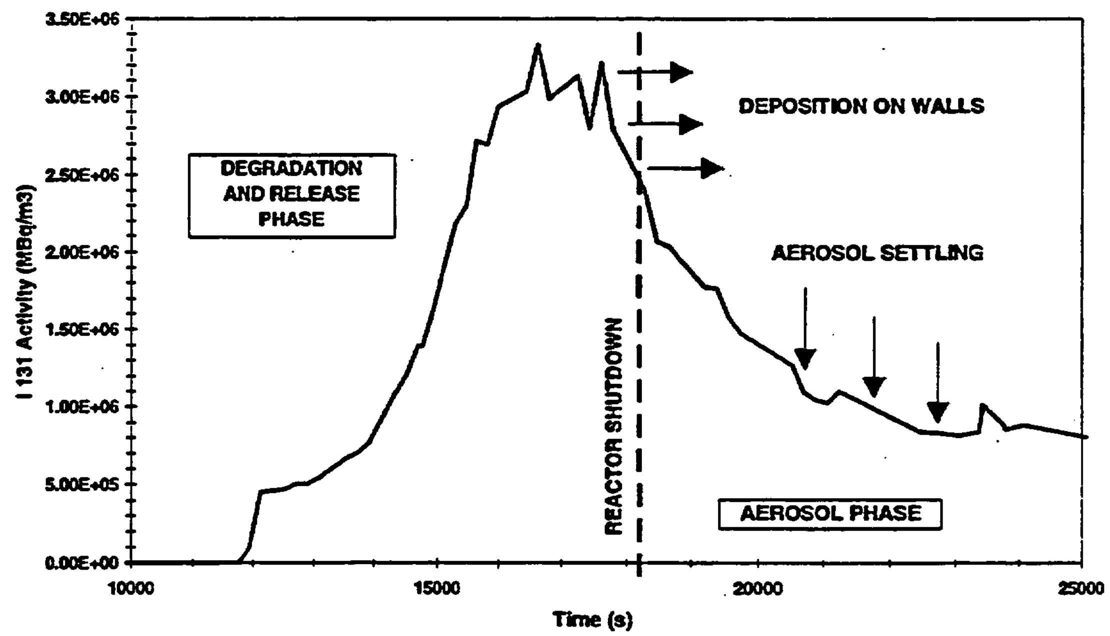

Fig. 8 : Evolution of Iodine Activity in the Atmosphere during FPT-0

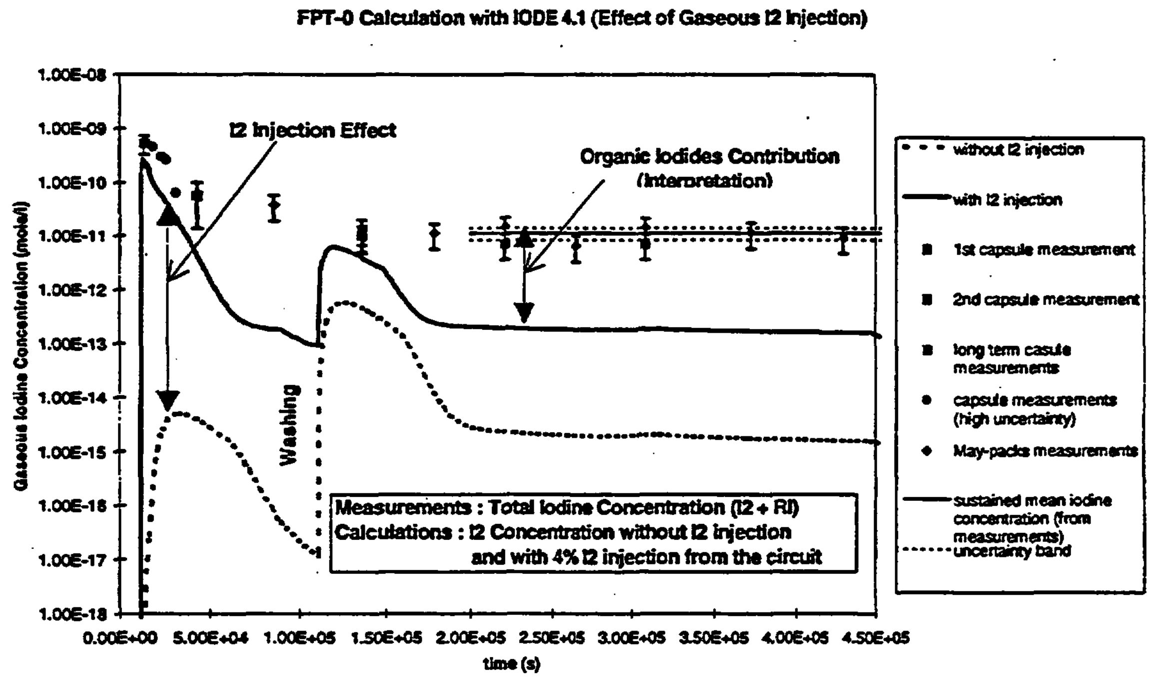

Fig. 9 : Evolution of Gaseous Iodine Concentration in the Containment Atmosphere during FPT-0

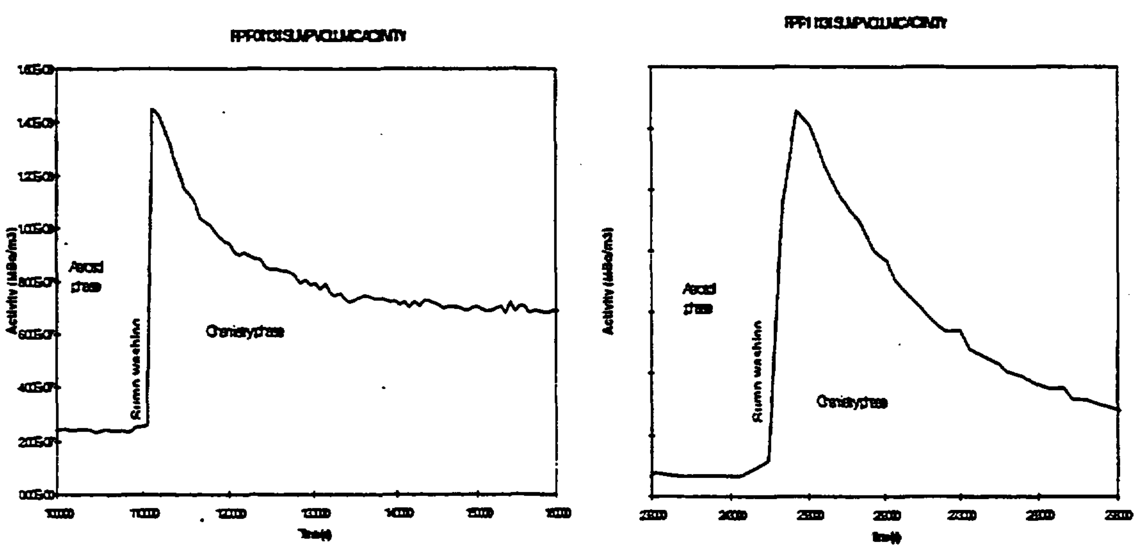

Fig. 10 : FPT-0 and FPT-1 Iodine Activity in the Sumpu

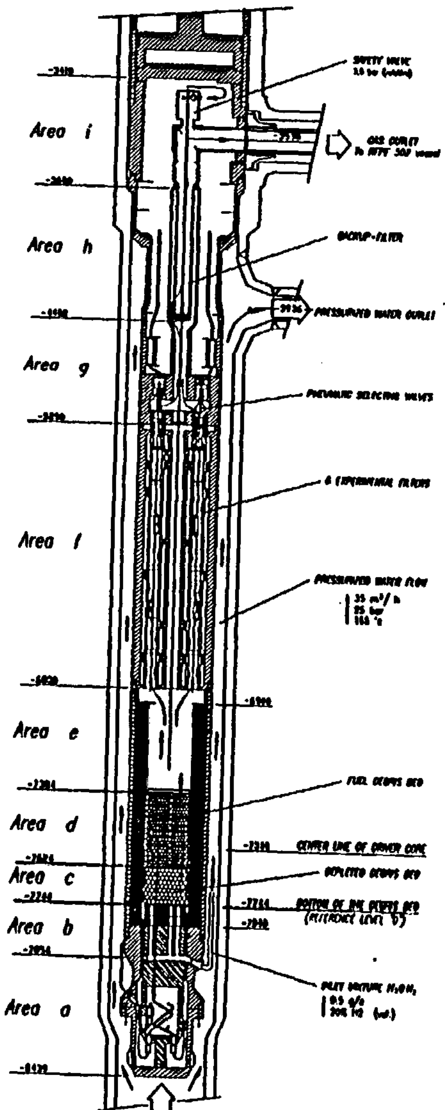

Figure 11 : FPT-4 Test Section

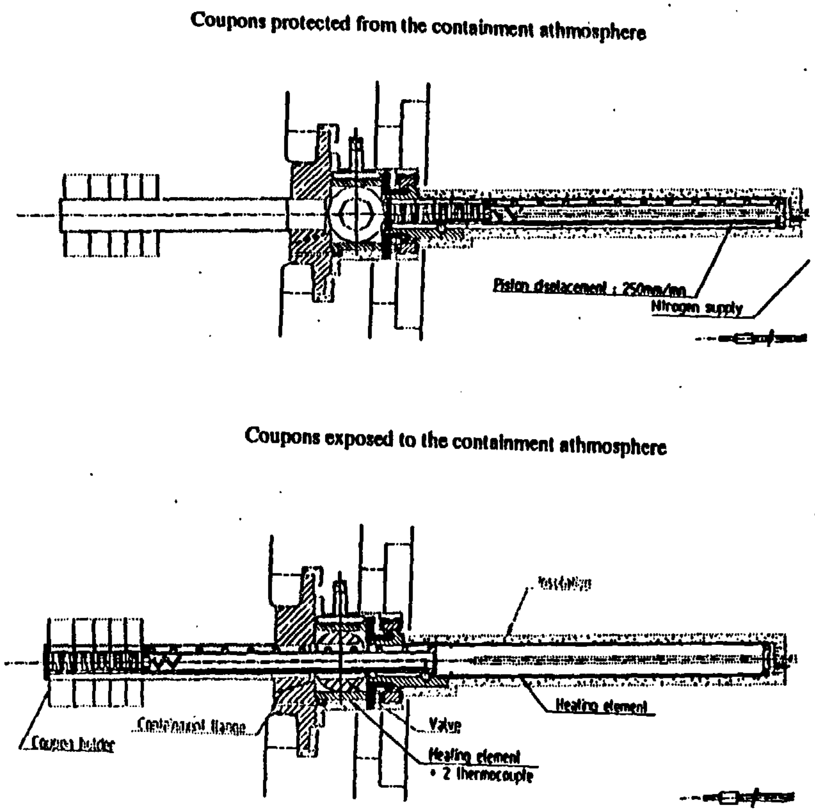

Figure N°12 : Recombiner Coupons for FPT-2

ON THE RE-ENTRAINMENT OF AEROSOLS FROM A BOILING POOL AFTER A SEVERE CORE MELT ACCIDENT

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Keywords: aerosol entrainment, containment venting, entrainment modeling

Abstract - *One major objective of the REVENT program is to experimentally determine the amount of aerosols re-entrained from a boiling pool during the controlled filtered venting of the containment after a severe core melt accident. Experimental studies were carried out in a scaled-down model containment (factor 1:20). Balance equations are used to describe the break-up of bubbles at the pool surface, and, to quantify the entrainment factor. An analysis has been carried out which indicates that the entrainment factor depends largely on the size of the bubbles in the pool as well as on the velocity field above the pool surface. These findings have led to experimental studies of the bubble size and the (transient) velocity field inside the model containment.*

1. Introduction

In western nuclear power plants several layers of protection are provided to ensure safe operation. However, probabilistic studies show that the global frequency of a severe core melt accident is of the order of 10^{-5}/year per plant. In the case of such an accident the containment only provides limited protection against release of fission products. The controlled filtered venting of the containment is an additional safety concept and one possible means to avoid containment failure by over-pressure. Rust *et al.* [2] describe the venting concept implemented in the Swiss NPP's. Kudo *et al.* [4] report results from depressurization studies which have been carried out at JAERI, Japan.

Understanding the release of active and non-active substances into the containment atmosphere and their transport inside the containment are parts of our research effort. Aerosol release and aerosol behaviour during several accident scenarios were investigated in a large number of separate effect studies and integral experiments. But, most of the previous research programs do not consider the concept of containment venting, which will lead to severe changes in the containment atmosphere. Containment venting may cause flashing of the water pool at the bottom of the containment or, at least, will cause boiling of the pool. Thus previously deposited aerosols will be re-entrained into the open venting system. Releases, which were otherwise small, may increase by orders of magnitude due to this effect.

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The convective flow field inside a containment depends on a large number of parameters. For a thorough study of aerosol transport in the containment atmosphere, it becomes obvious that it is not well understood how transient phenomena affect the entrainment mechanism. As a consequence, most computer codes used to predict the aerosol transport and deposition in the containment do not take into account such effects. Recent experimental efforts (e.g. Ardey and Mayinger [5]) are focusing on the effect of transient flows on aerosol entrainment.
The major objective of the REVENT program is to quantify the amount of aerosols which are entrained from the surface of a boiling pool, transported through the containment atmosphere and separated in the filter systems. Different rates of depressurization, $\Delta p/\Delta t$, are considered [3]. Details of the experimental facility are presented in section 2. Both water soluble (Na_2SO_4, CsI, KI) and non soluble (SiC) model substances were used as model fission products during the pilot scale experiments. In section 3, relevant transport processes are discussed and results from integral measurements are presented. In section 4, balance equations are used to study the influence of quantities which were not accessible experimentally (e.g the mean bubble diameter) on the entrainment mechanism. The objective of this paper is to present a theoretical analysis focusing on the influence of such parameters as (a) the size of the bubbles in the pool and (b) the fluid velocity in the containment atmosphere as well as (c) to encourage thorough experimental studies on how aerosol entrainment is affected by transient flow phenomena.

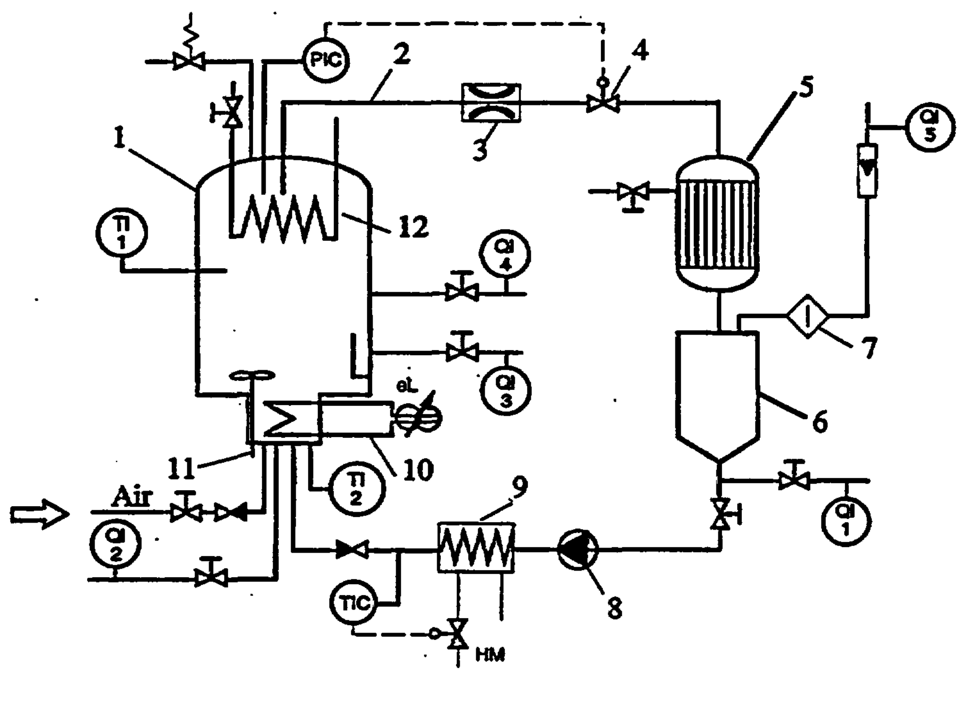

Fig.1. Experimental facility.

2. Experimental

The amount of aerosols released from the surface of a boiling water pool under steady state conditions and moderate depressurization were measured in the REVENT facility shown in Fig. 1. Measurements were carried out in a scaled-down model containment (factor 1:20). It consists of an isolated pressure vessel with a volume of 5 m^3 (1). The pressure inside the vessel is 4 bar. The temperature corresponds to the equilibrium temperature, $\theta = 143°C$. In the sump of the vessel water boils above an electrical heating plate, (10), with a diameter of 55 cm. The heating plate simulates the decay heat and is operated at a constant power $\dot{Q}_H$ of approximately 20 kW. In this paper, (mass) concentrations are denoted with C [mg/l]. The water in the boiling pool contains model fission products (soluble or non soluble) at a concentration C_{BP}. The steam atmosphere above the sump contains aerosols (mean diameter d_p). A cooling coil located at the top of the vessel allows the heat flux $\dot{Q}_C$ to be removed. A discharge pipe, (2), is mounted at the top of the model containment. A fraction of the steam atmosphere passes the discharge pipe, an orifice (diameter 10 mm), (3), and the valve (4). The valve adjusts the mass flow of steam, $\dot{m}_C$, so that the pressure inside the model containment is kept at the constant level of 4 bar. After the valve (4), $\dot{m}_C$ passes the condenser (5). The condensate is fed into a liquid-gas separator, (6). In the case that $\dot{m}_C$ contains non condensable gases (this case is not considered here), the gas fraction would be released to the atmosphere at the top of the separator. The condensate then passes through a re-circulation pump, (8). A heat exchanger, (9), allows the temperature of the condensate to be adjusted. To obtain quasi steady state conditions, $\dot{m}_C$ was re-circulated into the sump.

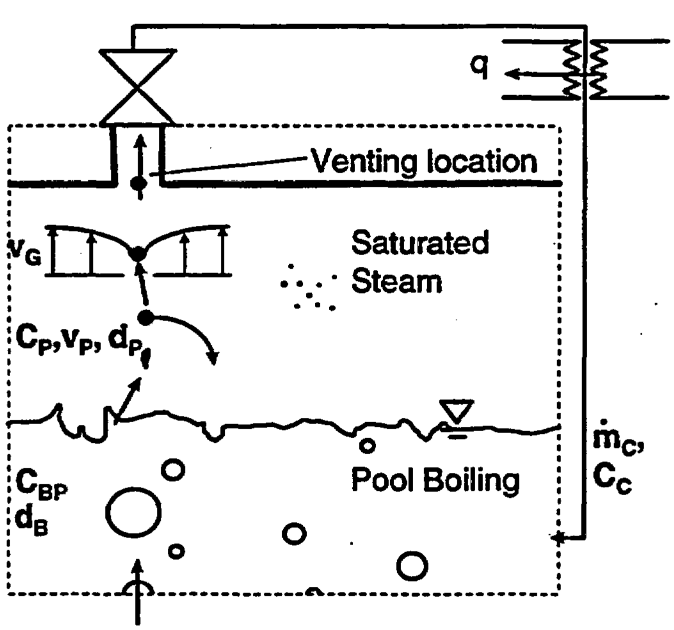

Fig. 2. Schematic diagram of the transport processes inside the model containment.

3. Transport Processes and Experimental Results

Figure 2 illustrates relevant transport phenomena inside the containment atmosphere. The concentration of the model fission product in the sump is C_{BP}. The condensate concentration is C_C. Therefore, the concentration in the dispersed liquid phase, C_P, is the same as in the boiling pool. As mentioned earlier, for the present experiments the containment atmosphere was saturated and did not contain non condensable gases.

Experiments presented in this paper were conducted for a steady state operation $(\Delta p / \Delta t \doteq 0\,\mathrm{Pa/s})$ at a pressure of 4 bar. Figure 3 shows the results for C_C obtained by Müller [3] as a function of C_{BP} for the different model substances (▲ CsI, ■ KI, ◆ Na_2SO_4) [3]. The reason of choosing these model substances is their relevance to accident scenarios. Because the concentrations are small, the effect on the density of water, ρ_f, has been neglected.

As one would expect, the values of C_C are found to increase for increasing C_{BP}.

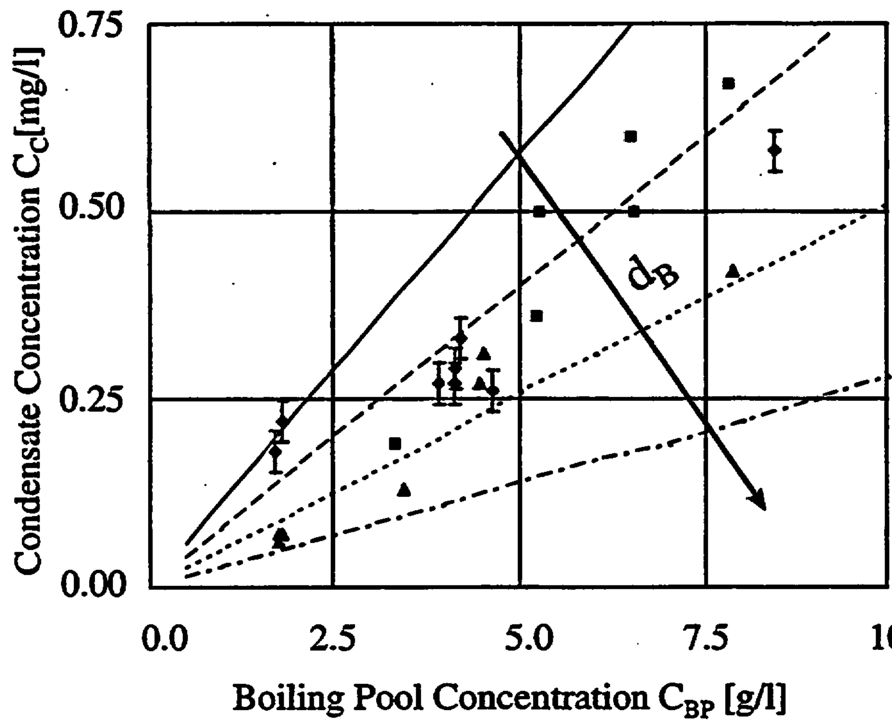

Fig. 3. Experimental results for the condensate concentration C_C as a function of C_{BP} [3].

The entrainment factor E is defined as the mass flux of entrained drops, $\dot{m}_f$, divided by the mass flux of vapor, $\dot{m}_g$. E can be obtained from the measurement data if C_P is assumed to be equal to C_{BP}. This assumption is restricted to steady state conditions where condensation and evaporation effects at the surface of the droplet can be neglected. The mass flux of entrained aerosols is $\dot{m}_f = C_C / C_{BP} \cdot \dot{m}_C + \dot{m}_{f,wall}$. The term $\dot{m}_{f,wall}$ respects deposition at the containment walls.

The mass flux of vapor is $\dot{m}_g = (C_{BP} - C_C) / C_{BP} \cdot \dot{m}_C + \dot{m}_{g,wall}$, where $\dot{m}_{g,wall}$ is the contribution due to condensation at the containment walls. With $C_C \ll C_{BP}$ the entrainment factor follows:

$$E \doteq \frac{C_C / C_{BP} \cdot \dot{m}_C + \dot{m}_{f,wall}}{C_{BP} \cdot \dot{m}_C + \dot{m}_{g,wall}}. \tag{1}$$

If $\dot{m}_{f,wall}$ and $\dot{m}_{g,wall}$ are both small, the dimensionless entrainment mass flux, E, can be explained as the gradient in Fig. 3.

At this point in time, the transport processes inside the model containment can only be estimated from such integral measurements. In order to develop a more thorough understanding of the underlying transport phenomena the importance of such physical parameters as the velocity field inside the containment atmosphere as well as the influence of the mean bubble diameter, d_B, are studied by an theoretical analysis. The assumptions for its validity are discussed in section 4.

4. Theoretical

Because of the small superficial vapor velocity (j_g=1.5 cm/s) the bubbly flow regime is found in the pool. The mean bubble diameter is d_B. At the pool surface the bubble breakup produces small liquid droplets with the diameter d_P. At the pool surface, aerosols are produced as jet or film drops. The importance of the two mechanisms depends mainly on the bubble diameter. For small bubbles ($d_B \leq 5$ mm) the formation of film drops dominates [7].

The heating power, $\dot{Q}_H$, is about 20 kW. The cross-sectional area of the pool is $A=0.28\ m^2$. The total volume of liquid droplets ejected for each bubble of volume V_{gB} that bursts is

$$V_{fP}(d_B) = \frac{\pi}{2} \int_0^{d_{P,c}} d_P^2 k_P(d_P, d_B) \mathrm{d}d_P, \tag{2}$$

where $k_P(d_P,d_B)$ is the droplet size distribution (not normalized), $d_{P,c}$ denotes the critical droplet diameter. This quantity will be defined in Eqs. (6) and (7). Aerosols with diameters $d_P>d_{P,c}$, will fall back to the pool. The superficial velocity of the droplets can be written as follows

$$j_f = \frac{\dot{V}_F}{A} \doteq \frac{1}{A} \langle \dot{n}_B(d_B) \rangle_B \cdot \langle V_{fP}(d_B) \rangle_B. \tag{3}$$

The expression $\langle \xi(d_B) \rangle_B = \int k(d_B) \cdot \xi(d_B) \cdot dd_B / \int k(d_B) \mathrm{d}d_B$ denotes an average over all bubble diameters, $k_B(d_B)$ is the bubble size distribution, and $\langle \dot{n}_B(d_B) \rangle_B$ is the number of average size bubbles passing the cross-sectional area A per unit time. With the volume flux of vapor, $\dot{V}_g = j_g A \doteq \pi / 6 \cdot \langle d_B \rangle_B^3 \cdot \langle \dot{n}_B \rangle_B$, the entrainment factor follows:

$$E = \frac{\rho_f \cdot j_f}{\rho_g \cdot j_g} \doteq \frac{\rho_f \langle V_{fP} \rangle_B}{\frac{\pi}{6} \rho_g \langle d_B \rangle_B^3}. \tag{4}$$

Following a suggestion of Ginsberg [7] Eq. (4) can be rearranged:

$$E = \left(\frac{\rho_f}{\rho_g}\right)\left(\frac{V_{fP}}{V_{gB}}\right) F\left(d_{P,c}\right). \tag{5}$$

In order to calculate E, the volume of droplets ejected per bubble of volume V_{gB} (bubble size distribution) and the droplet size distribution are therefore required. Expressions of V_{fP}/V_{gB} can be obtained from literature data. For the bubbly flow regime, Ginsberg [7] compared estimates of V_{fP}/V_{gB} following suggestions of Cipriano and Blanchard [8], Garner *et al.* [9], and Azbel *et al.* [10]. The different predictions were found to vary over three orders of magnitude.

In this section further simplifying assumptions are made: (a) all bubbles in the pool have an uniform diameter, d_B, (b) the velocity above the sump, v_g, has just a vertical component ($v_g = [j_g, 0, 0]^T$), where (c) j_g is the superficial velocity of the vapor. (d) Wall deposition of aerosols is neglected. Stokes flow is assumed. With a force balance for a stationary drop, $j_f = 0$, one obtains the critical droplet diameter with respect to sedimentation as a function of the superficial velocity of the vapor, j_g:

$$d_{P,c1} = \sqrt{\frac{18 \cdot j_g \cdot \upsilon_g}{g\left(\rho_f / \rho_g - 1\right)}}, \tag{6}$$

where ν_g is the kinematic viscosity and ρ_f the density of water, ρ_g is the vapor density. Droplets with larger diameters cannot be transported through the containment atmosphere. Equation (6) can therefore be understood as the sedimentation criterion depending on the flow field inside the containment.
A second criterion is given by the mean diameter of film drops produced during the rapture of a film film with the critical thickness δ_C [11] formed when a vapor bubble passes the pool surface. For the mean diameter of the entrained droplets one obtains:

$$d_{P,c2} = f\left\{\delta_C\left(\sigma, d_B\right)\right\}, \tag{7}$$

where σ is the surface tension at the water-steam interface. For a bubble size of about d_B=5 mm, the produced jet droplets are expected to be too large to fulfill the sedimentation criterion.
The lines in Fig. 3 represent the values of C_C obtained by the analysis carried out by Müller [3]. From the measurements, C_C is found to increase with increasing C_{BP}. Since it is assumed that the concentration of model substances is sufficiently small, ρ_f and σ are not affected by their presence. For the considered concentrations of model substances, no significant change in surface tension was measured between 20°C and 75°C and 1 bar. Measurements at a temperature close to 143°C are still under way since a pressurized measuring system is required.
Since $C_{BP}=C_P$ was assumed and the effects of wall deposition and wall condensation were neglected, constant entrainment factors, $E(d_B)=const.$, can be found along straight lines through the origin in Fig. 3. C_C is proportional to C_{BP} and the bubble diameter, d_B, affects the condensate concentration: smaller bubbles enhance droplet entrainment and increase C_C.

5. Discussion and Outlook

The comparison between experimental data for C_C , as well as Eq. (3) indicate that the size distribution of bubbles, $k_B(d_B)$, is of importance to understand the mechanism of aerosol formation above the pool. Therefore, attempts have been made to measure $k_B(d_B)$ in the sump of the REVENT facility. To verify assumptions (b) and (c) attempts are made to visualize the velocity field in the containment atmosphere by means of a laser sheet technique.
The presented analysis has been carried out using time averaged quantities. However, present experimental findings indicate that transient flow phenomena play an important role, especially when the convective motion inside the containment atmosphere is enhanced due to the presence of non condensable gases. A thorough study on how transient flows inside the containment affect the entrainment of liquid droplets is seen as a central goal of our experimental efforts.

Financial support provided by the Swiss Federal Nuclear Safety Inspectorate (HSK) is gratefully acknowledged.

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The Linear-Nonthreshold Dose-Response Hypothesis: A Critical Reevaluation.

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NCRP Scientific Committee 1-6

Abstract

In recognition of the need to reevaluate the linear-nonthreshold (LNT) dose-response model periodically in the light of advancing knowledge, such a reevaluation has been conducted recently by Scientific Committee 1-6 of the National Council on Radiation Protection and Measurements. From its review of the relevant data, the Committee has concluded that the weight of evidence suggests that lesions which are precursors to cancer (i.e., mutations, chromosome aberrations), and certain types of cancer as well, increase linearly with the dose in the low-dose domain. Hence it has concluded that although other dose-response relationships cannot be excluded, no alternative dose-response model appears more plausible than the LNT model on the basis of present scientific knowledge.

Introduction

It has been customary, for radiation protection purposes, to assume that the risks of mutagenic and carcinogenic effects of ionizing radiation increase as linear-nonthreshold functions of the dose (ICRP, 1991; NCRP, 1993). However, the existing data do not exclude the possibility that thresholds for such effects may exist in the low dose domain. It is generally acknowledged, therefore, that the linear-nonthreshold (LNT) model needs to be reevaluated periodically in the light of advancing knowledge (NAS, 1990; ICRP, 1991; UNSCEAR, 1994; NRPB, 1995; ACRP, 1996; NCRP, 1997). For this purpose, such a reevaluation of the LNT model has recently been conducted by Scientific Committee 1-6 of the National Council on Radiation Protection and Measurements (NCRP).

The scope of the evaluation, as outlined in the charge to the Committee, was specifically to review the weight of scientific evidence for or against the linear-nonthreshold dose-response model, without reference to any of the related policy ramifications. To provide the breadth of expertise needed for the task, experts from each of the relevant disciplines (i.e., biophysics, biostatistics, epidemiology, genetics, medicine, pathology, and radiobiology) were appointed as members of the Committee; i.e.:

Adelstein, S. James, M.D.(Professor of Radiology, Harvard Medical School);
Brenner, David J., Ph.D.(Professor of Radiology, Center for Radiological Research, Columbia University);
Clifton, Kelly H., Ph.D.(Professor, Department of Human Oncology, University of Wisconsin);
Finch, Stuart, M.D (Professor, Department of Medicine, University of Medicine and Dentistry of New Jersey);
Hall, Eric J., D.Sc.(Professor of Radiology, Center for Radiological Research, Columbia University);
Kearsley, Eric, Ph.D. (NCRP Staff Scientist)
Kronenberg, Amy, Ph.D. (Department of Life Sciences, Lawrence Berkeley, National Laboratory);

Liber, Howard L., Ph.D. (Associate Professor, Radiobiology Laboratory, Harvard School of Public Health);
Painter, Robert B., Ph.D. (Professor, Laboratory of Radiobiology, University of California);
Preston, R. Julian, Ph.D. (Chemical Industries Institute of Toxicology);
Shore, Roy E., Ph.D. (Professor of Environmental Medicine, New York University School of Medicine);
Upton, Arthur C., M.D. (Clinical Professor of Environmental and Community Medicine, UMDNJ-Robert Wood Johnson Medical School), Chairman.

In addition, in an effort to avoid overlooking pertinent data, the Committee sought supplementary input from the scientific community at large, through solicitations published in the open literature and through a formal workshop convened expressly for the purpose.

In evaluating the arguments for and against the LNT model, the Committee endeavored to consider all relevant epidemiological, experimental, and theoretical lines of evidence, including the extent to which adaptive responses (UNSCEAR, 1994) may be expected to alter dose-response relationships for the carcinogenic and mutagenic effects of radiation at low dose levels. A draft report prepared by the Committee will soon be submitted to the NCRP for formal review and will be available on the NCRP web-site (http://www.ncrp.com) during the review process. The report is summarized briefly below.

Biophysical Considerations

Because of the stochastic nature and microdosimetric pattern of energy deposition by ionizing radiation, a single radiation track is judged to be capable of causing a double-strand break or more complex lesion in the DNA of an exposed cell (Brenner and Ward, 1992; Goodhead, 1994). Although the probability of such a complex lesion increases with increasing LET, the probability is judged to be non-zero for X-rays and gamma rays; i.e., even when the dose is low enough so that multiple energy deposition events in a given site within the DNA of any one cell are rare, such sites in some cells can be expected to receive large enough amounts of energy to produce complex lesions. Effects that are produced autonomously in individual targets are, therefore, expected to increase linearly with the dose, since the energy deposited per charged particle does not change with the dose, but merely the proportion of targets that are affected (Goodhead, 1988).

The above conclusion is consistent with the available evidence, but it is recognized that the existing experimental and epidemiological data come from doses well above the range in which there is an average of no more than one radiation track per cell nucleus (Goodhead, 1988). Inferences about dose-response relationships in the mSv dose range inevitably, therefore, involve extrapolations fraught with uncertainty.

DNA Damage and Repair

Radiation can cause various types of alterations in the DNA of exposed cells (e.g, damage to nucleotide bases, single-strand breaks, double-strand breaks, DNA-protein cross-links, and multiply damaged sites), the frequency of which generally increases linearly with the dose (Ward, 1995). The simpler types of lesions , which occur at a high frequency in unirradiated cells, tend to be highly reparable. In contrast, the more complex lesions (double-strand breaks and multiply damaged sites) are rarely observed in unirradiated cells, and their repair tends to be error-prone at the relatively high dose levels where it has been amenable to investigation (Ward, 1995). Whether the repair of such lesions may be less error-prone in a cell that is traversed by only a single radiation track remains to be determined. Furthermore, although the repair of DNA damage may be enhanced in some cells by prior exposure to an appropriate "conditioning" dose of radiation (Le et al, 1998), it is not clear whether the repair of all types of lesions is enhanced, how widely the capacity to mount such an ameliorating response is shared by different types of cells, or whether the response can be elicited at all under conditions of low-dose-rate irradiation. Also, although cell cycle check points and other mechanisms normally act to facilitate the repair of DNA damage or to eliminate cells in which the damage remains unrepaired, such mechanisms may not operate effectively in cells in which one or more of the responsible homeostatic genes (e.g., p53) has been mutated or lost (e.g., Nicholson and Thornberry, 1997).

Mutagenic Effects

DNA damage that remains unrepaired or is misrepaired may be expressed in the form of mutations and/or chromosome aberrations. Mutations of virtually all classes (point mutations, larger deletions, and genetic recombination events) are produced by ionizing radiation (NAS, 1990; Sankaranarayanan, 1993; UNSCEAR, 1993). The mechanisms by which the disparate classes are produced differ in detail, however, with the result that the dose-effect relationships may vary accordingly. In contrast to point mutations, for example, which are postulated to result from a single "hit", or DNA lesion, and thus to exhibit a linear dose response, genetic recombination events between homologous chromosomes and mutations arising from a DNA double-strand break are postulated to require at least two "hits" for their production and thus to exhibit a a quadratic response except in the low-dose range, where both such "hits" result from the same radiation track and occur with linear kinetics (Sankaranarayanan, 1993).

Since the pioneer studies of Muller, dose-response relationships for the mutagenic effects of ionizing radiation have been investigated in detail in many species, including humans. As a result, extensive data are available on the response of cells irradiated in vivo and in vitro, with the latter yielding the most reliable and precise dose-response data thus far. Both sets of data show the frequency of mutations to increase with either linear or nonlinear kinetics, depending on the dose rate and LET of the radiation, the class of mutation being examined, the genetic background of the exposed cells, and other variables (Sankaranarayanan, 1993). In general, low-LET radiation has been found to be several times more effective at high dose rates than at low dose rates, but even at the lowest dose rates tested the frequency of mutations has appeared to increase linearly with the dose of low-LET radiation in most instances (UNSCEAR,1993). The data are also reasonably consistent in showing the mutagenic effectiveness of high-LET radiation to be significantly greater than that of low-LET radiation and to depend less, if at all, on the dose rate of irradiation (UNSCEAR, 1993); in fact, experiments with micobeam irradiation have shown that a single alpha particle traversal may suffice to triple the frequency of mutations in some cells (Hei et al, 1997). There is also evidence that acute irradiation can give rise to a persistent genomic instability; i.e., the mutation rate has been observed to remain elevated for many generations after acute irradiation in hamster cells, and the resulting mutants have resembled those associated with spontaneous, rather than radiation-induced, mutations (Little et al, 1997).

It is noteworthy that pre-exposure to a small "conditioning" dose of low-LET radiation has been observed to halve the frequency of mutations induced by a subsequent "test" dose in some types of cells irradiated in vitro (Sanderson and Morely, 1986; Kelsey et al, 1991; Rigaud et al, 1993; Zhou et al, 1994; Sasaki, 1995). Still to be determined, however, are whether such an adaptive response can be elicited under conditions of chronic low-level irradiation, the extent to which the response may reduce the risks of different classes of mutations, and the degree to which individuals of differing genetic backgrounds may vary in their capacity to mount such a response.

Chomosome Aberrations

Errors in the repair of DNA damage, in DNA replication, or in chromosome segregation can also lead to abnormalities in chromosome number and/or structure. The types of such chromosome aberrations vary, depending on the stage of the cell cycle in which they are produced (Brewen et al, 1973). In an unreplicated G_1 or S-phase chromosome, for example, failure to repair a double-strand break in the DNA may result in a chromosome-type aberration, such as a terminal deletion, in which both chromatids are involved; or the misrepair of two or more such breaks may result in an interchange aberration (dicentric, reciprocal translocation, or ring). In a replicated region of a chromosome, the lack of repair or misrepair of such damage may result in a chromatid-type aberration (involving only one of the two chromatids of the chromosome). The available data imply that the majority of all aberrations require at least two DNA lesions (double-strand breaks, base alterations, cross-links, or complex lesions) for their formation Brewen et al, 1973). It is noteworthy, however, that chromosome aberrations can also be caused to arise many cell generations after irradiation, as a result of radiation-induced genomic instability (e.g., Morgan et al, 1996).

Dose-response relationships for the induction of chromosome aberrations have been documented extensively in various species, including humans. After low-LET irradiation, the dose-response relationships for inter- and intra-change aberrations are typically linear-quadratic at high doses and high dose rates, and more-nearly linear, with a shallower slope, at lower doses and lower dose rates ; whereas after high-LET irradiation, the dose-response curves typically rise more steeply, are linear, and are relatively dose-rate-independent (Bender et al, 1988).

Pre-exposure to an appropriate "conditioning" dose of radiation has been observed to reduce by roughly one-half the yield of aberrations produced by a subsequent "test" dose (UNSCEAR 1994; Vijayalaxmi et al, 1995; Wolff, 1996). The protective effects of this adaptive response last only a few hours, however, and they vary markedly from person to person, some individuals being nonresponders. Furthermore, a dose of at least 5mGy delivered at a rate of at least 50 mGy per minute appears to be required to elicit the response (Shadley and Weincke, 1989). It is noteworthy, therefore, that the frequency of chromosome aberrations has been observed to be increased roughly in proportion to the accumulated dose in radiation workers and persons residing in areas of elevated natural background radiation levels (e.g, Pohl-Rühling and Fischer, 1983; Bender et al, 1988).

The consequences of a given aberration for the affected cell vary, depending on the particular type of aberration in question (Savage, 1979). The biomedical importance of such changes is underscored, however, by the growing evidence linking particular chromosome alterations to the causation of specific types of cancer and other diseases (e.g., Hagmar et al, 1994).

Carcinogenic Effects

Dose-response relationships for the carcinogenic effects of ionizing radiation have been investigated through experiments with cultured cells and with laboratory animals, as well as through epidemiological studies of irradiated human populations. The salient findings obtained through each of these approaches can be summarized as follows:

Oncogenic transformation of cells in vitro. Although the oncogenic transformation of cells in vitro is not equivalent to carcinogenesis in vivo, it has provided a useful model system for elucidating various aspects of the neoplastic process. As a result, extensive dose-response data have become available on radiation-induced oncogenic transformation of rodent cells in culture (Cox and Little, 1992). The process of transformation appears to involve a succession of steps, and in most cases conversion of a cell to a malignant state can be identified only by demonstrating its tumorigenicity for animals. The details of each step remain to be elucidated in full, but the activation of oncogenes and/or the inactivation or loss of tumor-suppressor genes have been implicated in some instances (e.g., Mendonca et al, 1998). . The typically high frequency with which the initial step is produced, and the evidence that neighboring cells may possibly influence the process (e.g., Siggs et al, 1997), suggest that epigenetic changes also may be involved. Furthermore, susceptibility to transformation varies makedly with the genetic background of the exposed cells, their stage in the cell cycle, and other variables (Cao et al, 1992). Not surprizingly, therefore, the dose-response curve for transformation is complex in shape and subject to variation, depending on the particular experimental conditions investigated.

Human cells in primary culture have thus far resisted efforts to transform them by radiation or other means; however, the dose-response data for a hybrid (HeLa-normal fibroblasts) human cell line, in which transformation has been shown to involve the deletion of a tumor-suppressor gene on chromosome 11 and other changes in chromosome 14 (Mendonca et al, 1998), are similar qualitatively and quantitatively to the dose-response data for radiation-induced transformation of C3H 10T^{1}/2 mouse fibroblasts (Redpath et al, 1989). Extensive dose-response data are available for C3H 10T^{1}/2 cells, but as yet there is relatively little information on the shape of the curve below 1 Gy. In general, low-LET radiation has been observed to be less effective than high-LET radiation and to decrease in its transforming effectiveness with decreasing dose rate (Hill et al, 1985; Balcer-Kubiczek et al, 1987). The effectiveness of high-LET radiation, by contrast, has been observed to remain constant or even to increase with decreasing dose rate (Hill et al, 1985). There is evidence, moreover, that traversal of the cell nucleus by an average of only one alpha particle may suffice to transform a significant proportion of exposed cells (Hei et al, 1996).

Exposure of C3H 10T^{1}/2 cells to a small "conditioning" dose of gamma radiation has been reported to protect such cells against transformation by a subsequent "test" dose of radiation and even to reduce their risk of spontaneous transformation (Azzam et al, 1994). These observations are suggestive of an ameliorative adaptation response; however, they must be viewed with caution, pending further information. Given the narrow window of sensitivity to radiation-induced transformation of cells in G_2/M (Miller et al, 1992; Cao et al, 1992, 1993), it is conceivable that the "conditioning" dose merely depletes the population of cells most susceptible to radiation-induced or spontaneous transformation.

Carcinogenic effects in laboratory animals. Benign and malignant neoplasms of many, but not all, types have been shown to be inducible in laboratory animals by appropriate irradiation (UNSCEAR, 1986; Upton et al, 1986). Most of the neoplasms appear to be clonal growths, arising through a succession of stages (i.e.,"initiation", "promotion", and "progression"). The dose-response curves for such neoplasms vary markedly, depending on the type of neoplasm in question (e.g., Fig. 1), the species, strain, sex, and age of the exposed animals, the dose, dose rate and LET of the radiation, and other variables.

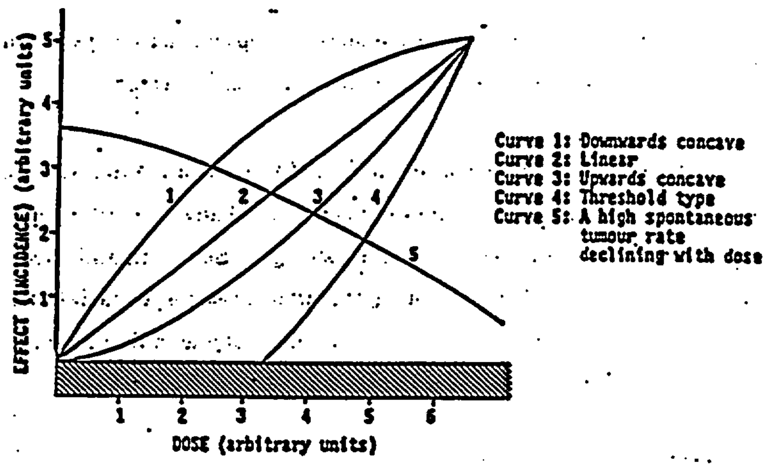

Figure 1. Variations in dose-response curves for different types of neoplasms in laboratory animals (from UNSCEAR, 1986).

Low-LET radiation has generally been observed to be appreciably less effective than high-LET radiation, and its effectiveness has been observed to decrease with decreasing dose rate, in contrast to the effectiveness of high-LET radiation, which has tended to remain constant, or even to increase, with decreasing dose rate. At dose levels above 1 Sv, substantial dose-response data are available for some types of neoplasms, and in certain instances the data are consistent with linear or linear-quadratic dose-response relationships for low-LET radiation and linear relationships for high-LET radiation, depending on the dose and dose rate of irradiation (e.g., Fig. 2).

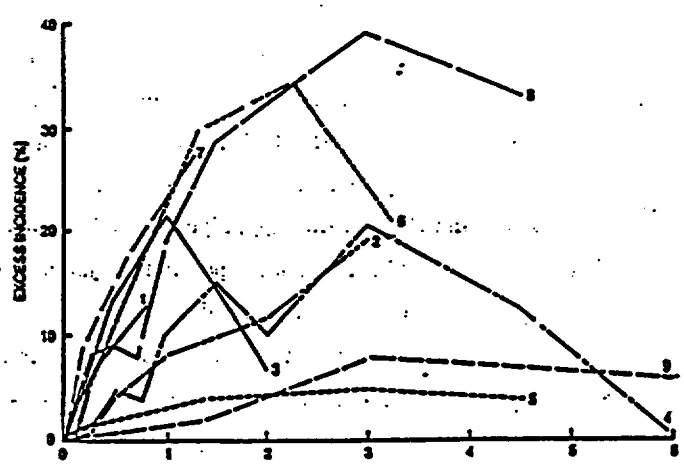

Figure 2. Incidence of myeloid leukemia in mice as a function of the dose, dose rate, and quality of radiation. Curves 1, 3 and 6: acute fast neutron irradiation in RFM, CBA, and RF/Up mice, respectively; curve 7: chronic fast neutron irradiation in RF/Up mice; curves 4 and 8: acute X-irradiation in CBA and RF/Up mice, respectively; curve 2: acute gamma irradiation in RFM/Up mice; curves 5 and 9: chronic gamma irradiation in CBA and RF/Up mice, respectively (from NAS, 1990).

In no instances, however, do the existing data suffice to define the dose-response relationship unambiguously in the range below 0.5 Sv. For certain types of neoplasms (e.g., osteosarcomas), moreover, doses orders of magnitude higher have been required to cause a detectable increase in the frequency of the tumors. The existence of a practical, if not real, threshold for osteosarcomas is also indicated by the fact that the average latent period for their induction by a bone-seeking radionuclide varies inversely with the dose rate, exceeding the natural life span at sufficiently low dose rates (Raabe et al, 1990).

In spite of the marked and unexplained variations in dose-response relationships among different types of neoplasms, the dose-dependent shortening of the lifespan resulting from the oncogenic effects of whole-body irradiation is remarkably similar in animals of different species (Sacher, 1966). Furthermore, to the extent that the reduction of longevity by whole-body irradiation may be interpreted to provide an integrated measure of the combined oncogenic effects of the radiation on different organs of the body, it is noteworthy that the effect appears to increase linearly with the dose after acute whole-body irradiation (UNSCEAR, 1982) and linearly with the daily dose on chronic, life-long exposure to doses in the range of 1-10 cGy per day (Fig. 3).

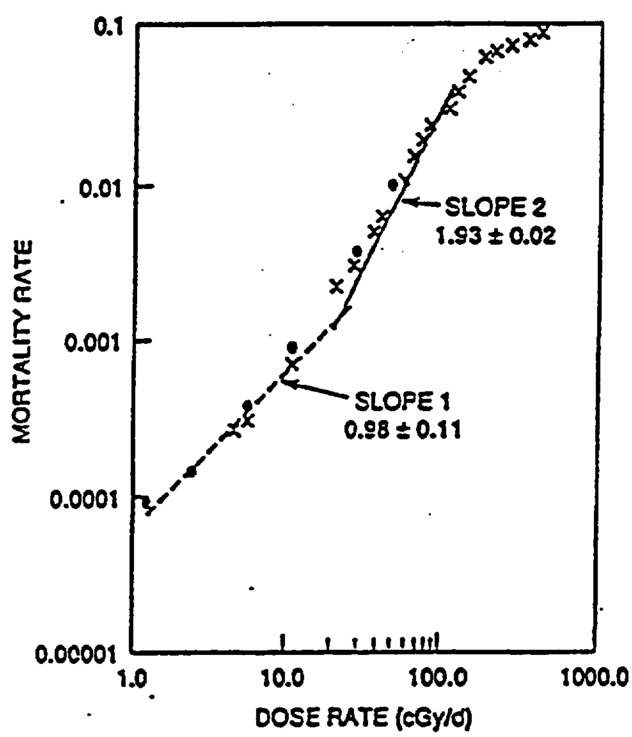

Figure 3. Influence of the dose rate on the rate of mortality in mice exposed daily throughout life to whole-body gamma radiation [from Fry (1994), based on data from Grahn (1970) and Sacher (1973)].

In many cases, the dose-response curves have been observed to be altered profoundly by modifying factors. The yields of tumors of the thyroid, ovary, pituitary, and mammary glands, for example, may be markedly increased or decreased by the presence or absence of the appropriate hormonal stimulation (UNSCEAR, 1986). Likewise, the induction of murine thymic lymphomas can be inhibited by the shielding of hemopoietic tissues during irradiation or by the injection of isologous bone marrow cells postirradiation (UNSCEAR, 1986). Also, in combination with other carcinogens, the effects of radiation have been observed to be additive, synergistic, or antagonistic, depending on the agents and exposure conditions in question (UNSCEAR, 1986). Although the mechanisms of action of the different carcinogens and modifying factors remain to be fully elucidated, most of them appear to affect tumor-promotion and the later stages of carcinogenesis, rather than tumor-initiation per se (UNSCEAR, 1986).

The precise nature and sequence of the successive steps that are involved in the production of a given neoplasm remain to be determined. It is noteworthy, however, that the frequency with which tumor-initiation is induced by irradiation, as investigated through the transplantation of irradiated clonogens, greatly exceeds the rate of any known radiation-induced mutation, implying that an epigenetic process may be involved (Mulcahy et al, 1984; Clifton et al, 1986; Kamiya et al, 1995). The possibility that tumor-initiation may involve the production of genetic instability is suggested by the progressively rising frequency of chromosome aberrations after irradiation in the mammary gland cells of mammary-tumor-susceptible BALB/c female mice and the lack of such an increase in the cells of mammary-tumor-resistant C57BL/6 mice (Ponnaiya et al, 1997). Other genetic changes, not necessarily related to tumor-initiation, also have been observed in a growing number of instances, including the activation of various oncogenes, the inactivation or loss of tumor-suppressor genes, and the influence of various other cancer-susceptibility genes (e.g., Hino et al, 1993; Bennett et al, 1995; Barlow et al, 1996; Reitmair et al, 1996; Chen et al, 1996; Guerrero and Pellicer, 1997).

Carcinogenic effects in human populations. Dose-dependent increases in the rates of many types of benign and malignant neoplasms are also well documented in irradiated human populations. As in laboratory animals, the frequencies of such growths have varied markedly with the type of neoplasm, the dose rate and LET of the radiation, the age, sex, and genetic background of the exposed population, and other variables (NAS, 1990; UNSCEAR, 1993). Some types of neoplasms (e.g., chronic lymphocytic leukemia) have not been detectably increased in frequency at any dose level, implying that they are not inducible by ionizing radiation, whereas certain others (e.g., osteosarcoma) have been detectably increased in frequency only at high dose levels (> 10 Sv). Since the existing dose-response data come primarily from observations at relatively high doses and high dose rates, however, they do not suffice to define the dose-response curve precisely in the low dose domain for any neoplasm (NAS, 1990; UNSCEAR, 1994; NCRP, 1997).

The above limitations notwithstanding, it is noteworthy that: 1) the overall risk of solid cancers in Japanese atomic bomb survivors is significantly elevated at doses of only 5-50 mSv (Pierce et al, 1996), above which the risk rises linearly with the dose up to 2.5 Sv (Fig. 4);

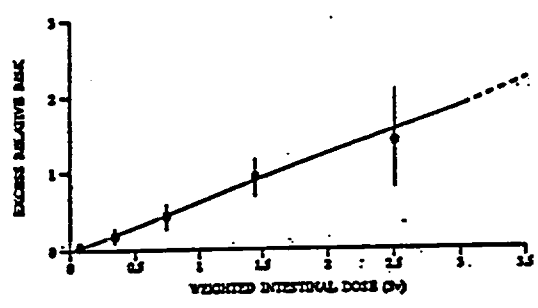

Figure 4. Dose- response relationship for solid cancer, all types combined, in atomic bomb survivors, 1958-1987 (from UNSCEAR, 1994)

2) the results of repeated case-control studies imply that the risk of developing cancer in childhood is increased significantly by prenatal exposure to as little as 10 mGy of X-radiation (Doll and Wakeford, 1997); 3) the risk of female breast cancer appears to increase as a linear-nonthreshold function of the dose of radiation, with fractionated doses of about 10 mGy per fraction, delivered in multiple fluoroscopic examinations during the treatment of tuberculosis with artificial pneumothorax, appearing fully additive in their carcinogenic effects on the breast (Boice et al, 1978; Howe, 1998); 4) the risk of thyroid cancer also appears to rise as a linear-nonthreshold function of the dose following acute irradiation in infancy or childhood, with the risk being substantially elevated by a dose of only about 100 mGy (e.g., Fig. 5); 5) the pooled data from several large cohorts of radiation workers disclose a dose-dependent excess of leukemia in this population (Cardis et al, 1995) that is similar in magnitude to to that which has been observed in atomic bomb survivors, in whom the excess apears to increase as a linear-quadratic function of the dose (Preston et al, 1994); 6) susceptibility to radiation-induced cancer is significantly increased in association with certain hereditory disorders (e.g, hereditary retinoblastoma, nevoid basal cell carcinoma syndrome, Li-Fraumeni syndrome) (Sankaranarayanan and Chakraborty, 1995).

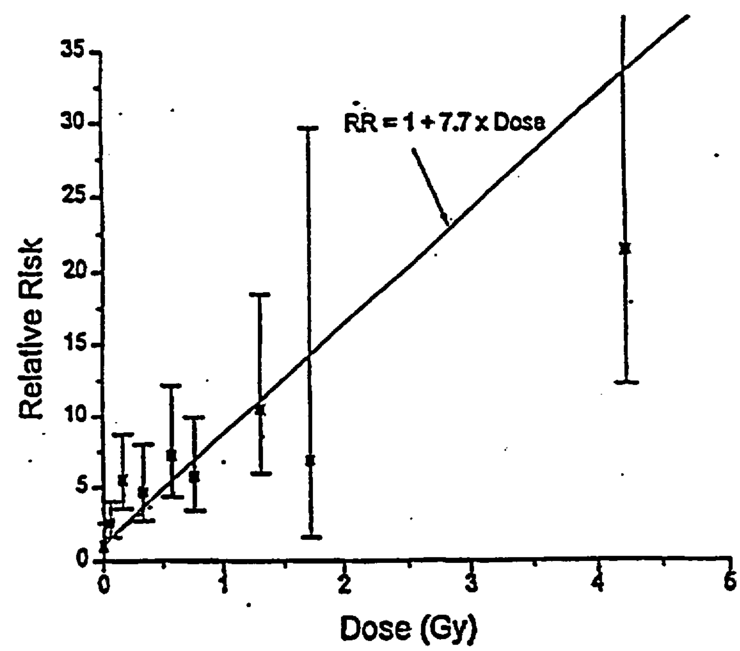

Figure 5. Dose-response data points and fitted curve for thyroid cancer from five pooled studies of external irradiation in childhood (from Ron et al, 1995).

Although cancer rates have not been found consistently to be elevated in populations exposed to low-level radiation, and in some instances the rates have even appeared to be reduced in such populations, the findings do not necessarily contradict the LNT model, in view of the small magnitude of the excess to be expected under such exposure conditions. The failure to detect it may conceivably result from random variation, confounding factors, or other methodological limitations of the studies in question (e.g., Little and Muirhead, 1996; Sinclair, 1998).

Adaptive Responses

The growing evidence that a small dose of radiation may sometimes elicit a transient adaptive response that enhances the resistance of the exposed cell or organism to a subsequent "challenge" dose has prompted many to question the plausiblity of the LNT model (Kondo, 1993; Luckey, 1994; Cohen, 1995; Patterson, 1995; Pollycove, 1995). Various types of adaptive responses have been observed experimentally thus far, including: 1) augmentation of the capacity to repair damage to DNA, genes, and chromosomes (observed in some plant, animal, and human cells); 2) acceleration of the rate of cellular proliferation (observed in different test systems); 3) enhancement of immunological reactions (observed in laboratory rodents); and 4) reduction of intercurrent mortality from infectious diseases (observed under certain conditions in chronically irradiated mice and rats) (UNSCEAR, 1994).

The type of response that has received perhaps the most study to date is the heightened capacity for repair of chromosome damage, first reported by Olivieri et al (1984) and since confirmed by other investigators (UNSCEAR, 1994). In this response, as noted above, a small "conditioning" dose of radiation results in an approximately 2-fold reduction in the frequency of chromosome aberrations produced by a "test" dose administered within a few hours thereafter. It is noteworthy, however, that a dose of at least 5 mGy delivered at a rate of at least 50 mGy per hour is apparently required to elicit the response in human lymphocytes (Shadley and Wiencke, 1988), and that a dose of this magnitude can be expected to cause double-strand breaks and multiply-damaged sites in the DNA of many cells (Ward, 1995). Furthermore, the response lasts for only a few hours at most and is apparently lacking in the cells from some individuals (UNSCEAR, 1994). The extent to which the response can be expected to afford protection against the mutagenic effects of low-level radiation is, therefore, highly uncertain.

By the same token, the reduction of intercurrent mortality from infectious diseases in laboratory mice and rats that has sometimes been observed to result from daily exposure to small doses of radiation has not been accompanied by protection against neoplasia (UNSCEAR, 1982). Nor has it been observed to occur in animals maintained under optimal conditions of husbandry (UNSCEAR, 1982). Whether the effect has any relevance to the LNT is, therefore, doubtful.

Conclusions

In spite of data documenting that adaptive responses may protect against the biological effects of irradiation under certain conditions, the weight of experimental and theoretical evidence suggests that for many of the biological alterations that are precursors to cancer (such as mutations and chromosome aberrations) there is a linear-nonthreshold relationship between risk and dose in the low dose domain. Likewise, although the dose-response relationship for cancer in humans and laboratory animals varies markedly with the type of neoplasm, the dose, dose rate and LET of the radiation, the age, sex, and genetic background of the exposed population, and other variables, the existing data suggest that the dose-response relationships for certain neoplasms do not depart significantly from linear-nonthreshold functions at low doses and low dose rates. For such neoplasms, therefore, although alternative dose-response relationships cannot be excluded, no dose-response model appears more plausible than the LNT model on the basis of present scientific knowledge.

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IAEA/NEA Occupational and Public Dose Studies: US and Global Trends and Benchmarking Databases

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INFORMATION SYSTEM ON OCCUPATIONAL EXPOSURE

Since its introduction in 1992 by the Committee on Radiation Protection and Public Health (CRPPH) of the OECD Nuclear Energy Agency (NEA) with co-sponsorship by the International Atomic Energy Agency (IAEA), the Information System on Occupational Exposure (ISOE) has grown significantly in terms of both utility and regulatory participation. As of 1998, the ISOE Program consisted of 73 utility and 22 regulatory authority participants from 23 countries. The ISOE Program is the first of its kind to offer consolidation of nuclear power plant occupational exposure information into one global database. Representing over 360 operational and 35 decommissioned or shutdown commercial nuclear power plants, participants have large amounts of information available for occupational dose studies and benchmarking. Functions concerning daily operation, refueling outages and other types of shutdowns are broken down in detail to track dose in every part of the nuclear power plant.

Figure 1: Generalized ISOE Organization

The ISOE program is responsible for five major products. Included for distribution to members are periodic special reports, responses to queries, annual reports and new software updates. International ALARA symposia are also an important tool for conveying new dose saving ideas and techniques in the global nuclear industry. NARTC organized the first ISOE ALARA symposium in March, 1997 in Orlando, Florida. The first European Community - ISOE Workshop on Occupational Exposure Management at Nuclear Power Plants was held in September 1998 in Malmö, Sweden. The next International ALARA Symposium sponsored by the NARTC will be held January 31 – February 3, 1999 in Orlando, Florida. The symposia have been successful in both attendance (approximately 150 registrants) and international participation (about 12-15 countries represented).

North American Regional Technical Center

The North American Regional Technical Center (NARTC) was founded in 1994. Members of the Power Reactor Section, Health Physics Society initiated the idea of joining the ISOE and locating the North American Regional Technical Center in a central location which is representative of the North American nuclear industry. The University of Illinois in Urbana, Illinois was chosen as a central location for all North American plants and regulatory agencies. In addition to industry and agency involvement, the NARTC benefits from a rich history of nuclear and radiation science achievements. The first sustained nuclear chain reaction in the world was achieved at the University of Chicago in 1944. The University of Illinois, College of Engineering, is ranked among the best engineering schools in the U.S. The state of Illinois has the largest number of commercial nuclear power plants available in the U.S. Since 1994 NARTC has grown to include 19 BWR, 27 PWR and 22 CANDU plants as members of the North American branch of the ISOE program. The NARTC gained regulatory participation as well, from Canada, Mexico, and the United States. The US Nuclear Regulatory Commission joined ISOE in April 1997 (see Attachment # 2).

The North American Regional Technical Center is one of the four technical centers of ISOE. NARTC is responsible for collecting information regarding the operation of Canadian, Mexican and U.S. nuclear power plants. Many of the products provided by other regional centers of ISOE are also provided by the NARTC while distributing North American BWR and PWR plant comparisons to regional members. Efforts have now expanded to provide North American utilities with In-Service Inspection, Special Maintenance, Sister Plant and similar dose comparisons for BWRs and PWRs. In its continued drive for excellence in the ISOE program, the NARTC hosted the first annual International ALARA Symposium in 1997 as noted above. The international response demonstrated the importance of implementing global involvement in occupational dose information exchange.

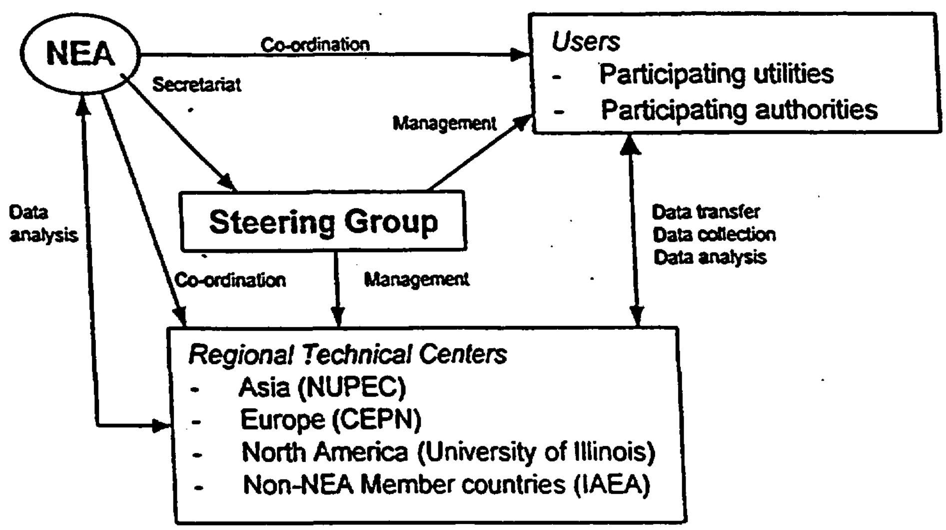
The NARTC is fortunate to have CANDU nuclear power plants within its regional branch of ISOE. Plans to establish CANDU-specific ISOE questionnaires are being implemented to address the inherent operational differences compared to BWRs and PWRs. Notably, however, CANDU plants are interested in motor operated valves, in-service inspection dose reduction techniques, and other issues common to BWRs and PWRs.

Information exchange is important to the functionality of the NARTC. The dosimetric data and discussions of significant experience found in the ISOE database are used by participants to achieve their separate objectives. In addition, requests for specific information are made to the Center mainly by telephone but also via e-mail or fax. Some examples of these requests include the total number of steam generator replacements and dose associated with replacement, lowest annual dose plants, lowest outage dose, and dose due to insulation and scaffolding work. Most of the feedback regarding the requests reflect U.S. nuclear power plants, but information from foreign nuclear power plants are included where available.

Regulator and utility communication on dose justification and optimization issues is of significant importance in the emerging risk informed and performance-based regulatory environment in the U.S. The ISOE program assists nuclear utilities in receiving timely and accurate dose and duration information from sister plants. Member utilities use the ISOE database and reports to benchmark their performance and improve radiological planning initiatives. Utility dose optimization objectives are supported by benchmarking plant performance with sister plant performance and previous site performance.

Dose Information Flow

Around January of each calendar year the NARTC initiates the information exchange cycle by sending blank ISOE questionnaires to both member and non-member nuclear power plants. The power plants are given until April to complete the ISOE data sheets for entry into the NARTC computer. Plants are asked to provide at least a minimum amount of information (e.g. outage dose, outage duration, start-up date, normal operational dose and other shutdown dose). Data verification and validation with RPMs occurs throughout May. The finalized NARTC database files are sent via e-mail to the Centre d'étude sur l'évaluation de la protection dans le domaine nucléaire (CEPN). CEPN consolidates data from all technical centers and distributes a complete and consolidated global database back to the technical centers. The NARTC updates its database to include global data and then distributes the complete global database to its member utilities. Figure 2 visually summarizes the typical information flow through ISOE and its technical centers.



Future of the NARTC

At the request of participants, the North American Regional Technical Center is focusing on more complex occupational dose analysis. Software enhancements will be provided to enable additional participant onsite analysis. Continued emphasis will be maintained on benchmarking and means to focus on risk significant issues. Its role in ISOE is expected to involve not only nuclear power plant occupational exposure, but also exposure at government nuclear facilities, hospitals and research reactors in Canada, Mexico and the U.S. For government facilities, occupational dose tracking would include work associated with yellowcake to final fuel enrichment. The NARTC would include U.S. DOE laboratories in a new database tracking program to perform long-term occupational dose studies. The NARTC will as feasible evaluate integration of the ISOE, REIRS and DOE databases in order to monitor dose for the whole fuel cycle.

ISOE is expanding its database to include NEA level 2 and 3 forms with input from the NARTC regarding construction of specialized forms for CANDU power plants. This expansion will allow more detailed descriptions of maintenance and outage work performed during the operating cycle.

Streamlining the information flow will gradually occur through the extensive use of e-mail and establishment of a web server for information submission and retrieval. All the data entry forms, as well as many of the NARTC products, will be accessible via the internet.

Specialized Studies

The information obtained for many of the studies completed to date originated from five main sources: RP/ALARA (Radiation Protection/As Low As Reasonably Achievable) Committee Meetings, Station ALARA Reports, NUREG 0173 Guides, ISOE Annual Reports and information sheets. Below is a brief description of the primary information sources for data retrieval.

- *RP/ALARA Committee Meetings*

  Preliminary information regarding plant operation and performance is obtained from BWR and PWR committee meeting reports. Participating nuclear utilities provide summary sheets outlining the plant's prior year totals and current goals and status. Other items on the information sheets include recent significant health physics experiences, perceived good practices, future activities and challenges, and notes on performance successes and improvements. Major work performed during the year is noted as well (e.g., steam generator replacement). Related information is also available at the Health Physics Information Forum sponsored by the Nuclear Energy Institute.

- *Station ALARA Reports*

  Station ALARA reports provide detailed descriptions and summaries of work performed during the calendar year. Many of these ALARA reports provide in-depth occupational dose breakdowns at the task level. A critique of each task is also provided, detailing positive and negative performance during the year. Many nuclear power plants also provide separate ALARA reports for major component replacement and plant modifications. Steam generator replacement reports are an example of a document provided by PWRs.

- *NUREG 0173 Guides*

  The NRC documents summaries of U.S. nuclear plant performance for the year by site in NUREG 0713 guides. Data since 1969 is documented in the reports and published annually for both operational and shutdown U.S. nuclear plants. All aspects of annual operation, with regard to occupational exposure, are recorded in the NUREG 0713 guides. Information from Appendix B and Appendix D is the primary source of occupational dose data from the NUREG guides.

  NUREG 0713 Appendix B documents annual whole body doses at licensed nuclear power facilities. Information regarding whole body dose to individuals by site is included, and dose is grouped in appropriate ranges. Appendix B is used to complete part of the ISOE forms and also serves as a dose verification tool since information regarding Total Effective Dose Equivalent (TEDE) is also provided.

  The number of personnel and the dose acquired is recorded and separated by work and job function in NUREG 0713 Appendix D. The information from Appendix D has been used to create historical dose benchmarking trends with respect to In-Service Inspection and Special Maintenance. Historical benchmarking of other aspects of annual operations is also possible. Reactor operations and survey, routine maintenance, waste processing, refueling and totals by job function are the remaining categories open for studies from Appendix D. NARTC is evaluating various strategies regarding how the ISOE databases can be used as a complimentary database to the Regulatory Guide 1.16 report on occupational doses. The goal of this evaluation is to simplify the collection of data and improve the accuracy and timeliness of the data collection and validation process.

- *ISOE information sheets and NEA 1 forms*

The NARTC actively participates in regional meetings and provides information sheets to radiation protection managers for completion. This provides the NARTC with preliminary information valuable for completing the ISOE NEA 1 forms and responding to informal queries. The NEA 1 form documents the annual operations of the nuclear power plants and provides information consolidation. Radiation protection managers that actively participate in ISOE complete these forms and submit the information to the NARTC. Any other information required is retrieved at other discussion forums and through station ALARA reports and the NUREG 0713 guides. Using these sources also provides a means of data verification.

BENCHMARKING AND LONG TERM STUDIES

Three key items of interest were recently addressed that reflected input from radiation protection managers and regulators. Benchmarking U.S. nuclear power plants with respect to pre-defined operational parameters was a primary goal. Preliminary work on the data presentation began in 1996, with increasing utility participation for data completion and continued graphical refinement. Summaries of these annual operational parameters were presented graphically and distributed to radiation protection managers and regulators for performance assessment. Other items of interest included doses expended for In-Service Inspection and Special Maintenance. Data for these two items were never before analyzed and presented in this way and proved beneficial in benchmarking and promoting dose reduction ideas, including some for future regulatory relief.

- United States Nuclear Power Benchmarking

The Information System on Occupational Exposure (ISOE) was established to collect and utilize detailed dose information regarding annual nuclear power plant operations. The importance of this particular database is that the occupational dose information collected is designed to provide task level dose and duration data to facilitate good plant work management practices. Currently the ISOE NEA1 database is used to provide benchmarking graphs regarding the annual operation of a commercial nuclear power plant. The areas of interest regarding plant operation include outage duration, outage dose, normal operational dose, total annual dose, and dose due to major component replacements. (The NEA3 database is used for sharing of experiences that can foster optimization of dose at other stations).

The primary goal of annual nuclear power plant benchmarking graphs is to provide radiation protection managers with a comparative tool to ensure dose expenditures are justified and to optimize work performance. The tool can be used by itself or as an initiator of discussions leading to justification and optimization enhancements. Another goal is to promote excellence among similar (sister) plants by providing quarterly updates to the radiation protection managers. Nuclear power plant dose performance is also ranked in quartile groupings by reactor unit.

Benchmarking initiatives were developed at the March 1997 International ALARA Symposium by participating radiation protection managers. Occupational dose data from U.S. nuclear power plants in 1995 was used and standardized benchmarking graphs appeared in 1997, displaying 1996 data. The initial benchmarking studies only scratch the surface of the potential value of the

ISOE database. Further breakdowns and more detailed dose comparisons regarding sister plants and dose per megawatt produced are planned based on 1998 recommendations from participating utilities.

Outage Duration

Benchmarking outage duration is a general indicator of outage performance. Many nuclear power plants use outage duration to compare and gauge themselves among other plants. Generally, short outages require long-term careful planning and extremely well organized tasks and procedures. Cooperation from knowledgeable and skilled workers, both contracted and utility, is also important for short outages and meeting work management goals.

Outage Dose

Refueling outage cycles range from 12 to 24 months, based in part on the type of fuel placed in the core. Factors other than fuel burn-up, such as steam generator replacement, major plant modifications and regulatory mandates, influence the duration and dose of the refueling outage. Only refueling outage dose acquired is recorded in this category. All other types of plant shutdowns are recorded in a category called Other Shutdowns.

Annual individual dose results in 1995-97 show a trend to higher individual dose, especially for in-house specialty-trained personnel. For example, an increase in the total number of radiological workers receiving more that 2.0 rem (20 mSv) and 3.0 rem (30 mSv) has been observed. More utilities are participating in resource sharing alliances to increase the number of experienced refueling outage workers, reduce costs and achieve economies of scale. Since a high percentage of a station's annual dose is acquired during outages, it is logical that annual individual dose will increase as individuals are assigned to more outage per year to take advantage of work efficiencies from using highly trained and experienced workers.

Normal Operational Dose

Daily nuclear plant operations and tasks fall under the normal operational dose category. In multi-unit sites, where resources are pooled from all reactor units continuously, normal operational dose is monitored and recorded by site. Unit specific dose is difficult to monitor since workers wearing personnel dosimetry at multiple-unit sites perform work functions on the whole site and common equipment. The accepted ISOE approach for recording normal operational dose for multiple unit sites is to divide the total normal operations result by the total number of units on the site. In some instances, nuclear utilities are able to monitor normal operational dose by unit if resources are not shared.

Total Annual Dose

The total annual dose accumulated by personnel at a reactor unit includes the sum of the normal operational dose for the unit, outage dose for the unit, and forced outage dose for that unit. The total amount of occupational exposure accumulated by all contract and utility employees present at the unit during the year are reflected in the total annual dose. Total annual dose is also used to measure nuclear plant performance among industry peers.

In-Service Inspection and Special Maintenance Dose Studies

In-Service Inspection (ISI) is considered a "Good Engineering Practice". ISI is a procedure developed under American Society of Mechanical Engineers (ASME) guidelines for corrosion and/or erosion examination. ISI dose comparisons were performed for 1978-1996. Similarly, special maintenance dose comparisons were performed. This included compiling dose data regarding selected major modifications (exceeding $100,000 U.S.), chemical decontamination, recirculation pipe and steam generator replacement, and removal of RTD bypass piping. This differs from regular maintenance practices because routine ongoing repairs such as those involving valve and seat replacement, as well as lubrication, are not included. The NRC REIRS database provides special maintenance activity breakdowns by contractor, plant and utility personnel.

Piping or weld problems that might be identified by the use of In-Service Inspection techniques may be resolved through the use of relatively simple, "routine" maintenance activities or may be resolved only with the application of special maintenance and/or modification activities. There is then a tie between the results of In-Service Inspection and the initiation of special maintenance. However, special maintenance may be required for reasons other than those of resolving problems identified by ISI techniques. While performance of ISI mandates occupational exposure will be received on an ongoing basis, special maintenance tasks are generally one-time tasks, with the occupational exposure driven by scope of the task. It is only because each plant tends to do some sort of special maintenance each year that occupational exposure due to special maintenance has a longer-term component.

U.S. Data Collection

Data was analyzed using a three-year rolling average format and other normalized formats (see Attachment 3 for several examples). The three-year rolling average format helps to smooth the curves and also to take into account the cycle length variations that begin to affect the curves as the U.S. plants began to move from annual refueling outages to refueling outages every 18 or 24 months. Data is also available in "raw" form and normalized to take into account the different number of operating reactors in the various years.

There is a qualifier that should be placed on the trending of other analyses performed. The numbers documented by the NRC REIRS database are not reported by extremely well defined category designations. Doses acquired performing various jobs may get documented in one category by one plant and in another category by a second plant. This may be true especially in the definitions that licensees give to special maintenance tasks. The Regulatory Guide 1.16 Reports therefore do not unambiguously identify doses in the in-service inspection and special maintenance categories. To that extent, interpretation of the data in the figures has some limitations, but the data should be reasonably consistent across the years for a particular plant and thereby should be reasonably useful for comparison across the years for all plants.

ISI Analysis

In the 1980's doses due to PWR In-Service Inspections remained fairly constant. Contributors to the steady dose trend in later years were steam generator evaluations and eddy current testing. Dose from ISI to the U.S. PWR fleet remains in the range of about 12 person-Sievert (1200 person-rem) per year, with the dose per reactor in the range of about 0.17 person-Sievert (17 person-rem) per year.

The occupational dose from In-Service Inspection activities at Boiling Water Reactors peaked in 1983. Dose received from ISI remains in the range of about 7.5 person-Sievert (750 person-rem) for the U.S. BWR fleet, with the dose per reactor in the range of about 0.18 person-Sievert (18 person-rem) per year.

For the year 1996, whereas dose for ISI at U.S. units averaged in the range of about 0.15-0.20 person-Sievert, reported data from other countries was in the range of 0.01-0.28 person-Sievert. The accrued dose at U.S. units tended to be higher than that in units of other countries. This may be due to differences in regulatory approaches toward weld inspections, snubber inspections and the like.

ASME and Power Industry Initiatives in ISI–Risk Informed Inspection

ASME has prepared several code cases evaluating the effectiveness regarding the past 20 years of nuclear plant piping In-Service Inspections, using the benefits of probabilistic fracture mechanics. A goal is the objective determination of what to inspect and how often those inspections should occur. ASME studies at PWRs indicate that up to 60% of the primary piping weld inspections at nuclear power plants may not be necessary.

Three U.S. nuclear plants have provided submittals to the Nuclear Regulatory Commission for In-Service Inspection requirements modifications (i.e., Arkansas Nuclear One - ASME Code Case N-578, Surry – ASME Code Case N-577 and Vermont Yankee – ASME Code Case N-560.) Issuance of Regulatory Guides by NRC on application specific guidance on in-service testing and in-service inspection is expected.

Results from PWR pilot studies implementing Risk Informed In-Service Inspection at Millstone 3 and Surry show increased safety and significant reductions in inspection costs and collective dose. Millstone 3 experienced an 84% reduction in inspections, while reducing the risk due to pipe failure by half. The overall effect of the inspection reduction is reflected in a 0.15 person-Sievert (15 person-rem) outage dose reduction. Surry experienced similar inspection and dose reduction. At the Surry plant, a 65% reduction in inspection was realized, the risk due to pipe failure was cut in half, and a savings of 0.10 person-Sievert (10 person-rem) per outage was achieved.

ASME Research is developing ground rules at the Browns Ferry Nuclear Plant (BWR) regarding Risk Based Inspection. The overall goal is to increase safety and decrease costs.

This effort should be seen in the context of an overall move toward Risk Informed Regulation, to ensure objective and efficient means of maintaining adequate protection of public health and safety. The NRC and the industry, via organizations such as the Nuclear Energy Institute, are developing approaches to greater use of probabilistic risk assessments and other risk assessment means in development of changes to regulations and regulatory guidance and to plant operational practices.

Special Maintenance Analysis

Lessons learned from the Three Mile Island accident in 1979 prompted additional regulatory requirements for U.S. nuclear power plants. Additional safety equipment and measures were implemented at all U.S. plants. Three Mile Island's effect on special maintenance for PWRs appears to have peaked in 1984, causing a substantial increase in dose across the early 1980s. In 1987 through 1991, a secondary rise in dose is observed. The impact of steam generator replacement, other major plant modifications, and chemical decontamination appears to be the primary cause of this increase. In the 1990s, a significant reduction in dose is seen. The annual dose to the U.S. PWR fleet remains in the range of about 18 person-Sievert (1800 person-rem), with the dose per reactor in the range of about 0.23 person-Sievert (23 person-rem) per year. There are still varied reasons for conduct of special maintenance. For example, plant modifications may be needed to implement changes in regulation or regulatory guidance, or to enhance unit availability, safety, or economics..

United States BWRs were also subject to Three Mile Island backfit requirements in the early 1980s. Occupational dose peaked in 1984 and has decreased since that time. Another contributor to 1980s accrued dose is drywell recirculation pipe replacement projects. A contributor to the significant reduction in BWR occupational dose from special maintenance from 1985 to 1991 was due in part to successful chemical decontamination of BWR recirculation and reactor water cleanup piping and the reduction in frequency of recirculation pipe replacement. Further dose reduction has continued since that time. Accrued dose remains in the range of 20 person-Sievert (2000 person-rem) per year, with the dose per reactor in the range of about 0.50 person-Sievert (50 person-rem) per year. The dose per reactor for special maintenance at U.S. BWRs is noted to be about twice that for U.S. PWRs.

For plant modification and/or special maintenance doses in 1996, the data for the U.S. units showed dose expenditures in the ranges of about 0.30-0.50 person-Sievert. Available data for other countries showed doses in the range of about 0.0-1.9 person-Sievert. In this case, the data for the U.S. units was in the range of doses expended in the other countries.

NARTC has been successful in collecting and analyzing annual occupational dose trends specifically related to In-Service Inspection and Special Maintenance activities. The goal of this study was to assist the industry and regulatory authorities in achieving dose reduction through prudent modification of ISI requirements..

<u>NEA Expert Group Report: Work Management in Nuclear Power Plants:</u>

The Information System on Occupational Exposure (ISOE) Steering Group published an expert group report on Work Management in the Nuclear Power Industry in 1997. The report is one of the first products resultant from the focus on reduced occupational dose and improved management practices that exemplifies the potential of the ISOE.

The preparation of the report was accomplished by Radiation Protection Managers from ten countries including Canada, Finland, France, Germany, Sweden, Switzerland and the US.

Feedback from ISOE member utilities on the report has been exceptionally positive. Some of the feedback received from US member utilities include: a) plants' using the report's outline and text as an ALARA assessment format, b) distributing copies of the book to plant managers and supervisors to reduce occupational dose and achieve greater work efficiency, and c) using the report's concepts in developing an ALARA Enhancement Action Plan, a list of short and long term initiatives to reduce occupational dose.

The report has been used in other countries as well. An example is found in Mexico, a fellow country represented by the North American Regional Technical Center (NARTC) of the ISOE. The Laguna Verde Nuclear Power Plant, Vera Cruz, Mexico, asked for additional copies of the book because the single copy on site had a reference book checkout list backlog of over 3 months. The Radiation Protection Manager received Plant Management approval of ALARA initiatives when they were referenced in the ISOE report. Also, feedback from Canada, the third country represented by the NARTC, suggests further use of the report is likely as the commonalties of nuclear power plant management are recognized to transcend the differences in reactor type.

The report has been translated into six foreign languages to enhance the value of the book by power plant personnel. The foreign translations include Chinese, French, German, Japanese, Russian and Spanish with translations into additional languages being evaluated.

Public Dose

The North American Regional Technical Center, ISOE, has developed gaseous and liquid effluent databases for US nuclear power plants starting with calendar year 1994. The effluent data from 109 operating nuclear power plants is collected from the US Nuclear Regulatory Commission's Public Document Room or the nuclear utility. The data is entered on EXCEL data tables and data analysis is conducted annually to determine population doses.

The US effluent data is shared with the United Nations Scientific Committee on Effects of Atomic Radiation (UNSCEAR) located in Vienna, Austria. UNSCEAR reports are published about every four years and provide a global perspective on the exposure of the public to man-made sources of radiation. The report addresses exposure to the public from the following sources of radiation:

- Natural Sources
- Medical Exposures
- Nuclear Power Plants
- Nuclear Explosions and Weapons Production
- Major Accidents Involving Nuclear Material
- Occupational Exposures

The forty-seventh UNSCEAR report is currently in preparation and is expected to be released in 1999, according to Dr. Burton Bennett, Secretary of the United Nations Scientific Committee on Effects of Atomic Radiation. The US effluent data is also provided to participating North American utilities.

Example Country Report (USA, 1996):

In each ISOE Annual Report, participating countries are invited to provide commentary about significant experiences and trends. The following paragraphs describe U.S. input for one such report.

In 1996, nuclear power plants in the USA focused on work management initiatives to reduce refueling outage duration, occupational dose and operating costs. This focus was developed due to company-wide programs to prepare for de-regulation and competition in the US electric energy market over the next several years. Also, the continued success of specific US nuclear plants, e.g., Limerick and Peach Bottom, in achieving shorter refueling outages for the past several cycles stimulated other US plants to achieve similar outage goals. Recognizing the benefits of international cooperation, one reason for Limerick's and Peach Bottom's success in achieving shorter refueling outages was attributed to the adoption of European approaches to work management and plant maintenance.

The US plants with the shortest outages in 1996 are as follows:

	Days	Man-mSv
Limerick, Unit 1	24.8	1,529
Peach Bottom, Unit 2	19.5	1,320
South Texas, Unit 1	22.6	1,136

Some US nuclear power plants were shut down for much of 1996 for extended maintenance outages (improvements in the material condition of the plants before de-regulation) and due to regulatory requirements to reconcile operational practices with design basis analyses and documents prior to continued operation. An increasing trend in occupational dose was observed in these plants due to more maintenance and inspection activities in the radiological areas of the plants.

PWR Highlights:

US pressurized water reactors implemented new shutdown chemistry protocols to reduce occupational dose by stabilizing radiation fields during refueling outages. Steam generator replacements continued in the US, both in actual execution and in development of future steam generator replacements plans. Point Beach Unit 1 started their steam generator replacement outage in the fourth quarter, 1996 and finished in 1997. The total dose for the Point Beach Unit 1 total dose for the steam generator replacement was 1,880 person-mSv.

Commonwealth Edison Company's Byron Nuclear Power Station developed plans to replace steam generators in 1997 for Unit 2. The feasibility of steam generator replacement for Commonwealth Edison Company's Zion Nuclear Power Station was studied in 1996.

BWR Highlights:

The US Boiling Water Reactor experience in 1996 included increased inspections of reactor vessel internals for cracks. Many US plant are accelerating plans to implement hydrogen water chemistry and depleted zinc injection to reduce adverse chemical environments for reactor internals and control radiation fields. Also, noble metals as a protective coating in reactor internals was tested at the Duane Arnold Nuclear Energy Center in Iowa with promising results.

Health Physics Initiatives:

Two technical topics being addressed by utility and regulatory health physicists in 1996 were skin dose limits for hot particles and effective dose equivalence studies. Research studies were sponsored by industry and regulatory agencies to study dose effects of hot particles on pigs skin. A scientific report is being prepared by the National Council on Radiation Protection (NCRP). Opportunities for regulatory relief of hot particle skin dose limits will be evaluated following the release of the NCRP report.

Effective dose equivalence studies have been conducted by Texas A & M University using Monte Carlo computer analysis. Results of the studies show that placement of a single dosimetry badge on the front upper body of an occupationally exposed workers provides adequate monitoring the worker's dose. Implementation of these concepts will reduce the need to issue multiple badge dosimetry packets to workers assigned to work activities in radiation areas of the power plant and potentially will lead to more accuracy in reported dose equivalent.

US utilities are implementing remote monitoring systems for key in-plant work areas such as refueling floor, BWR drywell and radwaste areas. The worker is outfitted with several electronic dosimeters which are capable of remotely transmitting their readouts to centralized health physics control points. The control points are equipped with closed circuit video monitors and electronic dosimeter readout monitors to track several crews' dose accumulation. Cellular phones are also provided to facilitate communication between the workers and the health physics control technicians. Remote monitoring programs reduce the number of health physics technicians needed in field, potentially provides closer supervision of work in radiation areas and reduces person-rem for the work force.

Some US utilities are using personnel exit monitors as an additional check for internal dose to plant workers. Since workers must pass through a portal personnel monitor upon exiting the radiological controlled area of the plant, the monitors represent an additional monitoring system for internal dose. This system called passive monitoring supplements the existing whole body counting at these sites.

SUMMARY:

The North American Regional Technical Center started as a industry initiative and has expanded into a program that provides occupational dose trends and analysis for the US and global industry. The challenge in the future will be to continuously improve the quality, scope and timeliness of the occupational dose databases and the dose analysis reports.

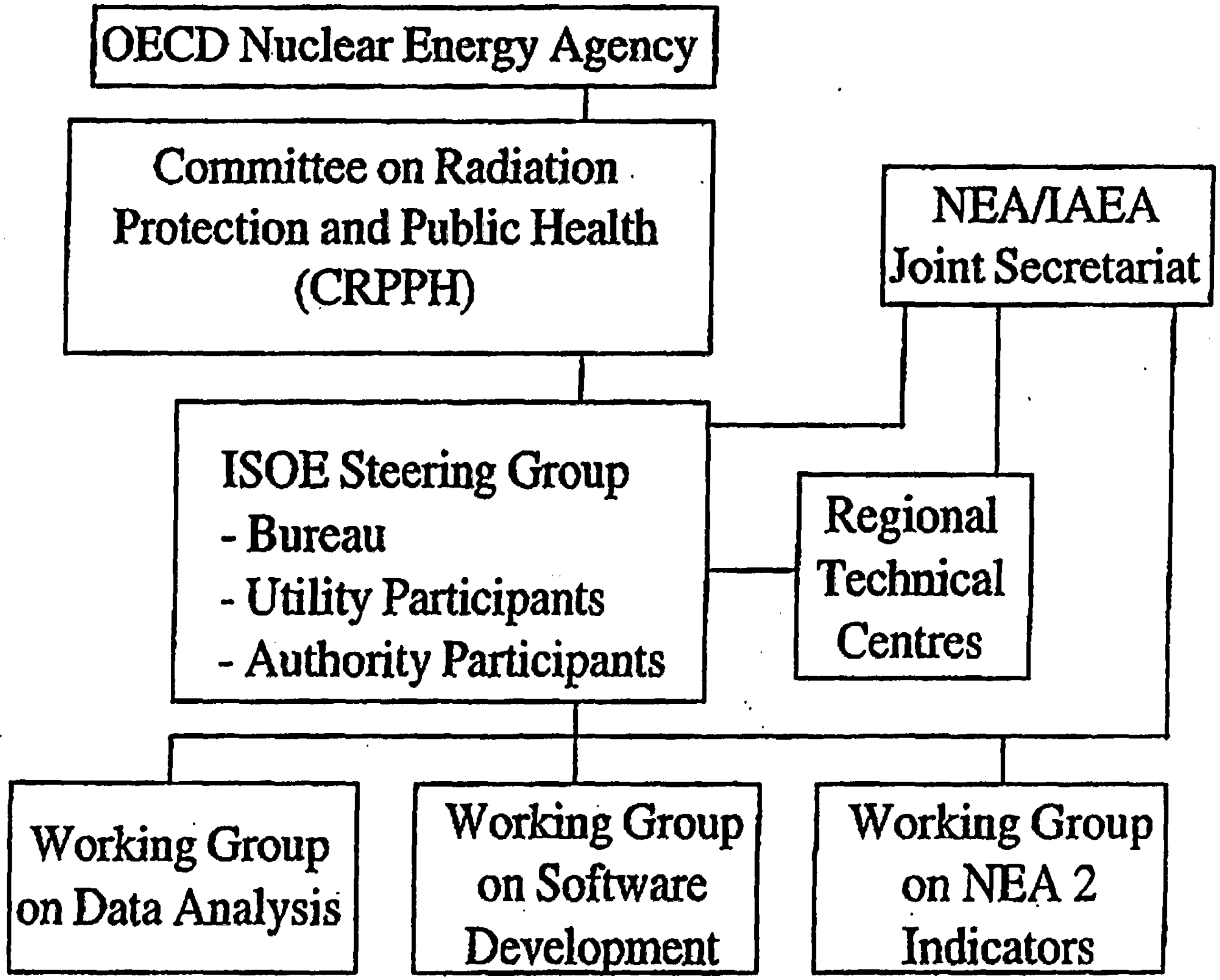
List of Figures

List of Attachments

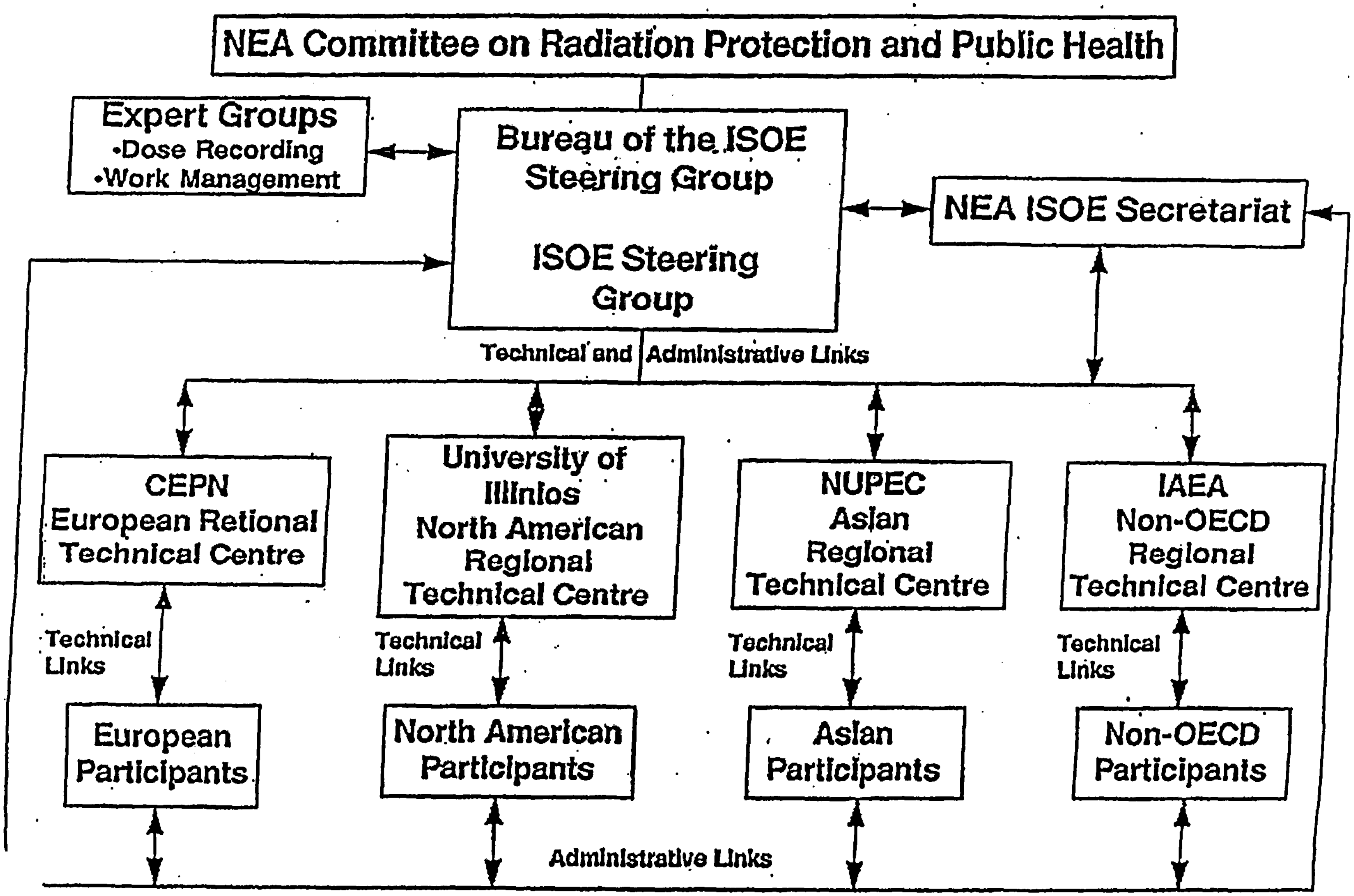
Reference:

Occupational Exposures at Nuclear Power Plants, 1986-1996, NEA, 1998
Work Management in the Nuclear Power Industry, NEA, 1997
IAEA/NEA Occupational and Public Dose Studies, NARTC/ISOE

Organisation of The ISOE Programme



The Information System on Occupational Exposure: ISOE



1998 NARTC UTILITY MEMBERS

Country	Utility	Plant Name
Canada	Ontario Hydro	Bruce A - Units 1, 2, 3, 4 Bruce B - Units 5, 6, 7, 8 Pickering A - Units 1, 2, 3, 4 Pickering B - Units 5, 6, 7, 8 Darlington - Units 1, 2, 3, 4
	Hydro Quebec	Gentilly 2
	New Brunswick EPC	Point Lepreau
Mexico	Comision Federal de Electricidad	Laguna Verde - Units 1, 2
United States	Arizona Public Service Company	Palo Verde - Units 1, 2, 3
	Baltimore Gas & Electric Company	Calvert Cliffs - Units 1, 2
	Boston Edison Company	Pilgrim - Unit 1
	Carolina Power & Light Company	H. B. Robinson, Unit 2
	Commonwealth Edison Company	Braidwood - Units 1, 2 Byron - Units 1, 2 Dresden - Units 1, 2, 3 LaSalle County - Units 1, 2 Quad Cities - Units 1, 2 Zion - Units 1, 2
	Consumers Power Company	Palisades, Unit 1
	General Public Utilities Nuclear Corp.	Three Mile Island - Unit 1 Oyster Creek - Unit 1
	Illinois Power Company	Clinton - Unit 1
	Indiana/Michigan Power Company	Donald C. Cook - Units 1, 2
	New York Power Authority	Indiana Point - Unit 3
	Pacific Gas & Electric Company	Diablo Canyon - Units 1, 2 Humboldt Bay - Unit 1
	PECo Energy	Limerick - Units 1, 2 Peach Bottom - Units 1, 2, 3
	Pennsylvania Power & Light Company	Susquehanna - Units 1, 2
	Southern California Edison Company & San Diego Gas & Electric Company	San Onofre - Units 1, 2,3
	South Carolina Electric & Gas Company	V. C. Summer - Unit 1
	Texas Utilities Electric	Comanche Peak - Units 1 & 2
	Wisconsin Electric Power Company	Point Beach - Units 1 & 2

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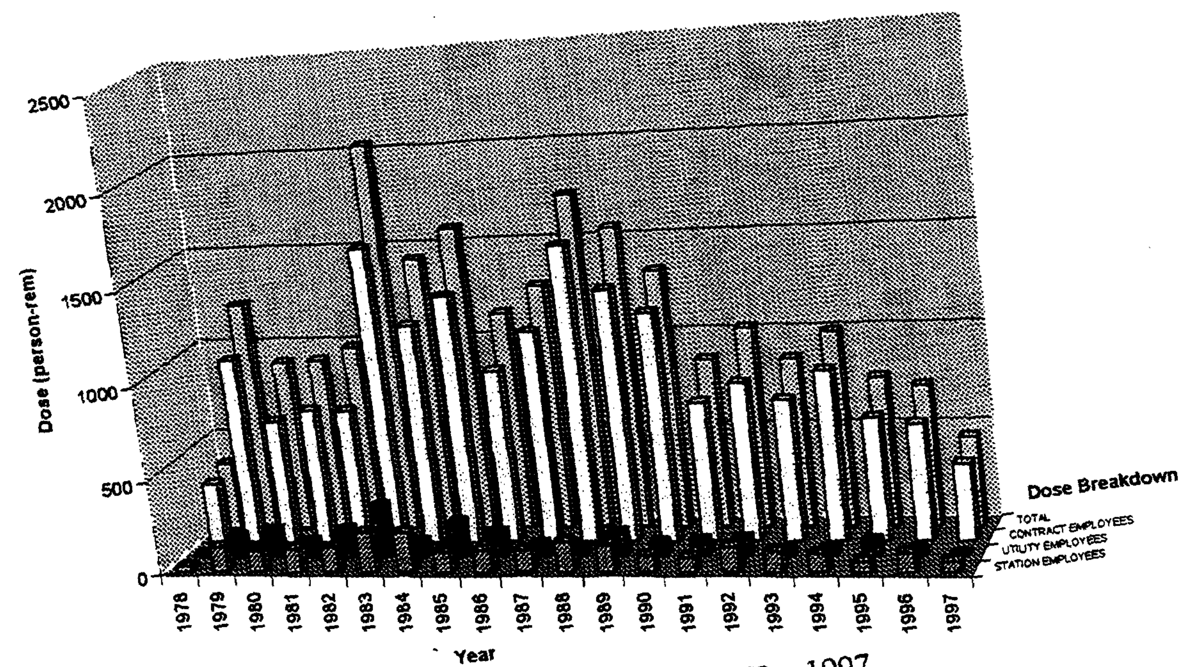
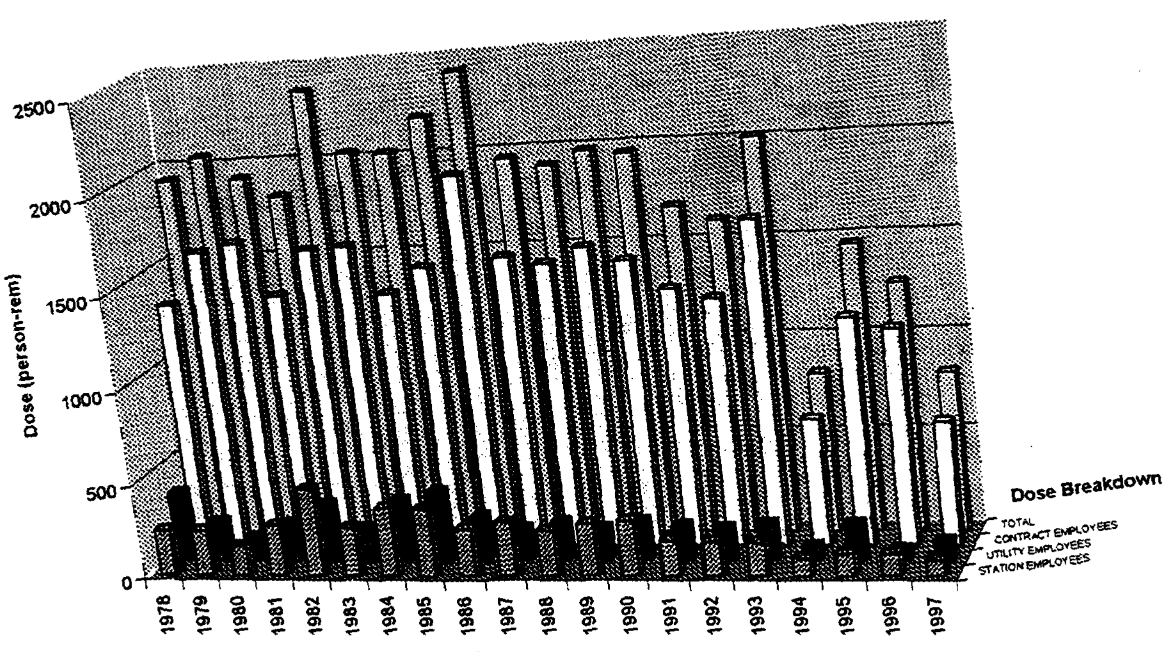
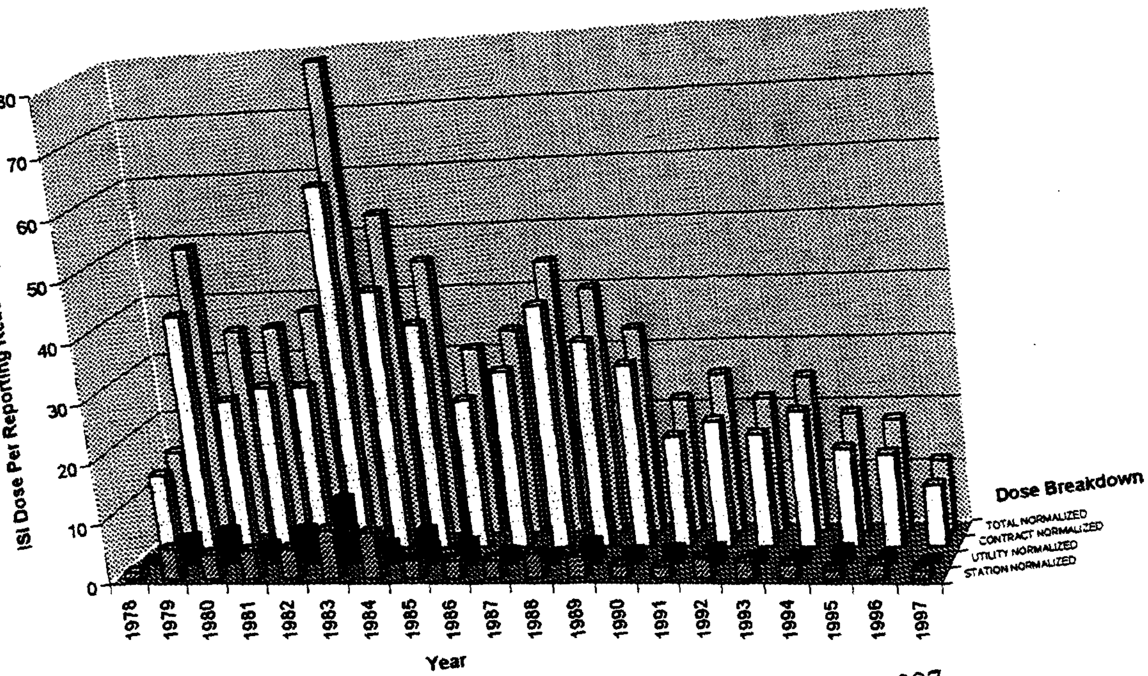
ISI and Special Maintenance graphs



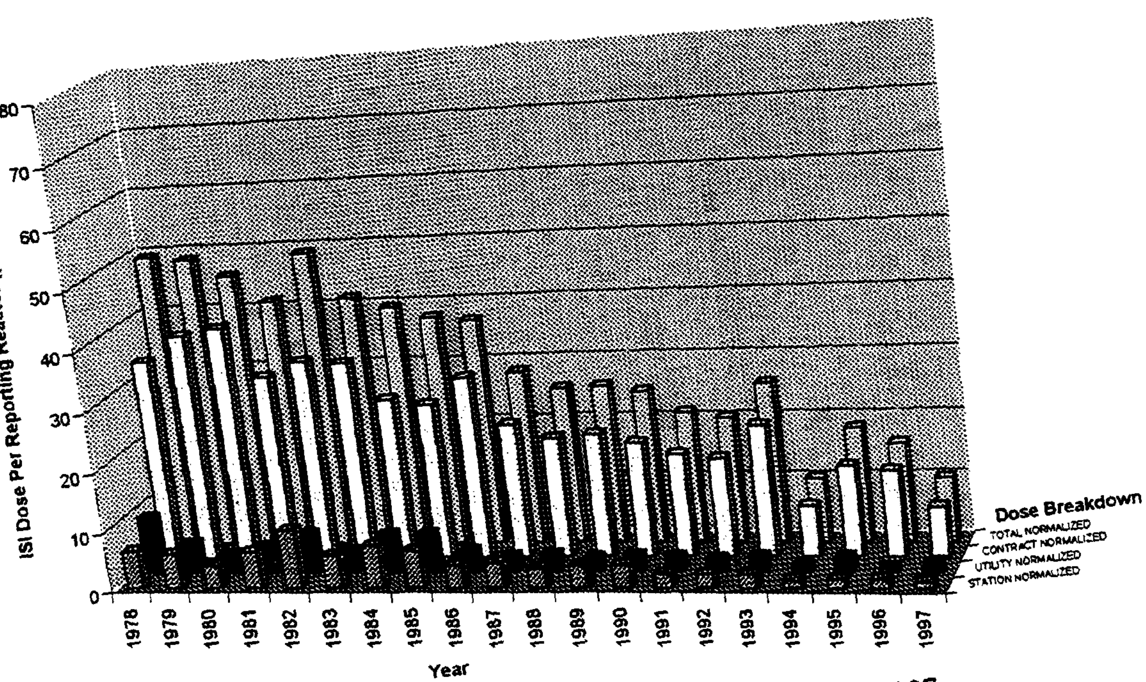
BWR ISI Dose 1978 – 1997



PWR ISI Dose 1978 - 1997



Normalized BWR ISI Dose 1978 – 1997



Normalized PWR ISI Dose 1978 – 1997

Utilizing A Decision Framework With DandD Models and Parameter Analyses For License Termination

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Abstract

A decision methodology and software tools have been developed by Sandia National Laboratories and the Nuclear Regulatory Commission (NRC) to support implementation of the dose assessment requirements in NRC's Radiological Criteria for License Termination rule (10 CFR Part 20 Subpart E; NRC, 1997). The decision process provides a logical and consistent framework that supports licensee planning of decommissioning activities and NRC review of license termination requests. The decision framework includes the entire range of dose modeling options a licensee may utilize, from NRC prescribed screening to complex site-specific models. The Decontamination and Decommissioning (DandD) software package provides a user-friendly analytical tool that implements the NRC screening methodology, allowing licensees to convert residual radioactivity contamination levels at their site to annual dose. The screening methodology is an integral part of the larger decision framework, allowing and encouraging licensees to optimize decisions on choice of alternative actions at their site, including collection of additional data and information.

Introduction

The decision framework and software tools, developed to implement an optimized approach to license termination dose assessments, site characterization and remedial actions at sites covered under the U.S. Nuclear Regulatory Commission's (NRC) 10 CFR Part 20 [NRC, 1997], are summarized in this document. The framework has been generalized to be used throughout the site decontamination, decommissioning, and license termination process, at the full range of potential sites. The screening tools are designed for generic analyses and maybe applied to any site with only limited justification and site specific information. Model comparisons and test cases are being conducted to evaluate the models and framework. The model comparisons and test cases are being used to refine the tools and framework and to provide information for developing detailed guidance documents.

D&D Decision Framework

The decision framework is designed for coordinated use by the licensee, NRC, and other stakeholders. By doing so, the process allows the licensee to:

- synchronize planning efforts with NRC
- define site characterization activities that are directly related to regulatory decisions,
- optimize site characterization, remediation, and land-use restrictions decisions based on cost and time, and
- elicit other stakeholders' input at crucial decision points.

The use of the framework in a coordinated effort streamlines the process of coming to closure on decisions and provides a sound technical basis for those decisions.

The framework provides a comprehensive approach for treating the uncertainty associated with contaminated sites, including quantification, propagation, and reduction of uncertainty.

The framework and methodology have the following attributes:
- ensures that the NRC's, the licensee's, and other affected parties' efforts and expenses are commensurate with the level of risk posed by the site;
- incorporates treatment of uncertainty that ties data collection activities directly to the regulatory dose-based performance objectives

The decision framework is shown in Figure 1 using a generalized flowchart. To begin evaluation of a site within this framework, the licensee and NRC would evaluate compliance based only on existing and available data and information. If compliance can be demonstrated based on this information, then there is no need to proceed further and the site's license can be terminated. If compliance cannot be demonstrated with this initial set of information and analyses, then the licensee would proceed with identifying the optimal license termination strategy by looking at their options, including uncertainty reduction, and proceeding with the best alternative. The framework provides a logical, integrated approach for assessing and demonstrating compliance, providing documentation, and involving concerned parties.

In order to conduct dose assessments in support of decommissioning and license termination, a number of initial activities must be completed. Existing information on the site must be gathered to understand the general nature of site contamination and the physical systems and processes that the site represents (Step 1). Next, licensees must determine potential future human activities and future states of the physical system (scenarios) related to the site that could impact human health due to existing contamination, and then understand physical processes related to the potential pathways for transport of contamination and exposure of radionuclides to the environment and human receptors (Step 2). With an understanding of the physical system and potential human activities, one can then develop conceptual models of the site (Step 3).

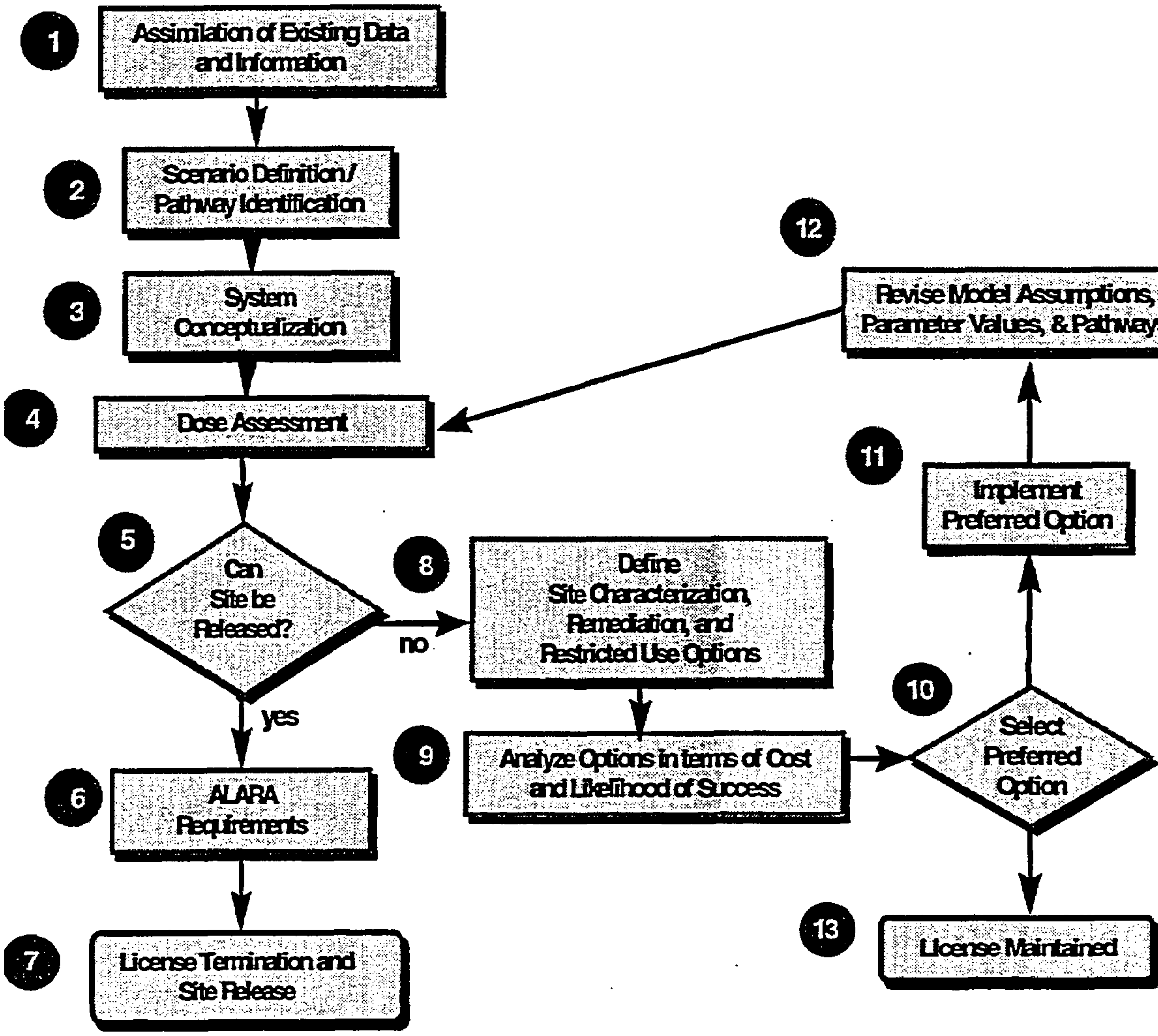

Figure 1: D&D General Decision Framework

Those conceptual models are translated into mathematical models and implemented in corresponding analytical or numerical models and solved by using computer codes. The objective is to calculate a dose to humans (Step 4) which is then compared with regulatory performance objectives (Step 5) to assess whether the site complies with requirements. If the licensee can confidently demonstrate that the site complies with the regulatory performance objectives, they would then proceed to evaluate whether any additional activities should be conducted to decrease doses to as low as reasonably achievable (Step 6). Upon completion of the ALARA analyses and perhaps implementation, the site license can be terminated and the site released. If under Step 5, the NRC and the licensee cannot demonstrate with confidence that the regulatory performance objectives can be met, then the licensee would look at alternative actions for demonstrating compliance with an acceptable level of certainty (Step 8). These options may include the collection of additional information to reduce uncertainty in the current analysis and hence defend lower calculated doses, remediation of parts or all of the site, setting land use restrictions for part or all of the regulatory time period, or any combination of these. After the viable options are defined, they are then analyzed in terms of cost of the action, how likely the action is to be successful, how long it will take to implement the action, and perhaps other important factors (Step 9). Based on this analysis, the preferred option is selected (Step 10) and carried out (11). Based on the resulting modification to the site or the information about the site, the analysis is refined accordingly (Step 12) and compliance is reassessed.

The framework is designed such that the level of complexity and rigor of analysis conducted for a given site should be commensurate with the level of risk that the site poses. Although all sites are expected to step through Steps 1 through 7, the amount of work that goes into each of these steps should be based on the expected levels of contamination and the health risks they pose. For example, a site with a sealed source would obviously not be expected to conduct calculations that are the same complexity as a site with extensive soil and groundwater contamination.

In this framework, all sites may start at the same level of very simple analyses, but it is expected that only certain sites would progress to very complex dose assessment and option analyses. Some sites may not need to conduct any option analyses and some sites may need to evaluate a limited set of relatively simple and inexpensive options. For example, a site with a contained and well-characterized source of contamination that is obviously simple to remove would not require extensive analysis of alternative data collection and remediation options. On the other hand, a site with potentially high levels or extensive contamination may use this process to analyze a variety of simple and complex data collection and remediation options to define the best decontamination and decommissioning strategy.

At the highest level, this process is organized based on logical steps through the decision making process.

This framework implements a connected information flow process, each step generates and provides the foundational information for carrying out the subsequent step.

The process that a licensee follows as they proceed through each step may depend on the release criteria defined in 10 CFR Part 20. That is, licensees will have different options, constraints, defense, administrative, and technical requirements for unrestricted and restricted releases.

Data are used to support scenario definition, pathway identification, conceptual model development, model assumptions and model parameter values in Steps 2 and 3. The licensee has 3 options in this analysis:

- use the NRC developed, default models and parameter values,
- use the default models and site-specific parameter values, or
- develop and defend site-specific models and parameter values.

If the default models and parameter values are used, only source data are required. Additional information is required to support and defend the system conceptualization if alternative models or parameter values are used. Potentially useful information include: processes that utilized the potential contaminants, releases and mitigative actions, hydrologic conditions (soil moisture content, conductivities, depth to groundwater, hydraulic gradients, hydraulic conductivities), soil type and texture, geochemical conditions (Kd, pH), atmospheric conditions (annual averages or time and date specific conditions), and geology (unconsolidated sediments, fractured rock).

A key point of this framework is that new site data collection does not take place until Step 12. New data collection is deferred until the data that would make a difference in decision making and are cost effective to collect can be defined through cost/benefit and data-worth activities (Steps 8, 9 and 10). Otherwise, money may be spent on collection of superfluous data.

Step 1
The licensee gathers and interprets all pertinent and legitimate *existing* site data and other relevant information that can be used to define characteristics of the residual radioactive contamination at the site. In defining the residual contamination, all existing information on the possible contaminants, their amount, location, release, removal, existing concentrations (in soil, building materials, and groundwater) should be evaluated. Where data are unavailable, the licensee may estimate the potential mass of contaminants based on initial inventories (mass balance approach), and the existing spatial distribution of those contaminants based on the processes involved in generating the original materials (e.g., ore processing, contained source, laboratory analyses) and the potential release mechanisms. Generic methods for estimating the model source term given different types and amounts of information about the potential contamination are being developed. Use of the DandD code requires that spatially distributed contaminants be represented with a single concentration. The uncertainty in the extent and amount of residual contamination for each substance will depend on the amount and variability of the data. This uncertainty should be represented or bounded in the later dose assessment in order to evaluate the worth of collecting additional data about the residual contamination (Step 3). The uncertainty in the extent and amount of residual contamination can be accounted for in the dose assessment by employing assumptions about the source magnitude and distribution that are consistent with the framework. It provides a mechanism for evaluating which exposures are of concern and which uncertainties are controlling the results. In this case one of the uncertainties is the distribution of the contaminant mass.

Step 2
In this step, the licensee defines potential human activities and identifies migration and exposure pathways that need to be considered for dose assessment for their decommissioned site. For the purpose of implementing 10 CFR Part 20, scenarios are defined as plausible alternative patterns or sets of human activities and future states of the physical system under study. As such, scenarios provide a description of the plausible future land uses, human activities and temporal evolution of the natural system. For any defined scenario, contaminants have the potential to migrate through various environmental media and to expose a human receptor through a number of physical processes. These migration and exposure processes represent the pathways through which the contaminants move from the source to the receptor. The scenarios of human activities and the pathways of exposure provide the foundation for developing

the conceptual models (Step 3).

The licensee has 3 options in defining scenarios and identifying pathways:

1. using the default models and parameter values (generic screening),
2. using the default models with site-specific pathways and characteristics or
3. developing site-specific models, scenarios and pathways

Site specific pathways and scenarios may be defined. However, these definitions must be defensible and should be consistent with the iterative approach defined in this guidance document. That is, the simulated dose should decrease with each iteration if the scenarios and pathways are changed based on additional site-specific information. Each of the general site release conditions (unrestricted and restricted) involves potentially different considerations with respect to applicable human activities. However, the first time through the decision process the NRC requires that only unrestricted release options be evaluated and that unrestricted release is the preferred option. Scenarios and pathways for restricted release are identified and evaluated in Steps 8 and 9.

The following scenarios from NUREG/CR-5512 [Kennedy and Strenge, 1992] are acceptable with or without site specific data or analyses:

- Residential Farming
- Building Occupancy

The DandD software tool implements NRC's screening methodology and when used with the default parameter values provides a defensible dose assessment. The default parameter values are based on a probabilistic analysis that incorporates the uncertainty in the parameter values and provides information on data worth given the site-specific source term.

Step 3

System Conceptualization, as defined here, includes conceptual and mathematical model development and assessment of parameter uncertainty. The conceptual model is a description of the physical system, the processes transporting contaminants from the source to the receptor, controlling the concentration and location of the contaminants and the location and behavior of the receptor. The mathematical model is a set of equations that can be used to estimate the behavior of the described system. The parameters are defined by the mathematical model. The parameter values and uncertainty in those values for the modeled system are used to represent the uncertainty in the behavior of the system. System conceptualization is the process of systematically evaluating the level of uncertainty associated with a specific site and the quantification of that uncertainty. In order to manage the treatment of uncertainty associated with dose assessment at a given site, the four steps of scenario definition, pathway identification, model development, and assessment of parameter uncertainty are treated as a hierarchy.

As with the pathways, conceptual and mathematical models have been defined for the NUREG/CR-5512 methodology (Kennedy and Strenge, 1992). If the default models and parameter values are used, the licensee would only be required to provide information to defend the parameterization of the source term.

If site-specific models are developed (either through changes to the default parameter values, model assumptions or development of new models), then the licensee must defend the model and parameterization of the system.

Step 4
In this step, the licensee calculates potential doses using mathematical representations of the conceptual models. This step involves the execution of the numerical model(s) that implement the mathematical equations and provide the basis for (1) assessing compliance with the individual dose criteria and (2) an analysis of the impact of uncertainty in models and input parameters on the model output. In doing so, this step includes the propagation of uncertainty in parameters through exposure models and should provide a quantitative representation of the uncertainty in the dose given those models and parameters.

NRC has implemented the default pathways, model assumptions and parameter values in the DandD code.
The licensee is encouraged to actively work with the NRC during this step to evaluate the appropriateness and adequacy of the analyses before moving on and expending resources on follow on steps.

Step 5
The results of the consequence analysis are evaluated to determine if the site meets the criteria for release based on the information available to this point. In the *initial* set of analyses, the evaluation must be made with regard to unrestricted release. A defensible decision about release at this point is possible if the approach used to define residual contamination, scenarios, pathways, conceptual and mathematical models, and data outlined in Steps 1-3 above has been followed. That is, the analysis should be based on existing information only, should completely account for uncertainty, and the NRC and other affected parties should have been involved in each step.

On the *initial* pass through Step 5, the possible outcomes and decisions that exist are:

- the simulated dose is less than the regulatory criteria under §20.1402 and the site can be released as unrestricted (move on to Step 6 to address ALARA requirements and administer license termination and unrestricted release); or
- the simulated dose exceeds the regulatory criteria and possible follow-on actions need to be defined and evaluated (proceed to Step 8 and define alternative actions).

On *subsequent* passes through Step 5, the possible outcomes and decisions that exist are:

- the simulated dose is less than the regulatory criteria under §20.1402 and the site can be released as unrestricted (move on to Step 6 to address ALARA requirements and administer license termination and unrestricted release); or
- the simulated dose is less than the regulatory criteria under §20.1403 and the site can be released as restricted (move on to Step 6 to address any necessary additional ALARA requirements that were not addressed under Steps 8-11 and administer license termination and restricted release); or
- the simulated dose exceeds the applicable regulatory criteria and possible follow-on actions need to be defined and evaluated (move to Step 8 to define alternative actions).

It is critical to note that if compliance is not demonstrated at this point, DCGLs would not be defined here. If remediation is chosen as the preferred option or part of the preferred option for the follow on action to this step, then DCGLs should be defined under Step 9 and remediation to the DCGLs implemented under Step 11. If compliance is demonstrable, this implies that either the concentrations prior to any remediation result in acceptable doses or the site has been cleaned to concentration levels that result in acceptable doses.

Step 6
If the licensee has defensibly demonstrated compliance with the unrestricted release individual dose criteria under §20.1402, then additional ALARA actions at this step should include typical good practice efforts (e.g., floor and wall washing, removal of readily removable radioactivity in buildings or soil areas). The licensee would not be expected to conduct additional modeling analyses to evaluate ALARA options under these circumstances.

The licensee is encourage to actively work with the NRC to discuss, define, and concur on alternative ALARA actions under this step prior to implementing any actions.

Step 7
If the site meets the release requirements and the ALARA analysis and implementation is comple, the decommissioning plan will be developed or modified at this point to reflect this action and then submitted to the NRC for approval of release of the site. The licensee will also submit documented dose analyses, any necessary ALARA documentation, and NEPA documentation. The NRC determines if the documentation surveys and public involvement were sufficient.

This step represents an action justified by the prior steps in the framework and is the end point for sites being released as either unrestricted or restricted.

Step 8
If after the evaluation of the site it has been determined under Step 5 that based on existing information, the site cannot be released, then the options for site characterization activities, remediation strategies, restrictions on the use of the site, and combinations of these should be defined. Generally, the options that exist are activities that:

- reduce uncertainty (information/data collection)
- reduce contamination (remediation), or
- limit exposure (land-use restrictions).

The licensee is encouraged to actively work with the NRC during this step of the framework to evaluate the appropriateness and adequacy of the analyses before moving on and expending resources on follow on steps.

It is expected that only certain sites would progress to very complex dose assessment and option analyses, whereas the option analysis may be relatively simple and straightforward for other sites. Sites that pose minimal risk will likely only need to evaluate a limited set of relatively simple and inexpensive options. For example, a site with a small, contained source of contamination that is obviously simple to remove would not spend extensive work analyzing large suites of alternative data collection and remediation options. The same may be true for certain sites that pose significant risk, but where the options for proceeding forward are reasonably limited and straightforward. However, the cost may not be insignificant. On the other hand, a site with high levels of contamination that are widely distributed can use this process to analyze a variety of simple and complex options to define the most effective and cost-efficient decontamination and decommissioning strategy.

Combinations of two or more options may provide the optimal solution. For example, the licensee may choose to collect data to reduce uncertainty in the distribution of contaminant to reduce remediation costs. Another example is the application of land-use restrictions to some portions of the site and remediaiton and unrestiricted release to other portions of the site to reduce long-term maintenance, monitoring and assurance costs. Generically, examples of combined alternatives include:

- site characterization combined with remediation, followed by unrestricted release,
- a series of site characterization activities followed by unrestricted release,
- a series of site characterization and remediation activities followed by unrestricted release,
- remediation combined with land-use restrictions followed by restricted release,

The licensee is encouraged to involve all stakeholders, including the NRC, at this step to help identify options.

Step 9

In this step, each of the options is analyzed in terms of cost and likelihood that the activity will be successful. In order to analyze the latter of these, an analysis of the potential outcome will need to be performed for each of the options. Depending on the option, this consequence analysis could be very simple (e.g., the option is complete remediation and the consequence is effectively restoring the system to its original state or to an acceptable state) to as complicated as refining and expanding the analysis in Step 4. Note that the consequence analysis should address the uncertainty associated with the potential outcome of each option. Thus the desired final outcome of this consequence analysis is a determination of the likelihood or probability that employing a given option will result in meeting a specific performance objective. For each option, the cost and time required to complete that option is also estimated. Some detail is given below.

If the activity is successful, the calculated dose is acceptably low. For example, if the licensee chose to spend money to collect additional information on specific soil properties and remediate a small portion of the site, and as a result were able to demonstrate that the dose was below 25 mrem, then their activities would have been successful and the site could be released as unrestricted.

In addition, there exists a cost associated with conducting the activity. If the activity is successful, then the overall process is effectively done, no follow on activities are necessary, and no other significant costs would be incurred. On the other hand, if the activity is unsuccessful, the total cost will be the cost to conduct the activity plus the cost to conduct any necessary follow-on activities to get the dose to an acceptable level.

Note that *actual* success or failure would not be realized until the completion of Step 11. Therefore, the exercise being conducted under this step is defining what would be required for success under the given activity, evaluating the chances of that success occurring, and evaluating the cost that would be incurred if the activity were successful and the costs that would be incurred if the activity failed.

The probability or likelihood of success is defined for the different options as follows for:

- Site Characterization / Information collection, it is the likelihood of being successful in collecting the data that is needed to reduce the uncertainty in the output to change from an unacceptable dose to an acceptable dose (within specified constraints of time and cost)
- Remediation, it is the likelihood that contamination will be reduced to a level that will result in acceptable dose (within specified constraints of time and cost) and
- Land-Use Restrictions, it is the likelihood that a specified restriction will be durable and effective in reducing exposure for the time period required by the NRC (within a specified cost).

Based on these definitions, success is analogous to site release. Therefore, the likelihood of success would be the same as the likelihood of site release.

An example of how the options could be organized is provided in Table 2.9.1 (for a set of hypothetical alternative actions). For certain sets of alternatives, the decision regarding the preferred option will be obvious. For example, a low cost, high probability of success option will always be selected over a high cost, low probability of success option. However, this may not always be the case and more consideration of options may be necessary.

The decision as to which option to select may be the joint responsibility of a number of parties, including the licensee, the NRC, and perhaps other stakeholders. The decision process could include other factors in addition to probability of success and cost (e.g., time to complete the activity, environmental justice, protection of cultural resources, etc.). These other influencing factors can be articulated and presented as part of the results of each of the options defined in the options analysis table. Consequently, the result of Step 9 is a logically represented list of options, and the corresponding cost, likelihood of site release, and other important considerations given that the option is pursued. This analysis will provide the information necessary in Step 10 of the decision framework.

Step 10

At this step, the decision makers choose the option that will be pursued given the factors described in Step 9 of this framework. This step represents the second major decision point in the framework.

Step 11

This step is where actual reduction of uncertainty (site characterization), contamination at the site (remediation), or imposition of land-use restrictions (restricted use enacted) would occur. For site characterization activities and remediation activities, because there is some uncertainty whether the activity will be successful, there is a possibility that after the action is conducted, follow-on actions will have to take place.

If the activity is successful, there will exist a specific outcome in terms of the calculated dose being acceptable. For example, if the licensee chose to spend money collecting additional information on specific soil properties and remediating a small portion of the site, and was subsequently able to demonstrate the dose will be below 25 mrem, then their activities were successful and the site can be released without restrictions. In other words, if the models and treatment of uncertainty are defensible (see Steps 2 - 4), then this case would be equivalent to meeting the dose criteria for license termination.

If, after the activity is conducted, the resulting simulated dose cannot defensibly be shown to meet the dose criteria, then the activity is considered to be unsuccessful. In either case, the process proceeds to

Step 12 to revise the conceptualization of the site in terms of modifications to parameter values, model assumptions, pathways, and scenarios.

Step 12
If a data collection activity was performed in Step 11, then depending on the results, the new data can be used to eliminate potential pathways, refute certain model assumptions, justify new parameter values, refine parameter distributions, or refine the estimated extent and amount of residual contamination. Following revisions, the process returns to Step 4, dose assessment analyses are performed again, and the site is evaluated against the appropriate release criteria. If a remedial action was performed in Step 11, then a final status survey will be conducted to confirm the efficacy of the remediation. The characterization of the source is then modified appropriately and the process returns to Step 4 for final dose assessment analyses. If institutional controls are put in place as part of the action performed in Step 11, then the pathways and scenarios are modified as appropriate and the process returns to Step 4 for dose assessment analyses under the appropriate dose criteria.

Step 13
At this point, the decision makers have determined that the site is not likely to meet the unrestricted or restricted release criteria or that additional evaluation or remediation is too costly or will be too lengthy and no further assessment or site characterization will be performed. In this case the NRC and the licensee determine that the license may be maintained indefinitely until other future options become feasible through new technology development or other resources. Under these circumstances, this option represents only a temporary holding place and is not truly an end point. Eventually some action will be required to release the site.

DandD Release 1.0

The Decontamination and Decommissioning (DandD) software package, developed by Sandia National Laboratories for the Nuclear Regulatory Commission (NRC), provides a user-friendly analytical tool to address the technical dose criteria contained in NRC's Radiological Criteria for License Termination rule (10 CFR Part 20 Subpart E; NRC, 1997). Specifically, DandD implements the NRC's screening methodology, allowing licensees to convert residual radioactivity contamination levels at their site to annual dose. DandD is consistent with both 10 CFR Part 20 and the corresponding implementation guidance currently under development by NRC. NRC's screening methodology employs generic scenarios, fate and transport models and default parameter values. The models and default parameter values were developed to support decisions to release certain sites given only information about the level of contamination. Therefore, a licensee has the option of specifying only the level of contamination and running the code with the default parameter values, or, in the case where site specific information is available, altering the appropriate parameter values then calculating the dose. DandD implements the screening models for the residential and building occupancy exposure scenarios. The screening methodology is an integral part of the larger decision framework, allowing and encouraging licensees to optimize decisions on choice of alternative actions at their site, including collection of additional data and information. The default parameter values are base on a systematic analysis of the uncertainty in the key parameter values given only information about the source of contamination and a minimal amount of hydrologic data to support the use of the models at a specific site. The underlying probabilistic assessment of parameter uncertainty provides the basis for optimizing the analysis and decision process.

For the simplest level of analysis, the user is required to provide a minimum amount of site-specific information. In general, only information about source concentration is required for screening. This level of analysis is automated in DandD, and therefore provides certain licensees with a simple and cost-effective method to demonstrate compliance using a minimum amount of information. This level of analysis implements the generic scenarios and models from NUREG/CR-5512, Volume 1 (Kennedy and Strenge, 1992), and uses deterministic values for all model parameters through a systematic process of assessing the variability of each parameter across all sites and then defining default values that produce generic dose estimates that are unlikely to be exceeded at any real site.

The default parameter values for the NUREG/CR-5512 models (which have been updated and are implemented in DandD) are based on probability distributions representing the variability across all sites in the country. As a consequence, the licensee would likely need little supporting information to defend significant changes to the parameter values. For example, the probability distribution used in defining the default values for the depth to groundwater for the NUREG/CR-5512 residential scenario models is based on the variety of possible hydrogeologic settings. Many sites should be able to defend a greater depth to groundwater than the default value. This approach of moving away from the generic default values used in the NUREG/CR-5512 modeling could be used by all sites until the point that further reduction in simulated dose would require model changes. This would require the licensee to step away from using DandD. At that point, new models and parameter values would have to be developed and defended by the licensee. Model changes should lead to less conservative models and lower doses with each iteration, because the NUREG/CR-5512 models are designed to be inherently conservative, however the conservatism of presently-used models has yet to be fully evaluated or quantified.

Default Parameter Set

The process used for determining the default parameter set included:

- identifying "key" model parameters
- update the characterization of uncertainty in those parameter values
- analyze the probability of underestimating dose (an inversion) given the uncertainty in key parameter values (for the generic screening models, assumptions and scenarios).
- select a set of default parameter values based on a specified upper bound on the probability of underestimating the dose (Pcrit) and minimizing the number of default parameter values that are set at extremes (maximize joint exceedence probability).

The behavioral and metabolic parameters represent the characteristics of the generic critical group, as such, the defaults are set at the mean value to represent the average member of the screening group (building occupant or residential farmer). The uncertainty in the physical parameter values is represented using probability distribution functions (pdfs), then analyzed using Monte Carlo simulations. The output of each model run is used to create a distribution of the simulated dose as a function of a unit concentration for each isotope in the DandD library.

The criterion that is used to establish Pcrit is: if new site information is added, the estimated dose is very likely to decrease. Since the source probabilities are not known, all sources must be evaluated in the parameter analysis, and Pcrit represents an upper bound on the probability of an inversion (see Figure 2). This process allows selection of a single set of generic parameter values for screening that is consistent with the decision framework. Currently, there is a Monte Carlo version of DandD in development. The Monte Carlo version will utilize the same models, Pcrit, and parameter uncertainties as the default parameter analysis. However, since the source term will be specified (known) and there will not be a single set of default parameters, the results of the screening analysis will be consistent, dose-based screening criteria for all source terms. The new tool will provide additional information on data worth given the site-specific source term.

The framework and software tools are being tested using existing NRC sites. This testing will provide the basis for refining the models, framework, and guidance on implementing the framework. The test cases will provide useful examples for other licensees to learn from and improve the decision process.

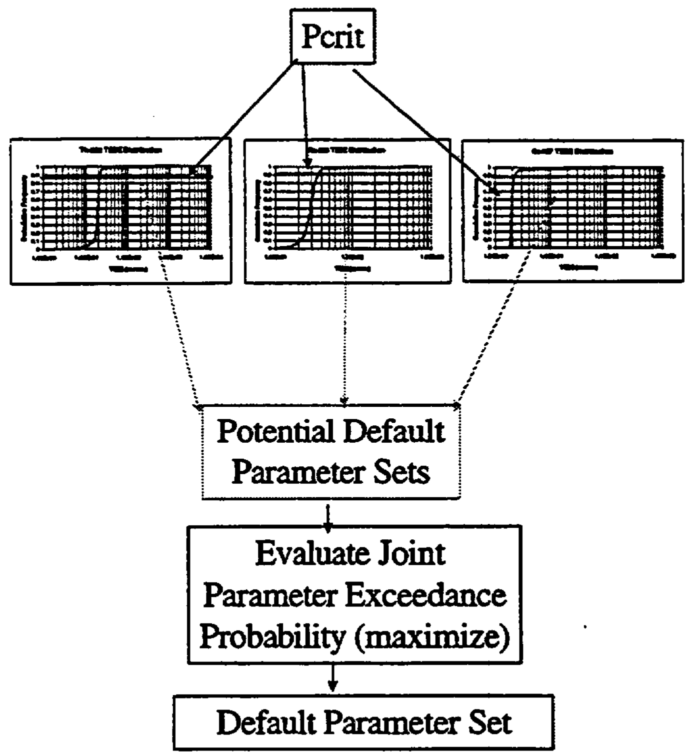

Figure 2: Default Parameter Analysis Process

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NRC PERSPECTIVE ON SELECTED MATERIALS INTEGRITY ISSUES: SESSION OVERVIEW

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Predicting and maintaining the physical integrity of components in service can present a challenge in many arenas in which the USNRC has regulatory authority. This physical integrity can be challenged by aging effects, irradiation effects, fabrication defects, and repair defects, as well as other in-service demands. Ensuring that the components maintain physical integrity is important for maintaining safety and reliability of systems in nuclear power plants, as well as in the safe transportation, storage, and disposal of spent nuclear fuel. In response to the concerns regarding integrity of selected components in service, the Office of Nuclear Regulatory Research (RES) has funded work to evaluate weld cracking and provide a flaw size screening criteria for spent fuel dry storage cask welds. In addition, RES is initiating cooperative research programs to investigate the feasibility of welded repairs in highly irradiated stainless steels and evaluate the integrity of spent fuel and spent fuel casks that have been stored up to 18 years in dry storage.

Introduction

Demands on the physical integrity of a component in service can arise due to the effects of aging and irradiation, defects due to fabrication and repairs, as well as other in-service pressures. The material can reveal such challenges to its integrity through tell-tale indications such as cracking, separation/delamination, radiation-induced swelling, void formation, and brittle behavior associated with lack of fracture toughness.

This session addresses materials integrity issues for three examples of technical areas that are of current interest to the USNRC, which highlights the broad range of integrity concerns and the varied and challenging environments in which the materials must perform. Two of the three examples focus on spent fuel dry storage casks, and one example describes weld repair concerns for components in boiling water reactor (BWR) reactor vessels.

Welding of Highly Irradiated Materials

The USNRC began an evaluation of the feasibility of repair welding components in the BWR in-vessel environment as a result of an increased level of activity in the commercial nuclear industry to address generic issues concerning the reactor internals, especially those issues related to repair options. In particular, the BWR Vessel and Internals Project (BWRVIP) had indicated their interest in pursuing repair options for in-vessel components, and were expecting to submit their evaluations of the viability of this approach to the NRC staff in the near future.

It was readily apparent to both the BWRVIP and the NRC staff that little field experience was available in the area of underwater welding of highly irradiated materials, and research in this area was mostly limited to applying repair welding techniques to specimens irradiated to simulated fusion reactor flux and fluence levels. Moreover, due to the nature of simulating the changes to the material that resulted from the neutron exposure, there was some conflicts in the results of the data gathered by the researchers in this area.

The primary complication to welding materials such as stainless steels that have been exposed to a high fluence is the cracking that results from the growth and coalescence of helium bubbles formed from the interaction of the neutrons with boron and nickel present in the stainless steel. The helium that forms is virtually insoluble in the stainless steel, and the heat input from the welding process causes the bubbles to grow, and failure results when the residual tensile stress state adjoining the weld over stresses the remaining ligaments between the bubbles.

Some previous research in this area had utilized a method of "doping" the material with helium, rather than irradiating the material, and the results from this method could not be easily correlated with the results gained from irradiating the material. Very little of the previous research data on welding was gathered at helium levels of interest to the commercial nuclear industry, so there was uncertainty as to if conventional or modified welding techniques could be used successfully in that regime. Further adding to the uncertainty of repair welding in locations in-vessel was the uncertainty of knowing a priori what helium levels could be found in the components or locations that needed repair. It was not apparent to the NRC staff that the helium content of the material could be reliably predicted by computing thermal fluences for in-vessel locations or by any other means except for actually measuring the helium from a small sample taken from that location.

What was also readily apparent to both the BWRVIP and the NRC staff was that resolving the uncertainties inherent in weld repair for highly irradiated materials would be a costly venture for either entity. Because of the opportunity for gathering data in an area in which data is extremely scarce but expensive to produce, the BWRVIP and the NRC staff saw the opportunity to participate in a cooperative research endeavor through the Memorandum of Understanding between NRC and EPRI on Cooperative Nuclear Safety Research. EPRI and the NRC have agreed to cooperatively investigate the feasibility of welded repairs of highly irradiated stainless steels in BWRs, and the framework and the objectives of the program will be discussed in the first talk of the session.

Cooperative Research on LWR Spent Nuclear Fuel in Dry Storage

As a result of informal discussions with EPRI and DOE at scientific meetings and forums discussing spent nuclear fuel issues, the NRC staff became aware of an opportunity to assess the materials integrity of spent nuclear fuel and the spent fuel casks used for a dry storage demonstration project at the Idaho National Engineering and Environmental Laboratory (INEEL). Because of the significant level of interest that is shared by EPRI, DOE, and the NRC on the integrity of waste packages and spent nuclear fuel in dry storage, and any credible degradation processes operating on the cask and fuel, the three organizations agreed to pursue a cooperative research effort to evaluate the behavior of spent fuel and dry cask internals for casks that have experienced extended storage periods. The objectives of this program and background on the DOE spent fuel dry storage cask demonstration project will be given in the second talk of this session.

This project is of considerable interest to the NRC staff, most notably as license extensions from the NRC will be needed to extend dry cask storage in Independent Spent Fuel Storage Installations (ISFSIs) beyond 20 years. As with the previous topic, welding of highly irradiated materials, this is a technical area in which there is very little readily available data on materials integrity over extended periods of time. Again, gathering the data to provide assurance of materials integrity of the casks and spent fuel would be cost prohibitive for any entity undertaking such a project alone, so a cooperative research venture under the Memorandum of Understanding between NRC and EPRI on Cooperative Nuclear Safety Research to collect data on spent fuel and cask performance in long term dry storage is being pursued. It is expected that the data from this program will be used to benchmark predictions made in the past about long term material behavior and performance in this demanding environment.

Materials integrity issues of concern for the storage casks include various forms of corrosion, embrittlement, weld cracks, UV attack of coatings, degradation from hydrogen generation, and others; materials integrity issues of concern for the spent nuclear fuel include various forms of corrosion, creep rupture, hydride formation/reorientation, embrittlement, hydrogen generation, bowing/bending/swelling, as well as others. Within the constraints of available funding, plans are to film the condition of the spent fuel and cask system, and perform destructive analysis on selected spent fuel rods.

Weld Cracking and Flaw Size Screening Criteria for Spent Fuel Dry Storage Casks

Recent experience with dry storage casks in ISFSIs has indicated that one type of these casks is susceptible to cracks in the weld closures for the lid to vessel welds. The third talk of this session will describe the material degradation issues that led to cracking in the welds, and an approach that was recommended to resolve weld cracking in the lid to vessel welds. One of the recommendations from the weld cracking evaluation was to perform an ultrasonic examination (UT) procedure, Time of Flight Diffraction (TOFD) to verify the structural integrity of the weld. The fourth talk will describe an acceptable flaw screening criteria, for flaws that are found from the UT examination of the weld.

As a result of the failure analysis of the cracked welds, the following were thought to contribute to degradation of the integrity of the weld: undocumented weld repairs, moisture in the weld environment, hydrogen induced cracking, and improper fit-up during welding. Both the third and fourth talk discuss how these factors can lead to inadequate welds and suggest techniques for improving welding processes to eliminate these potential problems.

Investigating the Feasibility of Welded Repairs of Highly Irradiated Stainless Steels in Boiling Water Reactors

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ABSTRACT

As reactors age, the availability of repair technology is of more interest. Use of welded repairs within the reactor vessel is desirable due to their durability, strength and broad applicability. Welded repair methods allow refurbishment of components without the installation of costly and cumbersome mechanical devices and the associated design modification considerations. Use of welded repairs however presents its own set of unique issues to be resolved. One major issue is the formation of cracks when welding highly irradiated stainless steels. This cracking occurs when helium, a product of the transmutation of boron and nickel in the presence of neutrons, is present in base materials during welding.

The US Nuclear Regulatory Commission and Electric Power Research Institute/Boiling Water Reactor Vessel and Internals Project (EPRI/BWRVIP) are currently involved in a cooperative effort to evaluate the feasibility of welding on highly irradiated stainless steels in the boiling water reactors. The program will attempt to determine: locations for which repair is feasible, the level of helium that eliminates welding as a repair option, welding techniques that can be employed, and the range of helium concentrations for which each technique is applicable.

A near term benefit is to find a method for determining (and eventually predicting) the helium concentration in locations within the vessel that require repair. An additional benefit of this program will be the opportunity to corroborate predictive neutron flux model results with actual field material samples.

INTRODUCTION

Many boiling water reactor (BWR) internal components are subject to damage due to intergranular stress corrosion cracking (IGSCC) or fatigue. Some of these components are in regions or are configured such that mechanical repairs are not possible or not economical. For these components, a welded repair may be the only possible or most cost-effective solution.

Welded repairs of several components and systems have been successfully performed both dry and underwater in various low-fluence locations through out the vessel, including locations on the steam dryer, feedwater sparger, and core spray T-box. Welded repairs have several major advantages, including not needing plant-specific fabrication or maintaining costly contingency hardware in stock for emergent repair needs during outages. The major disadvantage to performing weld repairs on in-vessel components is the potential for helium-induced cracking in components that have been exposed to high fluences. The problems with welding in-vessel will be discussed in this presentation, and proposed research to reduce the uncertainties and problems associated with welding will be presented.

THE PROBLEM WITH WELDING IRRADIATED STAINLESS STEELS

The Electric Power Research Institute/Boiling Water Reactor Vessel and Internals Project (EPRI/BWRVIP) and the US Nuclear Regulatory Commission (NRC) had both performed literature reviews on the topic of welding highly irradiated materials. In addition, EPRI/BWRVIP has used a calculation approach to determining thermal and high energy neutron fluxes in a typical BWR to estimate the potential for problems that would be caused during welding due to the neutron flux.

The previous research indicated that highly irradiated stainless steels suffer from severe cracking during welding due to gas bubble generation in the weld and heat affected zone and stresses generated in the solidifying weld material. Lightly irradiated stainless steel does not crack during welding and maintains full strength welds. Intermediate irradiated stainless steel can be successfully welded if welded using the appropriate techniques (for example, heat input and the application of stresses during welding both affect the success of the process).

There is tremendous uncertainty inherent in establishing a threshold for the fluence level for which conventional weld techniques would not be successful, due to the scarcity of data on the amount of helium that exists in components in-vessel. The helium exists as small, insoluble bubbles in the component materials due to the cumulative effects of neutron bombardment on materials that contain nickel and boron, and cause failure in the welds due to the rapid growth and coalescence of the bubbles during the welding process. Determining locations where welding is an appropriate approach for repair is the focus of research efforts jointly funded by the NRC and EPRI.

PRODUCTION OF HELIUM IN STAINLESS STEEL

Helium is produced during neutron bombardment principally by two fundamental reactions.

1. $^{10}B + n = {}^{7}Li + {}^{4}He$
2. $^{58}Ni + n = {}^{59}Ni + \gamma$
   $^{59}Ni + n = {}^{56}Fe + {}^{4}He$

Boron (B) is present in stainless steels as a tramp element in a concentration of approximately 10 to 30 parts per million (ppm). Nickel (Ni) is an alloying addition of about 8% in Type 304 stainless steel. The generation of helium (He) by these two processes takes place at different fluence levels.

WELDING ON IRRADIATED STAINLESS STEEL

Helium is insoluble in stainless steel and metals. It stays where it is generated until elevated temperatures are reached (>450 degrees C). The mechanical properties of He containing metals have been shown to be very similar to non-He containing metals. At temperatures near the melting point, the He agglomerates, forming bubbles which grow rapidly. Tensile stresses generated by the welding process in the heat affected zone can result in fractures of the base material as well as the weld. Welding parameters that control the heat input and the stresses can affect the formation of cracks in the welds.

UNKNOWNS AND UNCERTAINTIES OF WELDING HE CONTAINING MATERIALS

One of the problems with welding materials containing He is knowing the He content. Generally, the boron content is not known because boron is present as an impurity element. Also, the accuracy of analysis of boron content is this range is poor. Another difficulty is estimating the He content from the thermal fluence. Thermal neutron fluence is not known accurately outside of the core. Estimating fluence, and thus He, is dependent upon the accuracy of the calculation program and is a function of the time in service and the shielding load in the core. Also, the threshold levels of He needed to result in bad welds are not accurately known. And finally, welding parameters to produce good welds with some He present in the material are not established yet with any assurance.

OBJECTIVES OF THIS WORK

The objective of this research program is to determine levels of irradiation and He content in materials and components of interest in the reactor vessel. The level of irradiation and boron content of stainless steel is not accurately known, so sampling of irradiated stainless steels from in-service reactors is a viable approach. The Jet Pump Riser Brace Pad had been selected as a potential location for sampling in this investigation because the brace to pad weld is not easily fixed by a mechanical clamp. The samples will be analyzed for helium, nickel, and boron and an isotopic analysis may be performed to estimate the neutron irradiation level the component has experienced. This analysis of helium can be very accurately performed by Pacific Northwest National Laboratory using very small samples.

Another objective of this work is to define the limits of irradiated stainless weldability in a BWR based on composition, fluence and He content. To achieve this objective, an "acceptable" weld will have to be defined. One approach to achieving this objective consists of producing welds on weld pads for jet pump risers braces that have been irradiated to different levels of fluence using various welding procedures. Welding may be performed on irradiated mockups of in-vessel components in a laboratory to limit costs of performing work in a working reactor. Welding will be performed using air tungsten inert gas (TIG) or underwater techniques such as flux core or some shielded welding technique to control the environment and produce acceptable welds.

FUTURE WORK

The expected duration of this program is three years starting in fiscal year 1999. Proof of acceptable welds being produced on this component would provide the incentive to look at other areas of the BWR where welding techniques could be a better solution than mechanical fixes or complete replacement.

Cooperative Research on LWR Spent Nuclear Fuel in Dry Storage

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ABSTRACT

Since 1986 a large quantity of spent-fuel has been placed in dry cask storage in Independent Spent Fuel Storage Installations (ISFSIs) at commercial nuclear power stations. NRC license extensions are needed to continue storage beyond 20 years. Information on the long-term integrity of spent-fuel and dry storage casks under dry storage conditions is not currently available to support cask license extensions by the NRC.

The Idaho National Engineering and Environmental Laboratory (INEEL) has been involved in a dry storage test and demonstration program since 1985. Prototypes of several commercial dry storage casks were procured or constructed on-site, and multiple tests and evaluations of the viability of dry cask storage of spent-fuel were conducted. In 1985 and 1986, three casks were tested by placing Westinghouse PWR spent-fuel assemblies in the GNS Castor V/21, Westinghouse MC-10, and Transnuclear TN-24P metal casks. Fuel rods from 48 assemblies were subsequently consolidated into 24 canisters, which were tested in the TN-24P. In 1989, 17 of those 24 canisters were used to test the Pacific Sierra Nuclear VSC-17 concrete cask. Monitoring of cask temperatures was initially performed. Continued routine monitoring consists of visual surveillance of the casks, monitoring of the gas pressure inside the casks, and monitoring of radiation fields around the casks.

Preparations to determine how the spent-fuel and dry cask internals have behaved under extended storage conditions at the INEEL are now being made by Lockheed Martin Idaho Technologies Company, with funding from the NRC, EPRI, and DOE-RW. This research effort will provide enhanced cask monitoring, and will allow for inspection and performance of selected material tests of the spent-fuel and cask internals in the GNS Castor V/21 and the Pacific Sierra Nuclear VSC-17 casks. This program will start in FY-1999 and is planned to continue through at least FY-2001.

INTRODUCTION

Since 1986, commercial nuclear spent fuel has been placed in dry cask storage in Independent Spent Fuel Storage Installations (ISFSIs) at commercial nuclear power stations. The Idaho National Engineering and Environmental Laboratory (INEEL) has been involved in a dry storage test and demonstration program since 1985, sponsored by the Department of Energy (DOE) and the Electric Power Research Institute (EPRI). After loading the spent fuel in casks, the long-term behavior of the fuel and other components within the primary containment systems has not been verified by actual inspection. The spent fuel and the casks used in this demonstration project can provide an opportunity to evaluate the long-term behavior of the fuel and cask containment systems, in preparation for the upcoming license renewal period for the casks in current usage. DOE, EPRI and the US Nuclear Regulatory Commission (NRC) have agreed to participate in a cooperative research program to provide data on the long-term storage behavior of the spent fuel and storage casks for short term and long term storage applications.

DRY STORAGE CASKS

Prototypes of several commercial dry storage casks were procured or constructed on-site, and multiple viability tests and evaluations were conducted. In 1985 and 1986, three casks were performance tested by placing Westinghouse pressurized water reactor (PWR) spent fuel assemblies in the Gesellschaft fur Nuklear Service (GNS) CASTOR-V/21, Westinghouse Electric Company MC-10, and Transnuclear TN-24P metal casks.

The CASTOR V/21 contains 21 assemblies of intact Virginia Electric Power (VEPCO) Surry Reactor Westinghouse PWR fuel, and was fully loaded in 1985. The MC-10 has a 24-assembly capacity, and was loaded in 1985 with 18 assemblies of intact VEPCO Surry Reactor Westinghouse PWR fuel. The Transnuclear TN-24P has a capacity to hold fuel rods from 48 assemblies that are consolidated into 24 canisters, but now contains 7 of the original 24 (full load) canisters of 2:1 consolidated Westinghouse PWR fuel that were loaded in 1985. Some of the 24 assemblies were from the VEPCO Surry Reactor and some from the Florida Power Turkey Point Reactor. The spent fuel from Turkey Point was received from Engine Maintenance and Disassembly (EMAD) facility in southern Nevada, where it was in dry storage from 1979 until it was shipped to INEEL. Seventeen of these 24 canisters were transferred in 1989 to the Pacific Sierra Nuclear VSC-17 concrete cask.

CASK MONITORING PROGRAM

Since 1985, DOE and EPRI have sponsored a limited monitoring program at INEEL on the behavior of spent fuel and the fuel confinement systems. This monitoring has consisted of visual monitoring of the casks for obvious changes or degradation, monitoring the gas pressure inside the casks to evaluate the system integrity, and monitoring of radiation fields around the casks to evaluate shielding effectiveness. DOE has supported gas monitoring and analysis on a yearly basis since the casks were loaded, to catch early indications of fuel failure. INEEL determined initial baseline values for this monitoring after the casks were loaded, and this monitoring has continued until the present. The casks will have current values for the gas pressure and analysis as well as radiation field readings before the casks are opened for evaluation of the spent fuel and cask internals.

COOPERATIVE RESEARCH PROGRAM

Preparations to determine how the spent fuel and cask internals have behaved in the CASTOR V/21 and the Pacific Sierra VSC-17 cask are now being made. The program will start in fiscal year 1999, and is planned to continue through fiscal year 2001. The cooperative research program plan includes a video inspection of the spent fuel and cask internals, including known basket weld cracks. This inspection will be performed at the Test Area North facility at INEEL. Destructive examination and selected material tests will be performed at the Argonne National Laboratory facilities in Idaho Falls, Idaho and Argonne, Illinois. Tests are planned to evaluate degradation processes such cladding creep, oxidation, brittleness, hydride formation, corrosion, and others.

Cracking in Spent Fuel Dry Storage Casks

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Due to a limited storage capacity in spent fuel pools, some nuclear power plants are temporarily storing spent fuel on site in specially designed dry storage casks. One type of these dry storage casks has experienced weld cracking problems during closure welding. This particular cask design incorporates 2 lids which are both welded to the outer shell; the inner shield lid provides shielding from the spent fuel while the outer structural lid is used for structural integrity, additional shielding, and redundant sealing of the confinement system. NDE has found cracks up to 43 cm long in these lid to vessel welds. The cracks are attributed to undocumented weld repairs, moisture, hydrogen induced cracking (HIC), and improper fit-up during welding.

To prevent cracking in future casks several changes have been proposed: a 200°F preheat and postheat; use of low hydrogen electrodes; sequenced welding; and use of low sulfur, calcium-treated, vacuum-degassed steel in the construction of future casks. A delay time before inspection enhances the probability that any cracks which could develop after welding will be discovered. In addition to the current dye penetrant (PT), visual examination (VT), and helium leak check, a new ultrasonic examination (UT) will be required to provide reasonable assurance of the integrity of the weld. Flaws discovered during UT which do not meet a given screening criteria must be either repaired or reexamined using Linear Elastic Fracture Mechanics (LEFM) or Elastic Plastic Fracture Mechanics (EPFM). The analytical and experimental work used to determine this screening criteria are described in a companion paper.

This paper will describe the cracks found in these welds and explain the approach used to resolve this problem.

INTRODUCTION

As the amount of spent nuclear fuel approaches the finite capacity of each nuclear reactor site's spent fuel pool, some utilities have begun placing their fuel in dry storage casks. These specially designed containers provide a means of temporarily storing spent fuel on site. One particular dry storage cask design has experienced significant problems with weld cracking.

Essentially this design consists of a cylindrical shell and two redundant lids all of which are constructed from SA 516-Grade 70 steel. The inner shield-lid consists of a neutron shielding material sandwiched between two steel plates to provide radiologic protection from the spent fuel inside the container. The outer structural lid provides structural integrity to the cask, provides additional shielding from the spent fuel, and forms a redundant seal of the cask's confinement system. The cask is sealed by welding each lid to the cylindrical shell using shielded metal arc welding (SMAW), and flux core welding (FCAW) or gas metal arc welding (GMAW) processes. The structural-lid weld joint shown in Figure 1 forms a single bevel groove with a backing bar and requires multiple weld passes. Both the shield-lid to shell weld and the structural-lid to shell weld undergo dye penetrant (PT), visual examination (VT) and helium (He) leak tests. PT and VT examinations are conducted after the root and final weld passes while the He leak tests are only conducted after the final weld pass.

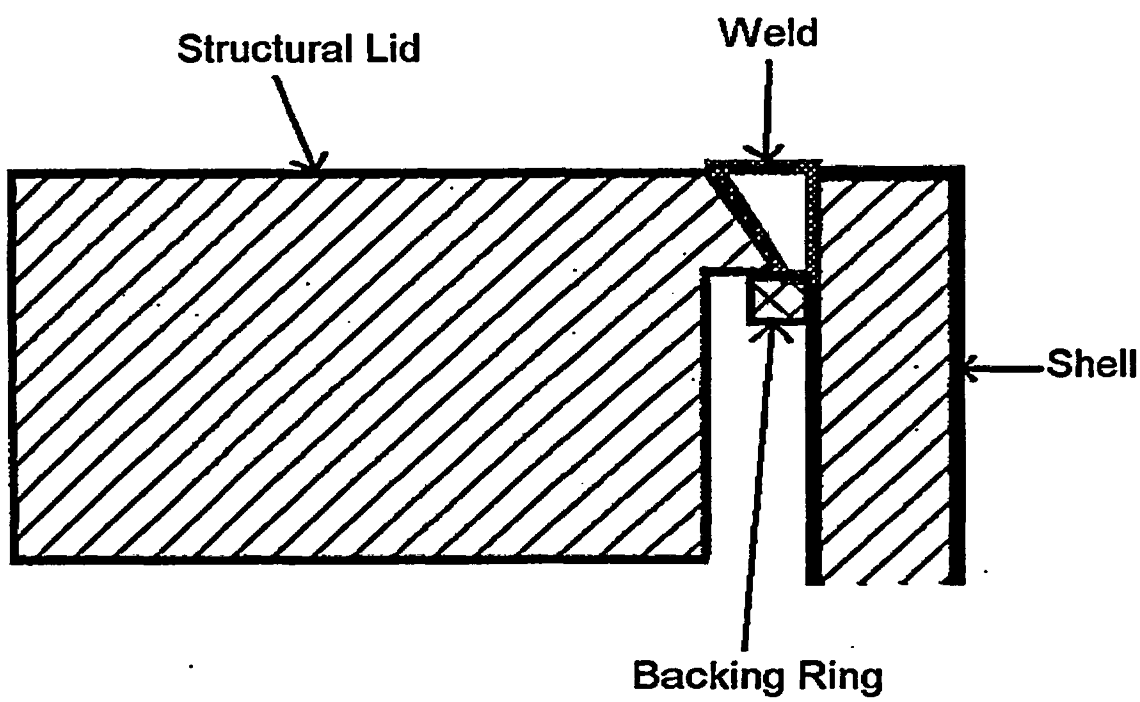

Figure 1 Structural lid to cylindrical shell weld geometry.

Since both the shield-lid and structural-lid weld function as part of the cask's confinement boundary, they are classified as important to safety. Failure of both of these welds would cause the He gas within the cask to escape. The lack of an inert He environment in the cask could allow the fuel cladding to degrade. This could eventually lead to problems in handling and recovering the spent fuel in the future.

BACKGROUND

Of the 19 casks currently loaded, 4 instances of weld cracking were identified during He leak testing or liquid penetrant examination of the lid to shell welds. All of the flaws were eventually repaired, the affected areas reexamined, and the casks placed into service.

After the first incidence in March of 1995, the utility conducted a root cause analysis of the problem and determined that the shell material defect resulted from a weld of unknown origin. In 1996 and 1997, three more instances of weld cracking occurred at separate utilities. This prompted an NRC inspection of the cask vendor and its fabricators in March 1997. This inspection concluded that the vendor's root cause analysis of the weld cracking and its corrective UT examination procedure to address this problem were both inadequate. The vendor was issued a nonconformance with 10 CFR 72.172 "Corrective Actions" for failure to take timely corrective action in identifying the root cause of the cracking. The NRC inspection team noted that the weld joint design is highly constrained and therefore susceptible to lamellar tearing. The inspection also identified the following three factors which contribute to the susceptibility of HIC in the welds: the shield lid to multi-assembly sealed basket (MSB) shell weld is performed when the water level in the cask is only 7.6 cm below the lid resulting in a very moist welding environment; the carbon equivalent of the materials used indicate susceptibility to HIC; and the welding procedures which allowed welding under ambient conditions without preheat.

On May 16, 1997, the NRC issued a Confirmatory Action Letter (CAL) to the cask vendor listing the following commitments to the NRC:

- to determine the root causes of weld cracking
- to assess the delayed cracking potential in the closure welds for the casks currently in use
- to determine appropriate corrective actions to prevent the recurrence of cracking in welds
- upon completion of these actions, to submit a written description of the evaluations listed above

In addition CAL's were issued to the utilities currently using the cask which documented their commitment to determine if their respective welding and inspection procedures provide reasonable assurance that undetected or delayed cracking will not occur in either lid to shell closure weld. Upon completion of this action and 14 days prior to loading another cask, each utility also agreed to submit a written description of any procedure or design modifications made as a result of their assessments.

The cask vendor formed a weld review team comprised of metallurgy, welding and non-destructive examination experts in order to perform the evaluations described in the CAL Similarly the NRC assembled its own team of staff experts to review their assessments.

ROOT CAUSES OF CRACKING EXPERIENCED IN CLOSURE WELDS

The first crack was discovered in March 1995 during a He leak test of the completed shield lid to shell weld. The crack was approximately 15 cm long and 0.3 cm deep. The flaw was approximately 0.3 cm above the weld fusion line and extended into the shell side of the weld. The flaw was initially attributed to subsurface lamination in the shell. However, further metallographic analysis showed the presence of an undocumented weld. The crack propagated along the prior austenite grain boundaries of this undocumented weld.

In May 1996 another utility found 3 cracks in the root pass of the structural lid to shell weld during the liquid penetrant examination. The crack lengths ranged from 0.6 to 2.5 cm and were located along the weld centerline. These cracks were caused by an uneven fit-up in the joint. The gap between the structural lid and backing ring varied around the circumference of the cask. In locations where this gap was widest, the welding pass was unable to adequately fill the gap leading to cracking during weld solidification. Weld porosity and cracking were also discovered in another weld joining the structural lid to the shield lid. These flaws were attributed to excessive moisture in the weld.

The third incidence of weld cracking was discovered in December 1996 during a He leak test of the shield lid to shell weld. The flaw was 10 cm long and located along the weld fusion line. Using only a visual examination of the crack, plant personnel concluded lamellar tearing was the cause. Unfortunately, no additional data on this crack is available for further analysis.

The last weld crack occurred in March of 1997. Liquid penetrant examination of the root pass of the shield lid to shell weld indicated a 1.90 cm long flaw in the weld fusion line. After grinding out the weld, it was discovered the crack was actually 43 cm long and extended through the thickness of the root pass into the shell. Additional testing was required to reveal the root cause of cracking in this case. The shell material of this particular cask had a slightly higher carbon equivalent than other casks leading to increased hardenability and susceptibility to hydrogen cracking. The weld wire used in this cask revealed a hydrogen content of 15.5 ml of H_2 at standard temperature and pressure per 100g of deposited weld metal (ml/H_2/STP/100g). Through thickness tensile tests were also performed on the shell material to determine the resistance of the material to lamellar tearing. The results showed a reduction in area of 35% which is well above the 20% required by the American Society of Mechanical Engineers (ASME) specification SA-770 for the Through-Thickness Tension Testing of Steel Plates for Special Applications. Based on this evidence it was concluded that the shield lid to shell weld was due to hydrogen induced cracking.

The various welding procedures and materials involved in the 4 weld cracking events were examined during the root cause analyses. It was noted that the cracks discovered in the last two incidents occurred at the same plant and had similar appearances suggesting an identical root cause for both cracks. The first two incidents, in which cracking was attributed to an undocumented weld and improper fit up, were found to have significant differences from the last two instances in which cracking was indeterminate or due to HIC. In the latter two cases fewer and smaller tack welds were used during welding causing less stability and a more uneven distribution of weld shrinkage strains. These two casks also had lower decay heat loads from the spent fuel resulting in a slightly lower material temperature prior to welding. This implies that an additional level of preheat could reduce the susceptibility of these welds to HIC.

The staff agreed with the conclusions of the cask vendor's weld review team in determining the root cause of each of the weld cracks.

POTENTIAL FOR DELAYED CRACKING IN EXISTING WELDS

The cask vendor's weld review team studied various delayed cracking mechanisms in the closure welds and determined hydrogen induced cracking the most likely. Plate laminations, lamellar tearing, or pre-existing defects could also cause weld defects, but these mechanisms would occur minutes after welding is complete. Therefore the NDE examinations should reveal them.

A source of diffusible hydrogen in the weld area, a material susceptible to hydrogen embrittlement, and stress are all required in order to form HIC. The weld wire samples from the 3 utilities showed hydrogen levels of approximately 15.5, 15.5, and 9.0 ml/H_2/STP/100g. These hydrogen levels are considered high enough to cause HIC. Using a carbon equivalent formula published by the International Institute of Welding (IIW), carbon equivalents for the various materials at the utilities ranged from 0.40 to 0.50. Materials with carbon equivalents greater than 0.40 are susceptible to hydrogen embrittlement. The joint configuration is highly constrained resulting in residual stresses in the weld which may be at or near the yield stress for SA516-70. Thus, all the conditions required for HIC are potentially present during the welding process.

The cask vendor's weld review team conducted a literature search compiling data on measured delay times between welding completion and the onset of HIC and concluded a maximum possible delay time of 3 hours for the closure welds. The team then reviewed inspection reports at each utility in which the casks are in use to determine the actual time between welding completion and inspection. The minimum inspection times for the shield lid to shell weld and structural lid to shell weld were 1.8 hr and 1 hr respectively. Based on the data from the literature and inspection reports, the weld review team concluded that even though the necessary conditions promoting HIC were present during welding of the previously loaded casks, the likelihood of HIC occurring after the inspection times is unlikely. The cask vendor's review team sited the fact that the weld crack attributed to HIC occurred 30 minutes after welding as additional proof of this conclusion.

The staff's weld team also concluded that the conditions needed to promote HIC were present during welding of the casks; however, the staff did not agree with the estimated 3 hour maximum delay time for HIC. The bases for the staff's disagreement are outlined below:

- Surface inspections of the closure welds does not necessarily indicate accurate delay times for cracking since HIC typically form subsurface cracks. In only the most severe circumstances would a HIC extend to the weld surface. Therefore, field inspections of the weld surface would not provide reasonable proof of either the lack or presence of any hydrogen-induced cracks.

- The data obtained by the review team on delay times were taken from experiments using single pass welds whereas the closure welds used in the casks are multi-pass welds. Additional data on delay times in multiple-pass weldments is needed in order to obtain a more reliable estimate.

- The measured delay times given in the literature are not defined consistently. In laboratory experiments delay times are defined as the time associated with a small, fixed level of crack propagation. This length of crack propagation is not the same among the various experiments. In actuality cracking begins and continues even after the experimentally determined delay time.

- Some of the delay time data found in the literature was excluded by the cask vendor's weld review team in determining the estimated delayed cracking time for the cask. The staff believes that the basis for excluding this data is inadequate.

The staff believes that all currently loaded casks could contain HIC which were not detected during examination.

PROPOSED CHANGES

The approach taken to resolve this issue involved 3 aspects: modification of the welding process to prevent the recurrence of weld cracking in the future; ultrasonic inspection (UT) of the closure welds to assure that the welding modifications were effective and to verify the structural integrity of the weld; use of a fracture mechanics based flaw screening criterion to evaluate any indications discovered by UT. The experimental and analytical work used to determine the flaw screening criterion is explained in detail in a companion paper.

The welding modifications instituted to prevent HIC in the future are described below:

- A 200°F preheat will be applied to the weld area and continued for 1 hour after completion of the weld to promote the diffusion of hydrogen out of the material before cooling to low temperatures. In addition the slower cooling rate reduces hardness levels and improves fracture and notch toughness levels in the material.

- The welding electrodes used will have hydrogen levels below 10 ml/H_2/STP/100g to limit the amount of available hydrogen in the weld.

- A two hour delay in the inspection time after welding will increase the probability that any HIC that develop will be discovered.

- Large tack welds or a balanced welding sequence will be used to prevent movement of the lid and to equally distribute the shrinkage stresses during cooling. This in addition to the post weld heat treatment will reduce the residual stress levels in the highly constrained weld.

- All future casks will be constructed from a calcium-treated, vacuum-degassed, low-sulfur steel plate for its superior through thickness mechanical properties and fracture toughness. The through-thickness strength is important because of the residual stresses imposed on the plate during welding while the improved fracture toughness aids in the design basis hypothetical drop accident of the cask.

Other corrective actions have been imposed to address the root cause failures associated with undocumented welds, improper fit-up, and moisture contamination in the weld.

In addition to the currently performed dye penetrant (PT), visual examination (VT), and He leak tests, a new UT will be required on the structural lid to shell weld to provide additional reasonable assurance of weld integrity. This UT procedure will be performed on both the casks to be loaded as well as currently loaded casks. A full-diameter, partial-height mockup of the canister's structural lid to shell closure weld was created and known flaws of various types, sizes, and orientations were inserted to test the viability of performing UT. A demonstration of the UT procedure showed that under field conditions the flaws critical to the structural integrity of the weld could be accurately and reliably detected.

A flaw size screening criterion was developed from measured material properties and stresses under the most limiting load conditions. Any UT indication which exceeds the screening criterion must either be repaired or reanalyzed using Linear Elastic Fracture Mechanics (LEFM) or Elastic Plastic Fracture Mechanics (EPFM) whichever is applicable. If the more detailed analysis shows that the flaw is still unacceptable, it must be repaired or removed.

CONCLUSION

The cask vendor has determined the root cause of each of the weld cracking incidences, assessed the potential for delayed cracking in loaded casks, and determined appropriate corrective actions to prevent the recurrence of weld cracks. The utilities have also agreed to perform UT on currently loaded casks to satisfy these concerns that delayed cracking may have occurred in these casks.

The Safety Analysis Report and the Certificate of Compliance for the cask will be amended to incorporate these corrective actions. In July 1998, the CAL was closed.

REFERENCES

1 Letter from Arthur J McSherry to William F. Kane (NRC),"Response to CAL 97-7-001", July 30, 1997.

2 Letter from Malcolm R. Knapp (NRC) to Edward Fuller, "Closure of Confirmatory Action Letter 97-7-001", July 22, 1998.

3 Letter from Susan Shankman (NRC) to Art McSherry, "NRC Inspection Report No. 72-1007/97-204 and Notice of Nonconformance", April 15, 1997.

4 Letter from Carl Paperiello (NRC) to John Massey, "Demand for Information", October 6, 1997.

5 Letter from Marissa Bailey (NRC) to William Kane (NRC), "Meeting Between Nuclear Regulatory Commission Staff and ", May 16, 1997.

6 Letter from Malcolm Knapp (NRC) to Art McSherry, "Confirmatory Action Letter", May 16, 1997.

7 Letter from Malcolm Knapp (NRC) to Thomas Palmisano, "Confirmatory Action Letter", May 16, 1997.

8 Letter from Malcolm Knapp (NRC) to C. Randy Hutchinson, "Confirmatory Action Letter", May 16, 1997.

9 Letter from Malcolm Knapp (NRC) to Richard Grigg, "Confirmatory Action Letter", May 16, 1997.

Determination of Flaw Size Screening Criteria for Spent Fuel Dry Storage Cask Welds

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Weld cracking problems have been experienced in a particular dry cask storage system [1]. Ultrasonic examination (UT) of the outer structural weld of this system's sealed canister will be utilized to locate and size any existing flaws in these vessels. Each flaw will then be scrutinized to determine if it's acceptable or requires additional analysis and possible repair.

Acceptable flaw sizes have been calculated using ASME Section XI, IWB-3600 and Appendix A in the following manner. A horizontal drop accident was considered as the limiting operating condition. The weld membrane stress due to horizontal drop accident conditions, as reported in the safety analysis report [2], is 43.3 ksi, and the residual membrane stress due to welding of the outer structural lid is conservatively assumed to be equivalent to the material's (SA516 steel) yield strength (38 ksi). SA516-Grade 70 Charpy V-notch impact energy and quasi-static fracture toughness specimens were tested for mock-up weldments which simulate the cask. These measured toughnesses were then adjusted to obtain a conservative lower-bound value and to account for dynamic loading rate effects. The potential membrane stresses and adjusted toughness values were input into an analytical model to determine the appropriate flaw size screening criteria at various service temperatures.

Trial UT inspections on a mockup of the canister closure weld with seeded defects confirmed the feasibility of performing the inspection under field conditions and showed that the UT method could reliably detect flaws smaller than the acceptable limit. This paper will present the details of fracture toughness testing of the cask closure weld materials, the methodology used to determine the acceptable flaw screening criteria, and an overview of the UT inspection procedure and results.

INTRODUCTION

The weld cracking problems associated with the closure welds in a particular spent nuclear fuel dry storage cask design have been previously documented [1]. The closure weld geometry shown in Figure 1 consists of a structural lid welded to a cylindrical shell both of which are constructed from SA516-Grade 70 steel.

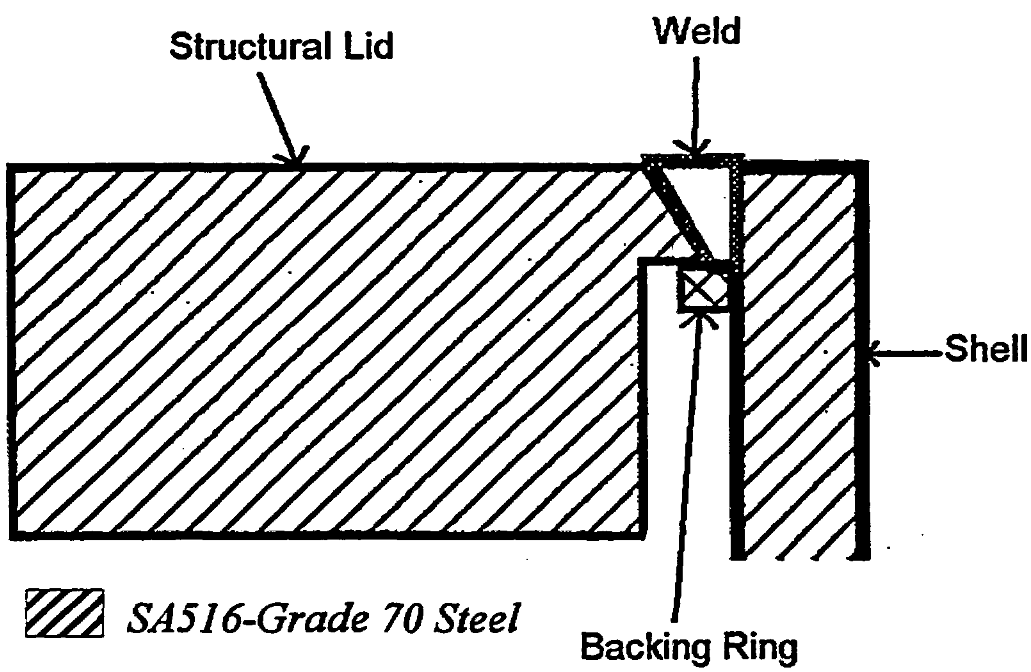

Figure 1 Structural lid to cylindrical shell weld geometry.

The welds are produced using either shielded metal arc welding (SMAW), flux core welding (FCAW) or gas metal arc welding (GMAW) processes. Post weld inspection has historically consisted of dye penetrant inspection (PT), visual examination(VT), and helium (He) pressure testing to search for leaks. Evidence of weld cracking has been found in all three plants which currently utilize this particular dry storage cask design.

It was determined that the root cause for some of these weld cracks is hydrogen induced cracking (HIC) [3]. Modified welding procedures [1] have been instituted to reduce the likelihood of

HIC in future construction. All closure welds in future construction will also undergo an additional ultrasonic examination (UT) using the Time of Flight Diffraction (TOFD) technique to verify the structural integrity of the weld.

The NRC staff was also concerned that the closure welds in all currently loaded casks could contain HIC which has not been detected during the PT, VT, or He leak inspections. Cask owners agreed to perform UT inspection on all currently loaded casks. Flaws found and sized during UT inspection in current and future casks will be evaluated using a fracture mechanics based acceptance criteria

The flaw size screening criteria was based on work conducted by the NRC staff, cask owner's group, and NRC contractors. The philosophy was to determine the largest allowable flaw size which could conservatively exist in the weld without leading to catastrophic failure (cleavage fracture). Flaws were assumed to exist at the most severe location and orientation. Also, the stresses induced from the emergency/faulted condition of a horizontal drop accident were assumed. The Certificate of Compliance for the cask prohibits the moving of loaded casks when ambient temperatures are below 0 °F (-17.8 °C). Therefore, the lowest possible temperature at which the horizontal drop accident could occur is 0 °F (-17.8 °C). These assumptions and the appropriate fracture toughness data were used to generate data for the flaw screening criteria for the SA516-70 steel, weld metal, and heat affected zone (HAZ).

FRACTURE TOUGHNESS TESTING

Material

Two 4'x8'x1" (1.22m x 2.44m x 2.54cm) normalized SA516-Grade 70 plates (Heat #3484) were acquired from the Azovstal Iron and Steel Works and cut into 6"x24" (15.24cm x 69.96cm) blanks. The chemical composition of these base plates were independently tested by Azovstal Iron and Steel Works, the Energy & Process Corporation (EPC), and the Carderock Division of the Naval Surface Warfare

Center (NSWC). Table 1 shows the results of these chemical analyses as well as the maximum specification requirements (unless a range is given) for SA516-Grade 70 steel. The results from the three laboratories show good agreement and are all within the SA516-70 specifications.

Table 1. SA516-70 chemical composition specifications and measured chemistries of SA516-70 baseplate used in analysis. Taken from Reference 4.

Element	SA516-70 Specification	Azovstal	EPC	NSWC
C	0.28	0.18	0.17	0.18
Mn	0.79 – 1.30	1.10	1.14	1.10
Si	0.13 – 0.45	0.29	0.29	0.31
S	0.035	0.008	0.010	0.01
P	0.035	0.012	0.014	0.007
Cr	0.34	0.06	0.05	0.06
Ni	0.43	0.06	0.05	0.06
Cu	0.43	0.03	0.04	0.04
Ca	—	NR	NR	0.007
Al	—	0.038	NR	0.048
Nb	0.03	0.010	0.001	0.003
N	—	0.011	NR	0.009
V	0.04	0.005	0.0005	<.002
Mo	0.13	0.010	0.001	0.002
Ti	—	0.003	NR	0.004
Cr+Ni+Cu+Mo	1.00	0.16	0.14	0.16
Cr+Mo	0.32	0.07	0.05	0.06

Table 2 shows tensile property specifications for SA516-70 steel: the minimum yield strength, range of ultimate tensile strengths, and minimum uniform percent elongation. The results of tensile tests performed by Azovstal and EPC are also shown in Table 2 [5]. The yield strengths measured by the 2 laboratories (55 ksi and 53 ksi) are well above the minimum required yield strength (38 ksi). Azovstal and EPC had identical measured ultimate tensile strengths for the baseplate which is toward the lower end of the specified range. Based on the results shown in Tables 1 and 2 the sample baseplate acquired from Azovstal meets all the required specifications for SA516-70 steel.

Table 2. SA516-70 tensile property specifications and measured tensile properties of SA516-70 baseplate used in analysis.

Property	SA516-70 Specification	Azovstal	EPC
Yield Strength	38 ksi (262 MPa)	55 ksi (379 MPa)	53 ksi (365 MPa)
Ultimate Tensile Strength	70-90 ksi (483-621 MPa)	76 ksi (524 MPa)	76 ksi (524 MPa)
% Elongation with 2 in. gauge length	21%	27%	36%

Mock-up weld coupons were constructed from the SA516-70 by each of the three affected plants to simulate the structural lid weld in Figure 1. A sketch of these weld coupon geometry is shown in Figure 2. Both manual and automated welding techniques are allowed during fabrication. All of the plants use SMAW for manual welding. Plants 1 and 2 use FCAW while Plant 3 uses GMAW for automated welds. Each plant constructed a manual and automated weld coupon mock-up using their approved welding fabrication procedures and filler materials

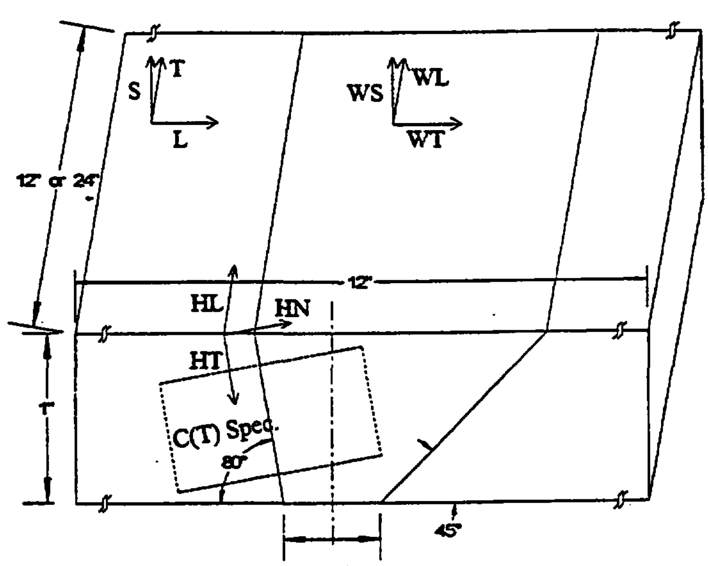

Figure 2. Weld coupon geometry and coordinate system. Taken from Reference 4.

Therefore, a total of 6 weld coupon types were manufactured. Specimens were extracted from each of these coupon types and tested by the cask owner's group and an NRC contractor.

Owner's Group Testing

Testing for the cask owner's group was performed at the Westmoreland Research and Testing Laboratory. Charpy V-Notch (CVN) impact tests were conducted on specimens taken from the weld metal (WM) and heat affected zones (HAZ) for each coupon type. Base metal (BM) specimens were tested in the transverse (BM_T) and longitudinal (BM_L) directions. The purpose of these tests was to identify the locations with the lowest Charpy energies for additional fracture toughness testing. All tests were conducted at 0 °F (-17.8 °C). The results are shown in Table 3 below:

Table 3. CVN test results of SA516-70 BM, WM, and HAZ material from the Westmoreland Research and Testing Laboratory. Taken from Reference 6.

Plant	CVN Energy in (ft-lb)					
	Manual Welding		Base Metal		Automated Welding	
	HAZ	WM	BM_T	BM_L	HAZ	WM
Plant 1	141	78	47	96	65	73
Plant 2	146	122	47	96	154	115
Plant 3†	147/141	110/118	47	96	126/156	52/42

The lowest CVN energies occurred in the weld metal for Plant 3's automated welding process. Also, the weld metal and base metal results tended to be lower than the HAZ results. Based on these results, eight locations were identified for further quasi-static J-integral (J_{IC}) fracture toughness testing: the weld metal for all the coupons, the base metal, and the HAZ for Plant 1's automated weld. The J_{IC} tests were conducted according to ASTM E1737-96 procedures at 0 °F. Three specimens from each of the 8 locations were tested and the results of are shown in Table 4.

†Plant 3 sent 2 samples of each specimen type to be tested resulting in 2 sets of data.

Table 4. J integral test results on SA516-70 BM, WM and HAZ from Westmoreland Research and Testing Lab. Taken from Reference 6.

Plant Material Welding Process	Charpy Energy (ft-lb)	Measured J_{IC} (lb/in)	Measured J_Q (lb/in)	Measured J_U (lb/in)	Calculated K_{Jx} (ksi $\sqrt{in}$)
Plant 3 Weld GMAW	42		880		170
			801		162
		776			160
Plant 3 Weld SMAW	110		2327		277
			2397		281
			1058		187
Plant 2 Weld SMAW	122		2234		271
			2453		284
			1912		251
Plant 2 Weld FCAW	115			1410	216
			1009		182
				621	143
Plant 1 HAZ FCAW	65		2614		294
			2444		284
			2064		261
Plant 1 Weld FCAW	73			616	143
				822	165
			1310		208
Plant 1 Weld SMAW	78			1685	236
			2673		297
			2375		280
All Plants Base Metal Longitudinal Direction	96		2655		296
			2737		300
			2462		285

The first column of Table 4 describes the location in which the fracture toughness specimens were taken. The second column shows the corresponding CVN energy for that specimen location as reported in Table 3. The reported J integral fracture toughness values given as either the initiation toughness (J_{IC}), the qualified initiation toughness (J_Q) or J_u are shown in the next three columns. The K_{Jx} fracture toughness values shown in the last column were calculated from the reported J integral value using the relation shown in equation (1):

$$K_{Jx} = \sqrt{J_x * \frac{E}{(1-\nu^2)}} \qquad (1)$$

where E = Young's Modulus

ν = Poisson's ratio

NRC Testing

Confirmatory fracture toughness testing of these materials was conducted for the NRC by the Carderock Division of the Naval Surface Warfare Center (NSWC). The following HAZ and weld metal fracture specimens were manufactured from Plant 1's weld coupon:

- seven, 0.5" (1.27cm) thick subsized compact tension, C(T), HAZ specimens from the FCAW coupon
- seven, 0.65" (1.65cm) thick subsized C(T) HAZ specimens from the FCAW coupon
- seven, 0.65" (1.65cm) thick subsized C(T) HAZ specimens from the SMAW coupon
- seven, full plate thickness single edge notched bend, SE(B), weld metal specimens from the SMAW process

Additionally, eight 1" (2.54cm) C(T) base metal specimens were machined in the TL orientation. All the weld specimens were oriented such that the thickness dimension of each fracture toughness specimen was parallel to the through-thickness direction of the weld coupon. The HAZ specimens' notches were parallel to the fusion line located within the coarse grained HAZ (Figure 2). The HAZ specimens were precracked extensively (approximately 0.1") to allow the fatigue crack to follow the weakest microstructure. The weld metal specimens' notches were located at the weld centerline. All fracture toughness tests were conducted as to ASTM E1737 at 0 °F under quasistatic loading rates. Cleavage toughness after small or no ductile tearing (J_u or J_{Ic}), the final toughness measured in the test (J_f), or the qualified initiation toughness (J_q) values were measured as appropriate. No valid J_{IC} were measured in any of the HAZ or weld metal tests. All but one of the base metal tests did produce valid J_{IC} results. Table 5 shows the appropriately measured J integral toughness value for each test specimen and the corresponding K_{Jx} value calculated using equation (1).

Table 5. J integral test results on SA516-70 BM, WM and HAZ from NSWC. Taken from Reference 4.

Specimen		J_{Ic} (lb/in)	J_q (lb/in)	J_u (lb/in)	J_f (lb/in)	K_{Jx} (ksi $\sqrt{in}$)
Plant 1 HAZ FCAW	subsize 0.5" (1.27cm) thick C(T) specimens		1098			187
			1001			179
			1487			218
					3607	339
					1860	243
					3937	354
			1756			237
Plant 1 HAZ FCAW	subsize 0.65" (1.65cm) thick C(T) specimens		1517			220
					3600	339
			1697			233
					2447	279
			1560			223
			2000			252

					3450	332
Plant 1 HAZ SMAW	0.65" (1.65cm) thick C(T) specimens				4257	368
			3226			321
			3497			334
					4000	357
			3545			336
			3492			334
			3069			313
Plant 1 Weld metal SMAW	1" (2.54cm) thick SE(B) specimens				2825	300
				2066	2066	257
					2751	296
					2700	293
					2596	288
					2436	279
					3016	310
Plant 3 Base Metal TL Orientation	1" (2.54cm) thick C(T) specimens	1614				227
		1618				227
		1725				234
		1656				230
		1724				234
		1701				233
			1746			236
		1754				236

Eight Charpy specimens were also machined from the baseplate material in the TL orientation: 4 were tested at 0 °F (-17.8 °C) and 4 were tested at -50 °F (-45.6 °C). The results of these Charpy tests are given in Table 6:

Table 6. CVN test results of SA516-70 base metal from the NSWC. Taken from Reference 4..

Specimen Number	Test Temp (F)	CVN Energy (ft lb)	Average CVN Energy (ft lb)	% Shear Fracture (%)	Lateral Expansion (mils)
4	0	65	64	70	58
5		60		60	58
6		67		70	60
7		66		70	59
8	-50	37	39	20	37
9		37		10	36
10		37		10	37
11		44		10	41

The CVN energies, percent shear fracture and lateral expansion is reported for each Charpy specimen along with the average CVN energy for each batch.

NRC STAFF ANALYSIS

The flaw acceptance criteria calculation proposed by the cask owner's group correlated fracture toughness from Charpy results using the Barsom Rolfe equation shown below in equation (2).

$$K_{ID}=\sqrt{5*C_{VNE}*E} \qquad (2)$$

where K_{ID} = dynamic fracture toughness (ksi $\sqrt{in}$)

C_{VNE} = Charpy V-Notch energy (ft-lb)

E = Young's Modulus (psi)

Figure 3 shows the static fracture toughness results (K_{JX}) versus Charpy impact energy for the data provided by the owner's group testing. Comparing this data with the predicted results of the Barsom Rolfe equation (shown as the solid line in Figure 3) it can be seen that there is no clear correlation between CVN and K_{IC}. Therefore, the NRC staff concluded that use of the Barsom-Rolfe correlation to infer dynamic properties for this material was not justified.

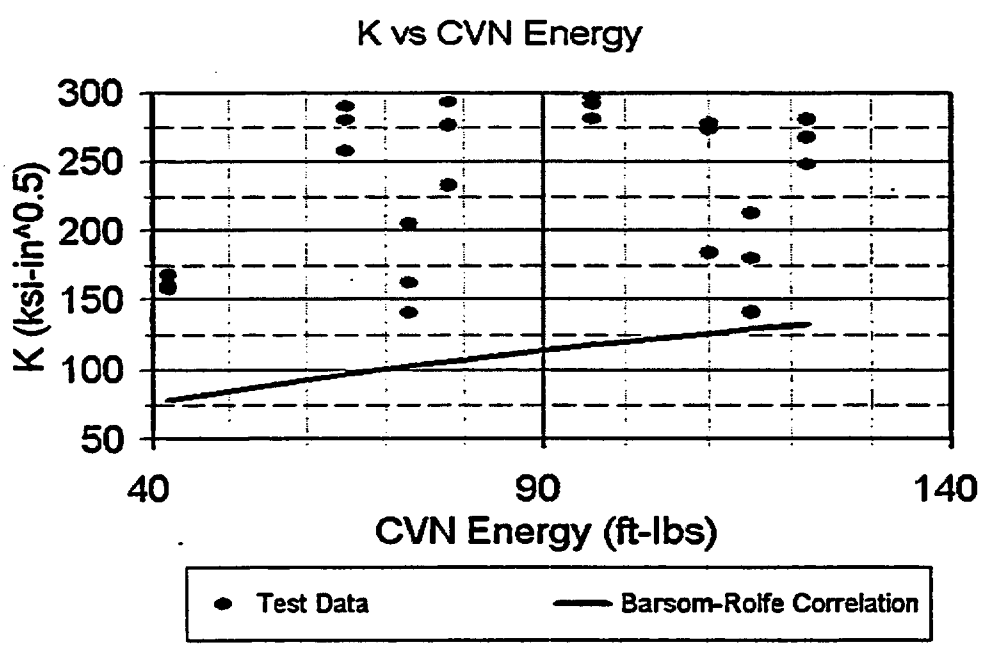

Figure 3 Static fracture toughness versus Charpy V-Notch impact energy results from Westmoreland.

The staff performed the following evaluation of the material fracture toughness data generated by both the NSWC and the cask owner's group to determine an appropriate lower bound dynamic fracture toughness value for further analysis.

All the fracture toughness data from Westmoreland and the NSWC were determined under quasi static loading rates. These values were first adjusted to account for dynamic loading effects by using the ratio between the ASME Section XI K_{IC} and K_{Ia} curves at a specific temperature. Appendix A of ASME Section XI defines a quasi-static fracture toughness curve (K_{IC}) and a dynamic fracture toughness curve (K_{Ia}) indexed to ($T-RT_{ndt}$) as follows:

$$K_{IC} = 33.2 + 20.734 * \exp [\, 0.02 * (T-RT_{ndt}) \,] \quad (\text{ksi}\sqrt{\text{in}}) \qquad (3)$$

$$K_{Ia} = 26.8 + 12.445 * \exp [\, 0.0145 * (T-RT_{ndt}) \,] \quad (\text{ksi}\sqrt{\text{in}}) \qquad (4)$$

where T = temperature (F)

RT_{ndt} = reference temperature used to index fracture toughness behavior defined in ASME, Section III, NB-2331 (F)

It was assumed that the RT_{ndt} of the SA516-70 baseplate, weld material and HAZ was equivalent and that RT_{ndt} (as provided by [7]) is -50 °F. Therefore, at the test temperature ($T-RT_{ndt}$)=+50 °F, and the design values provided by equations (3) and (4), result in K_{Ia} and K_{IC} values of 52.5 and 89.6ksi $\sqrt{\text{in}}$ respectively. All the quasi static fracture toughness values (K_{Jx}) obtained by the NSWC and the cask owners' group were adjusted by the ratio of K_{Ia} K_{IC} to (0.59) to correct for dynamic loading effects at a service temperature of 0 °F (-17.8 °C).

The *dynamically adjusted* fracture toughness data for the baseplate, weld, and HAZ materials tested by both Westmoreland and NSWC had a mean value of 144 ksi $\sqrt{\text{in}}$ and a standard deviation of 34.7 ksi $\sqrt{\text{in}}$. A *lower bound* fracture toughness value for this data set was calculated by taking the mean of the adjusted data and reducing it by two times the standard deviation (144 ksi $\sqrt{\text{in}}$ - 2 * 34.7 ksi $\sqrt{\text{in}}$ = 75 ksi $\sqrt{\text{in}}$)

ANALYTICAL MODEL

As stated earlier, the most severe stresses in the closure weld occur during the horizontal drop emergency/faulted accident condition. The closure weld stresses associated with this accident consist of a membrane stress (P_m) of 7.2 ksi and a local plus bending stress ($P_L + P_B$) of 43.3 ksi [2]. For conservatism the critical flaw size analysis used a membrane stress of 43.3 ksi and a weld residual stress ($P_{residual}$) equivalent to the minimum specification yield strength of SA516-70 steel (38 ksi).

Under emergency/faulted conditions, ASME Section XI, Appendix H requires a safety factor of $\sqrt{2}$ on membrane stresses and a safety factor of 1 on weld residual stresses. The applied stress intensity factor, $K_{Ia\ applied}$, is therefore defined as:

$$K_{Ia\ applied} = \sqrt{2} * K_{I\ membrane} + K_{I\ residual} \leq K_{ID} \qquad (5)$$

where $K_{I\ membrane}$= stress intensity factor due to membrane stress

$K_{I\ residual}$= stress intensity factor due to weld residual stress

K_{ID} = adjusted lower bound dynamic fracture toughness for the material (75 ksi $\sqrt{in}$)

Dividing both sides by $\sqrt{2}$ gives:

$$K_{Ia\ applied} = K_{I\ membrane} + K_{I\ residual}/\sqrt{2} \leq K_{ID}/\sqrt{2} \qquad (6)$$

where $K_{ID}/\sqrt{2}$ = 75 ksi $\sqrt{in}/\sqrt{2}$ = 53 ksi $\sqrt{in}$

According to equation (6), catastrophic cleavage fracture is avoided if the applied stress intensity ($K_{Ia\ applied}$) is less than the lower bound dynamic fracture toughness of the material. The explicit margin in equation (6) is the $\sqrt{2}$ safety factor on the operating stress.

The NRC staff also evaluated the effect of increasing the minimum allowable service temperature (0°F or -17.8 °C) on the adjusted lower bound fracture toughness property of the material. The adjusted lower bound fracture toughness at 0°F (75 ksi $\sqrt{in}$) was increased in proportion to the increase of the by the K_{Ia} fracture toughness curve (equation 4) at higher $T-RT_{ndt}$ values.

Mathematically this can be expressed in equation 7 below:

$$\frac{K_{Ia}^{ASME}(T-RT_{ndt})}{K_{Ia}^{ASME}(0-RT_{ndt})}=\frac{K_{ALBD}(T)}{K_{ALBD}(0)} \tag{7}$$

where T = service temperature (°F)

RT_{ndt} = -50 °F

K_{Ia}^{ASME} (T - RT_{ndt}) = ASME K_{Ia} evaluated at a service temperature, T

K_{Ia}^{ASME} (0 - RT_{ndt}) = ASME K_{Ia} evaluated at a service temperature of 0 °F

K_{ALBD} (T) = adjusted lower bound dynamic fracture toughness for the material at service temperature, T

K_{ALBD} (0) = adjusted lower bound dynamic fracture toughness for the material at service temperature of 0 °F = 75 ksi $\sqrt{in}$

Solving equation 7 for K_{ALBD} (T) gives:

$$K_{ALBD}(T)=\left[\frac{K_{Ia}^{ASME}(T-RT_{ndt})}{K_{Ia}^{ASME}(0-RT_{ndt})}\right]\times\left[K_{ALBD}(0)\right] \tag{8}$$

Table 7 summarizes the adjusted lower bound dynamic fracture toughness at service temperatures of 10, 20, 30 and 40 °F based on Equation 8. temperature calculated using equation 8. The last column in Table 7 provides the adjusted lower bound dynamic fracture toughness at each service temperature divided by the explicit safety factor (SF) of $\sqrt{2}$. This is the value used in all future analysis to determine the flaw screening criteria. It is interesting to note that the $K_{ALBD}(T)$ / SF is approximately equivalent to the ASME K_{Ia} value at each of the reported service temperatures.

Table 7. Adjusted lower bound fracture toughnesses at various service temperatures

Service Temp. (F)	$T-RT_{ndt}$ (F)	$K_{Ia}^{ASME}(T-RT_{ndt})$ (ksi $\sqrt{in}$)	$K_{Ia}^{ASME}(T-RT_{ndt})$ / $K_{Ia}^{ASME}(0-RT_{ndt})$	$K_{ALBD}(T)$ (ksi $\sqrt{in}$)	$K_{ALBD}(T)$ / SF (ksi $\sqrt{in}$)
0	50	53	1.00	75	53
10	60	57	1.08	81	57
20	70	61	1.16	87	62
30	80	67	1.27	95	67
40	90	73	1.38	104	73

As illustrated in Table 7, the shape of the fracture toughness curve in the transition region results in a significant improvement in the fracture behavior of this material over the temperature range shown.

The NRC staff conducted a fracture mechanics analysis of the integrity of the closure weld using the conservative membrane stresses and weld residual stresses from the horizontal drop scenario described earlier. It was assumed that the crack plane of the assumed flaw was located perpendicular to this applied stress. Crack aspect ratio is defined as the crack depth (a) divided by surface crack length (c). The postulated flaws used in the analysis were surface cracks with aspect ratios (crack depth divided by surface length) of 0.5, 0.2, 0.1 and 0. Note that an aspect ratio of 0.5 indicates a "semi-circular" crack while an aspect ratio of 0 describes a crack with an "infinite length." Embedded elliptical cracks could also occur in the closure welds, but were not included in the analytical model because the stress intensity factors for embedded cracks are lower than surface breaking cracks for the same aspect ratios.

For cracks with aspect ratios of 0.2, 0.1 and 0, the applied stress intensity factors were calculated using version 2.0 of the pc-Crack[††] computer code. Stress intensity factors for cracks with aspect ratios of 0.5 were calculated using influence functions in Tables A-3320-1 and A-3329-2 of the 1995 version of the ASME Section XI, Appendix A code. The pc-Crack computer code calculates the stress intensity factor only at the crack depth which coincides with the location of the maximum stress intensity factor for aspect ratios of 0.2, 0.1 and 0. However, for cracks with aspect ratios of 0.5, the maximum stress intensity factor occurs at the free surface tip of the crack and required influence function table solutions.

The maximum stress intensity factors as a function of crack depth for various aspect ratios are shown in Figure 4. Figure 5 illustrates the NRC flaw screening criteria developed from combining material toughness values at various temperatures in Table 7 with the Figure 4 results. The 'Acceptable Flaw Size' region in Figure 5 defines the flaw size regime which will not undergo catastrophic cleavage fracture during a horizontal drop accident at the specified service temperature.

[††]"Fracture Mechanics Software for Personal Computers," Structural Integrity Associates, Inc., San Jose, California, 1989.

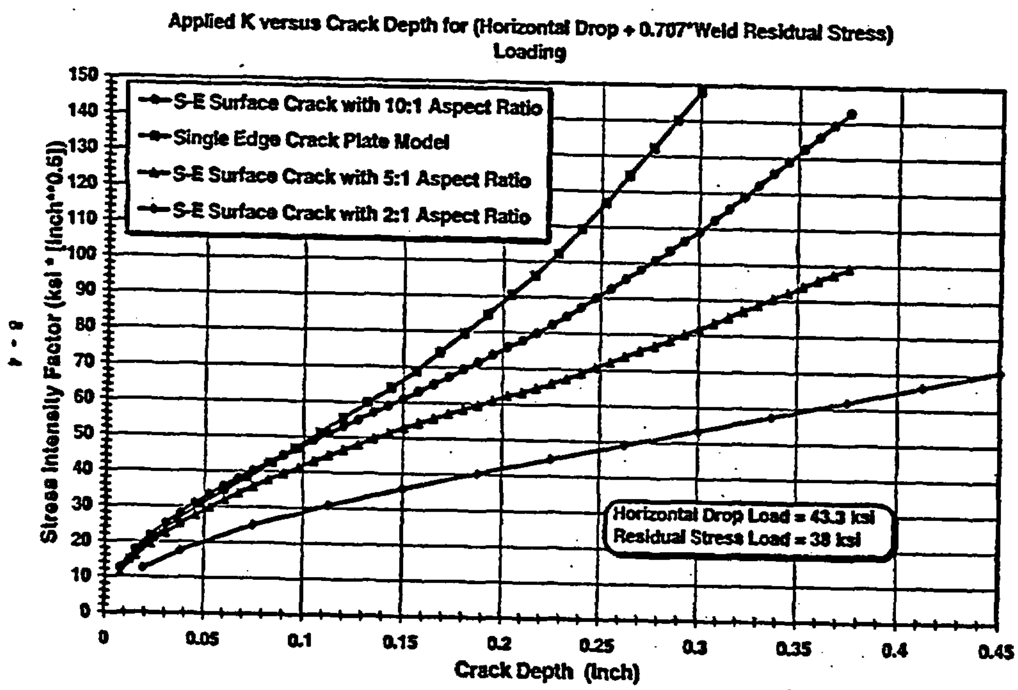

Figure 4 Maximum stress intensity factor as a function of crack depth for cracks of various aspect ratios

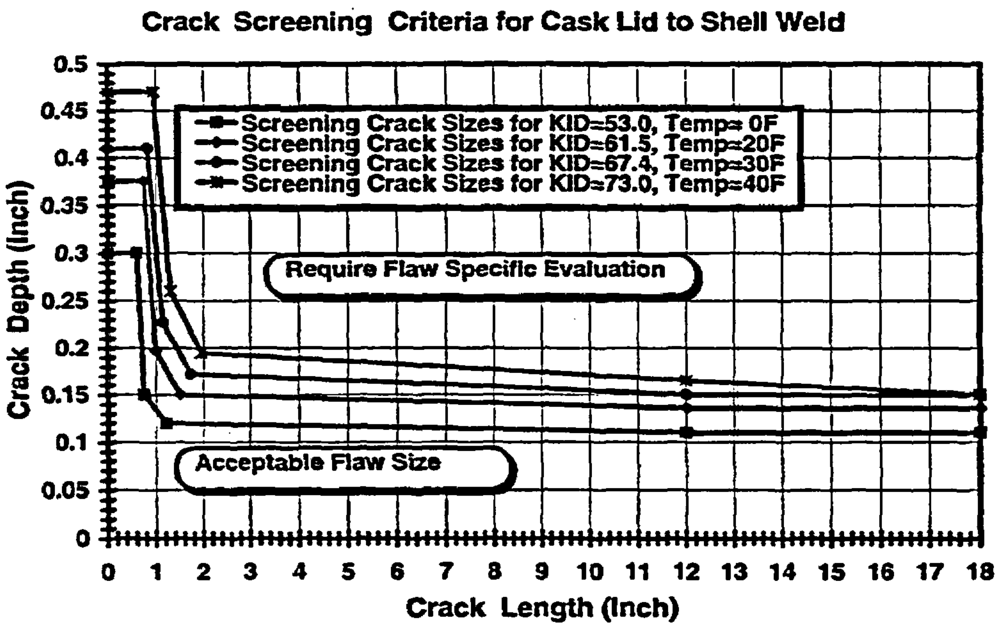

Figure 5 Flaw screening criteria determined from the fracture mechanics analysis at various service temperatures shown as crack depth versus crack length.

INSPECTION CAPABILITIES

A full diameter partial height mockup of the cask and closure weld was created in order to determine the feasibility of performing TOFD examination on the casks; to qualify TOFD as an inspection procedure; and to verify the capability of TOFD for detecting the necessary flaw sizes. Thirty-three known flaws of various sizes, orientations, locations and types were inserted into the closure weld. The flaws were approximately 0.5" (0.13cm) long with depths ranging from 0.05" (0.13 cm) to 0.25" (0.64cm). In addition to cracks, welding fabrication flaws such as lack of fusion and slag inclusions were also inserted into the mockup.

The UT examination was conducted at a licensee site in order to reproduce the field conditions under which the actual UT inspections will be performed. Four TOFD inspections were performed to determine the repeatability and accuracy of the procedure. All 33 flaws were detected in every inspection. In Figure 6 the average flaw depth reported from UT is plotted versus the actual flaw depths. A linear fit to this data set is shown by the solid line (Measured vs Actual) while the short dashed line (Ideal) defines a 1-to1 correlation between measured and actual values. The long dashed lines provide the upper and lower 95% confidence limit for the data. The mean linear fit of this data illustrates a general slight conservative bias in the TOFD technique. Also, outliers which fall outside the 95% confidence bounds are conservative. Based on these results the NRC staff concluded that the TOFD examination procedure could reliably and consistently detect flaw sizes important to the integrity of the closure weld.

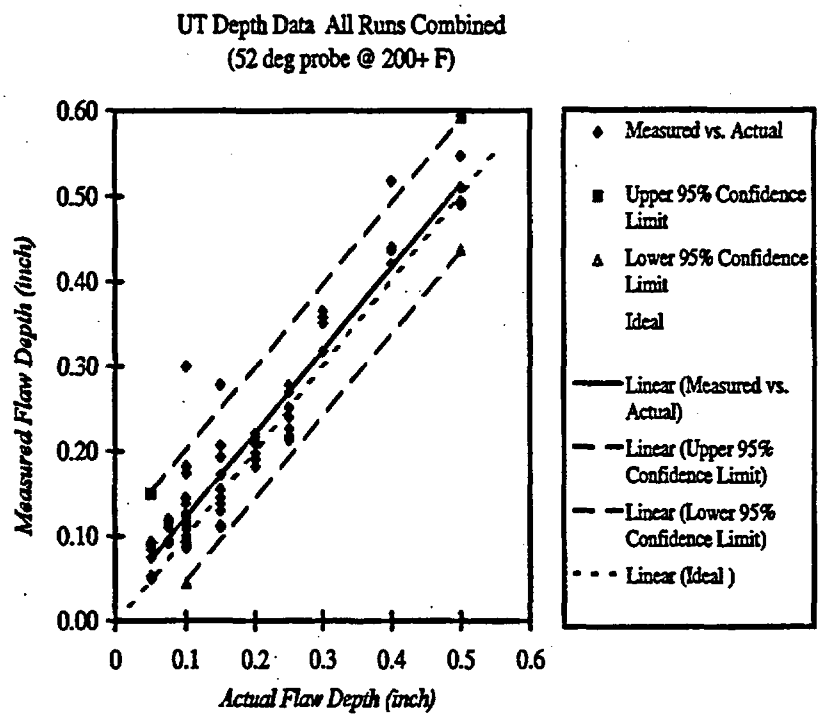

Figure 6 Comparison of actual flaw depth vs measured flaw depth in UT inspection of weld mockup.

REGULATORY GUIDELINES

The Certificate of Compliance for this particular dry storage cask was changed to specify the minimum allowable move temperature as 30 °F (-1.1 °C) instead of 0 °F (-17.8 °C) to take advantage of the improved fracture toughness properties at higher temperatures. The actual flaw size screening criterion during inspections is given below:

- for flaws with lengths ≤ 0.7" (1.78cm), the maximum allowable flaw depth is 0.37" (0.94cm)
- for flaws with lengths > 0.7" (1.78cm), the maximum allowable flaw depth is 0.16" (0.41cm)

Figure 7 shows the critical flaw size from the analysis at 30 °F (taken from Figure 5) and the required flaw screening criteria. Figure 6 shows the majority of data points which fall below the ideal curve are flaws with depths less than 0.3 inches (0.76 cm). Figure 7 shows that for crack lengths less than 2 inches (5.08 cm) the required flaw screening criterion curve (solid line) is well below the critical flaw sizes determined from the analysis (long dashed line) to account for uncertainty in the TOFD sizing accuracy. For crack lengths between 2 (5.08 cm) and 8 inches (20.32 cm) the 2 screening criteria are virtually identical. Only for very long cracks (greater than 8 inches) is the required screening criteria above the analytically determined screening criteria.

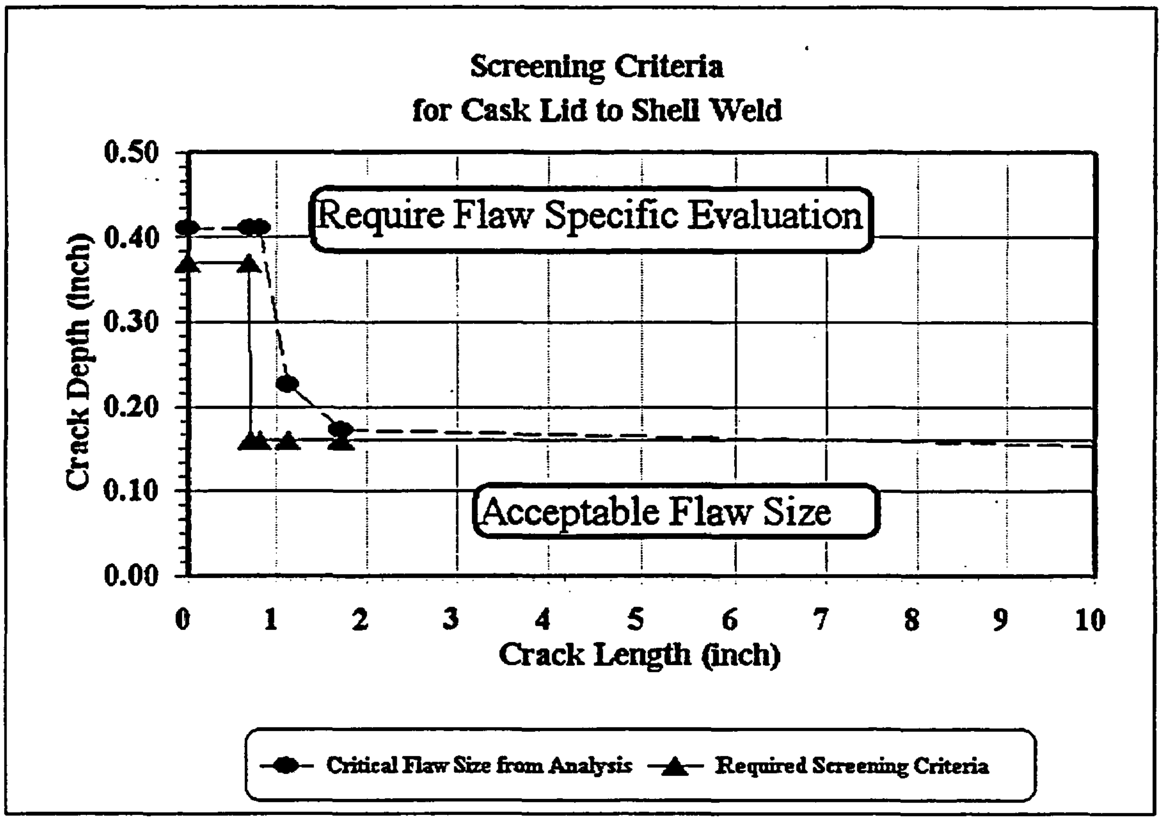

Figure 7 Critical flaw size from fracture mechanics analysis and required screening criteria shown as a function of crack depth versus crack length

If a flaw is found during inspection which falls outside the "Acceptable Flaw Size" region, a flaw specific analysis would be performed using either Linear Elastic Fracture Mechanics (LEFM) or Elastic Plastic Fracture Mechanics (EPFM) as appropriate. If additional analysis shows the flaw to be unacceptable, it must be repaired according to Article NC-4000 Fabrication and Installation, ASME Boiler and Pressure Vessel Code, Section III.

CONCLUSIONS

The flaw screening criterion and change in minimum allowable service temperature were the result of a collaborative effort among the licensees, the NRC staff, and NRC contractors. The material property data generated by the NSWC and Westmoreland was analyzed to determine a dynamic lower-bound material toughness of the material. This fracture toughness data was then combined with stress analysis results to determine critical flaw sizes. A UT examination of a cask closure weld mockup showed that the TOFD method could accurately size and consistently detect the flaw sizes important to the structural integrity of the weld. Based on this information, the following inspection screening criteria has been developed:

- for flaws with lengths ≤ 0.7" (1.78cm), the maximum allowable flaw depth is 0.37" (0.94cm)
- for flaws with lengths > 0.7" (1.78cm), the maximum allowable flaw depth is 0.16" (0.41cm)

Any flaw indications found during UT which do not meet the screening criterion will have to be analyzed further using LEFM or EPFM techniques.

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NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

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-0166
Vol. 1

2. TITLE AND SUBTITLE

Proceedings of the Twenty-Sixth Water Reactor Safety Information Meeting

Plenary Sessions, Pressure Vessel Research, Severe Accident Research, Fission Product Behavior, Nuclear Materials Issues and Health Effects Research, Materials Integrity Issues

3. DATE REPORT PUBLISHED

MONTH	YEAR
June	1999

4. FIN OR GRANT NUMBER

A3988

5. AUTHOR(S)

Compiled by Susan Monteleone, Brookhaven National Laboratory

6. TYPE OF REPORT

Conference Proceedings

7. PERIOD COVERED *(Inclusive Dates)*

October 26-28, 1998

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.)*

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

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.)*

Same as 8. above.

10. SUPPLEMENTARY NOTES

S. Nesmith, NRC Project Manager   Proceedings prepared by Brookhaven National Laboratory

11. ABSTRACT *(200 words or less)*

This three-volume report contains papers presented at the Twenty-Sixth Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel, Bethesda, Maryland, October 26-28, 1998. The papers are printed in the order of their presentation in each session and describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included papers presented by researchers from France, Germany, Italy, Japan, Norway, Russia, Sweden and Switzerland. The titles of the papers and the names of the authors have been updated and may differ from those that appeared in the final program of the meeting.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

BWR Type Reactors - Reactor Safety, Nuclear Power Plants - Reactor Safety, PWR Type Reactors - Reactor Safety, Reactor Safety - Meetings

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page) unclassified

(This Report) unclassified

15. NUMBER OF PAGES

16. PRICE

NRC FORM 335 (2-89)

Printed on recycled paper

Federal Recycling Program

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS
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