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Nondestructive Examination (NDE) Reliability for Inservice Inspection of Light Water Reactors

Annual Report
October 1989 — September 1990

Prepared by S. R. Doctor, M. S. Good, P. G. Heasler, R. L. Hockey,
F. A. Simonen, J. C. Spanner, T. T. Taylor, T. V. Vo

Pacific Northwest Laboratory
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Prepared for
U.S. Nuclear Regulatory Commission

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Abstract

The Evaluation and Improvement of NDE Reliability for Inservice Inspection of Light Water Reactors (NDE Reliability) Program at the Pacific Northwest Laboratory was established by the Nuclear Regulatory Commission to determine the reliability of current inservice inspection (ISI) techniques and to develop recommendations that will ensure a suitably high inspection reliability. The objectives of this program include determining the reliability of ISI performed on the primary systems of commercial light-water reactors (LWRs); using probabilistic fracture mechanics analysis to determine the impact of NDE unreliability on system safety; and evaluating reliability improvements that can be achieved with improved and advanced technology. A final objective is to formulate recommended revisions to the Regulatory and ASME Code requirements, based on material properties, service conditions, and NDE uncertainties. The program scope is limited to ISI of the primary systems including the piping, vessel, and other components inspected in accordance with Section XI of the ASME Code. This is a progress report covering the programmatic work from October 1989 through September 1990.

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Executive Summary[1]

A multi-year program entitled the Evaluation and Improvement of NDE Reliability for Inservice Inspection of Light Water Reactors (NDE Reliability) was established at the Pacific Northwest Laboratory (PNL) to determine the reliability of current inservice inspection (ISI) techniques and to develop recommendations that would ensure a suitably high inspection reliability if fully implemented.

The objectives of this Nondestructive Examination (NDE) Reliability program for the Nuclear Regulatory Commission (NRC) include:

- Determine the reliability of ultrasonic ISI performed on the primary systems of commercial light-water reactors (LWRs).
- Use probabilistic fracture mechanics analysis to determine the impact of NDE unreliability on system safety and determine the level of inspection reliability required to ensure a suitably low failure probability.
- Evaluate the degree of reliability improvement that could be achieved using improved and advanced NDE techniques.
- Based on material properties, service conditions, and NDE uncertainties, formulate recommended revisions to Sections XI and V of the ASME Code and the Regulatory requirements needed to ensure suitably low failure probabilities.

The scope of the program is limited to the ISI of primary coolant systems, but the results and recommendations are also applicable to Class 2 piping systems.

The program consists of three basic tasks: a Piping task, a Pressure Vessel task, and a New Inspection Criteria task. Because of the problems associated with the reliable detection, correct interpretation, and accurate characterization of defects during ultrasonic testing/inservice inspection (UT/ISI) of piping, the major efforts during this reporting period were concentrated in the Piping task and the New Inspection Criteria task. However, some work was performed under the Pressure Vessel Task.

The major highlights during this reporting period were:

- <u>ASME Code Activity</u>

  Participation in ASME Section XI activities continued toward achieving Code acceptance of NRC-funded PNL research results to improve the reliability of nondestructive evaluation/inservice inspection (NDE/ISI). The final version of proposed Code Case N-471 entitled "Acoustic Emission for Successive Inspections, Section XI, Div. 1" received Main Committee (M.C.) approval following revisions to accommodate previous M.C. comments. This document was approved by the Board on Nuclear Codes and Standards (BNCS) and was published in the 1990 Addenda to ASME Section XI. Draft Code rules/requirements for computerized UT imaging systems (i.e., SAFT, et al.) were developed and presented to the ASME Section V Subgroup on Ultrasonic Testing (SGUT). Both the SGUT and Section V responses were quite favorable, and task group assignments have been made to accommodate comments and additional input. The proposed rewrite of Appendix IV on multifrequency ET of steam generator tubes was also approved by both the M.C. and the BNCS for publication in the 1990 Addenda to Section XI (to be issued in early 1991).

- <u>Pressure Vessel Inspection</u>

  <u>Analysis of PISC-II Data</u>. The objective of this task is to determine the capability of U.S. ultrasonic inservice inspection of reactor pressure vessels. This objective is to be accomplished by utilizing data from PISC-II round robin trials, modelling and limited experimental work to supplement areas not adequately addressed by modelling or round robin trials. Comments from the NRC were addressed and a revised document was resubmitted to NRC in June 1990. Additional comments have been received from the June draft, and these comments are being addressed. A final

[1]RSR FIN Budget No. B2289; RSR Contact: J. Muscara

version of this report will then be submitted to the NRC program manager.

Equipment Interaction Matrix. The objective of this work is to evaluate the effects of frequency domain equipment interactions and determine tolerance values for improving ultrasonic inspection reliability. An analysis is being performed to evaluate frequency domain effects using both computer model, to calculate the flaw transfer function, and experimental measurements, to verify calculated values and determine if other extraneous effects are important. Model calculations were performed to determine the sensitivity of 45° and 60° SV pulse-echo inspection results to changes in equipment parameters when thick (6-12 in.) steel sections are examined. The ASME Code requirements for center frequency tolerance were found to be inadequate. This result is consistent with our findings for thin section materials (< 6").

- New Inspection Criteria

  Work continued on assessments of the adequacy of existing ASME Code requirements for ISI and on developing technical bases for improved ISI requirements that will contribute to safe nuclear power plant operation. Development of a comprehensive probabilistic approach for improved inspection requirements moved forward. A major focus of this effort has continued to be participation in an ASME Research Task Force on Risk-Based Inspection Guidelines. Calculations during this reporting period have applied probabilistic risk assessment (PRA) to establish inspection priorities for pressure boundary systems and components. Plant-specific PRA studies have been for the Surry Unit 1 Nuclear Power Station. This study has been performed with the cooperation of Virginia Electric Power Company, and has involved plant visits for system walkdowns and detailed interviews with plant staff. Estimates of failure probabilities have been an important input to the risk-based calculations for Surry-1. An expert judgement elicitation was performed in May of 1990 to estimate rupture probabilities for components in four critical systems (reactor pressure vessel, reactor coolant system, low pressure injection system and auxiliary feedwater system).

- Consult on Field Problems

  The objective of this work is to provide a rapid response to urgent and unexpected problems as they are identified by the Office of Nuclear Reactor Regulation (NRR) and the Office of Nuclear Regulatory Research (RES). The major activity under this task during this period involved a request to critique EPRI Report NP-6090. This report reviewed the feasibility of using a special ultrasonic method to quantify the degree of corrosion occurring in Mark I containment vessel walls. The PNL review concluded that several problems should be addressed before this technique is used to perform a vessel inspection.

- Piping Inspection Task

  This task is designed to address the NDT problems associated with piping used in light water reactors. The primary thrust of the work has been on wrought and cast stainless steel since these materials are harder to inspect than carbon steel. However, many of the subtasks' results also pertain to carbon steel. The current subtasks are: mini-round robin report, piping inspection round robin report, qualification document, cast stainless steel inspection, surface roughness, field pipe characterization, and PISC-III activities.

  MRR Report. The Mini-Round Robin (MRR) subtask was conducted to provide an engineering data base for UT/ISI that would help: a) quantify the effect of training and performance demonstration testing that resulted from IEB 83-02, b) quantify the differences in capability between detecting long versus short cracks, and c) quantify the capability of UT/ISI technicians to determine length and depth of intergranular stress corrosion cracks (IGSCC). NUREG/CR-4908 entitled *Ultrasonic Inspection Reliability for Intergranular Stress Corrosion Cracks: A Round Robin Study of the Effects of Personnel, Procedures, Equipment, and Crack Characteristics* was published in July 1990 to document the work conducted under this subtask.

  Qualification Criteria for UT/ISI Systems. The objective of this subtask was to improve the reliability of UT/ISI through the development of new criteria and requirements for qualifying UT/ISI systems. NUREG/CR-4882 entitled *Qualification Process for Ultrasonic Testing on Nuclear Inservice Inspection Applications* was published in April 1990.

Cast Stainless Steel Inspection. The objective of this subtask is to evaluate the effectiveness and reliability of ultrasonic inspection of cast materials within the primary pressure boundary of LWRs. Activities for this work period included macro-structural classification of CCSS material within a PISC block, use of the Rayleigh critical angle technique to characterize CCSS macrostructures, and phase mapping of ultrasonic fields in CCSS.

Surface Roughness. The objective of this subtask is to establish specifications such that an effective and reliable ultrasonic inspection is not precluded by the condition of the surface from which the inspection is conducted. Activities included continued refinement of the model by the Center for NDE (CNDE) at Ames Laboratory, and development of better experimental procedures by PNL for obtaining quantitative data for comparing to the model prediction.

Field Pipe Characterization. The objective of this subtask is to provide pipe weld specimens that can be used for studies to evaluate the effectiveness and reliability of ultrasonic inservice inspection (UT/ISI) performed on BWR piping. It was finally decided that the PISC III program was no longer interested in the safe-ends. Therefore, these five safe-ends will be buried.

PISC III. This activity involves participation in the PISC-III program to ensure that the work addresses NDE reliability problems for materials and ISI practices on U.S. LWRs. This includes support for the co-leader of Action 4 on Austenitic Steel Tests (AST); providing five safe-ends from the Monticello plant; providing a sector of the Hope Creek reactor pressure vessel containing two recirculation system inlet nozzles, coordination of the inspections to be conducted by U.S. teams on the various actions; input to the studies on reliability and specimens for use in the parametric, capability, and reliability studies of the AST. During this reporting period, the efforts continued towards coming up with working solutions to the problem of specimens for the capability studies and working to obtain participation of teams from the USA.

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1.0 Introduction

The Evaluation and Improvement of NDE Reliability for Inservice Inspection of Light Water Reactors (NDE Reliability) Program at Pacific Northwest Laboratory (PNL) was established to determine the reliability of current inservice inspection (ISI) techniques and to develop recommendations that would ensure a suitably high inspection reliability if fully implemented. The objectives of this program for the Nuclear Regulatory Commission (NRC) are:

- Determine the reliability of ultrasonic ISI performed on commercial lightwater reactor (LWR) primary systems.
- Use probabilistic fracture mechanics analysis to determine the impact of NDE unreliability on system safety and determine the level of inspection reliability required to insure a suitably low failure probability.
- Evaluate the degree of reliability improvement that could be achieved using improved and advanced NDE techniques.
- Based on material properties, service conditions, and NDE uncertainties, formulate recommended revisions to Section XI of the Regulatory and ASME Code requirements needed to ensure suitably low failure probabilities.

The scope of this program is limited to ISI of primary coolant systems, but the results and recommendations are also applicable to Class 2 piping systems.

The program consists of three basic tasks: a Piping task, a Pressure Vessel task, and a New Inspection Criteria task. Because of the problems associated with the reliable detection and accurate characterization of defects during ultrasonic testing/inservice inspection (UT/ISI) of piping, the major efforts were concentrated in the Piping task and the New Inspection Criteria task. However, work was conducted on the Pressure Vessel Task and will be receiving greater emphasis in future reporting periods.

This report is divided into the following sections.

- ASME Code Related Activities
- Pressure Vessel Inspection
- New Inspection Criteria
- Consult on Field Problems
- Piping Task Activities

2.0 ASME Code Related Activities

2.1 Summary

Participation in ASME Section XI activities continued toward achieving Code acceptance of NRC-funded PNL research results to improve the reliability of nondestructive evaluation/inservice inspection (NDE/ISI). The final version of proposed Code Case N-471 entitled "Acoustic Emission for Successive Inspections, Section XI, Div. 1" received Main Committee (M.C.) approval following revisions to accommodate previous M.C. comments. This document was approved by the Board on Nuclear Codes and Standards (BNCS) and was published in the 1990 Addenda to ASME Section XI. Draft Code rules/requirements for computerized UT imaging systems (i.e., SAFT, et al.) were developed and presented to the ASME Section V Subgroup on Ultrasonic Testing (SGUT). Both the SGUT and Section V responses were quite favorable, and task group assignments have been made to accommodate comments and additional input. The proposed rewrite of Appendix IV on multifrequency ET of steam generator tubes was also approved by both the M.C. and the BNCS for publication in the 1990 Addenda to Section XI (to be issued in early 1991).

2.2 Introduction

The objective of this task is to develop and/or evaluate new criteria and requirements for qualifying ultrasonic testing/inservice inspection (UT/ISI) and other NDE/ISI systems. The ultimate goal is for these criteria and requirements to be incorporated into Sections V and XI (SC-V and SC-XI) of the ASME Boiler and Pressure Vessel Code. If that goal cannot be met or if the requirements adopted by the ASME Code are inadequate, PNL may also be requested to prepare draft input for a Regulatory Guide as a back-up approach.

The document NUREG/CR-4882 entitled "Qualification Process for Ultrasonic Testing in Nuclear Inservice Inspection Applications" was used as the basis for a "proposed Appendix VII" that was developed in 1986-1987 by an ASME Ad Hoc Task Group. This proposed appendix to ASME Section XI was subsequently restructured and revised by the SC-XI Subgroup on Nondestructive Examination (SGNDE). The Ad Hoc Task Group document was restructured as two companion Mandatory Appendices by the SGNDE and considerable effort was required (both by PNL and the Code Committees) during the next three years to achieve final ASME Code approval of these documents. These two appendices are generally identified as a) Appendix VII on personnel training and qualification and b) Appendix VIII on UT system performance demonstrations. Additionally, administrative assistance was provided in support of related efforts to achieve Code acceptance and publication of a proposed Code Case on acoustic emission, a rewrite of Appendix IV to SC-XI to accommodate the multifrequency eddy current (ET) equipment that is used for ISI of steam generator tubes, and a Code Case to permit ET in lieu of visual examination for coated structural components (containment vessels).

2.3 Status of Work Performed

Proactive participation of PNL personnel in ASME Code activities continued toward achieving Code acceptance of NRC-funded PNL research to improve the reliability of NDE/ISI. During this reporting period, meetings of the ASME Section XI Subcommittee were attended November 6-9, 1989 in Orlando, FL, January 15-18, 1990 in Miami, FL, May 14-18, 1990 in Nashville, TN, and August 27-30, 1990 in Mystic, CT. In addition, meetings of the Section V Subcommittee on Nondestructive Examination, including the Subgroup on Ultrasonic Testing (SGUT), the Subgroup on Acoustic Emission (SGAE), and the Subgroup on General Requirements/Surface Examination (SGGR/SE) were attended February 5-6, 1990 in New York City, NY, May 14-18, 1990 in Nashville, TN, and September 10-12, 1990 in Pittsburgh, PA.

Agendas and minutes of the SGNDE meetings held in conjunction with Section XI Subcommittee meetings were prepared and distributed. J. C. Spanner serves as Secretary of the SC-XI SGNDE and as a member of the Working Group on Surface Examination and Personnel Qualification. T. T. Taylor chairs a Special Task Group to develop acoustic emission criteria and requirements, and serves as a member of the Working Group on Volumetric Examination and Procedure Qualification. A topical report (NUREG/CR-4882) was prepared and published to describe the criteria and requirements that were developed, as well as to document the background, rationale, and overall activities that have been conducted under this task.

Appendix VII on personnel training and qualification received final Code Committee approval in late 1988 and was published in the 1990 Addenda to ASME Section XI. Proposed Appendix VIII on UT/ISI performance demonstrations received final ASME Code approval in late 1989 and was published in the 1990 Addenda to ASME Section XI.

Proposed new Section XI criteria and requirements for applying the acoustic emission method for Section XI applications have been developed. A proposed new Code Case on acoustic emission has been developed and approved by the SGNDE and SC-XI for submittal to the M.C. This Code Case, entitled "Acoustic Emission for Successive Inspections Required by Section XI, Div. 1," received M.C. approval in January 1990 following revisions to accommodate comments accompanying three M.C. negative votes. A finalized version of this proposed Code Case, which incorporated the M.C. comments and various other approved changes, was submitted for BNCS letter ballot in March 1990. Notification of final BNCS approval of this document was received in May 1990, and this document was published in Supplement No. 5 in mid-1990 with the designation Code Case N-471.

A proposed rewrite of Appendix IV, entitled "Eddy Current Examination of Nonferromagnetic Steam Generator/Heat Exchanger Tubing," was revised to accommodate comments submitted by four M.C. members. This document, which represents an upgrading of requirements for consistency with industry practice regarding multifrequency eddy current testing, was approved by the M.C. during the January 1990 Code meetings, and submitted for BNCS letter ballot. By mid-year, this complete rewrite of Appendix IV had been approved by BNCS for publication in the 1990 Addenda to Section XI (to be issued in early 1991).

Proposed new Code requirements are being drafted to encourage broader utilization of the synthetic aperture focusing technique (SAFT) technology that has been developed under NRC-sponsored research programs at PNL and the University of Michigan. These proposed requirements were submitted to the ASME Section V Subgroup on Ultrasonic Testing (SGUT) to address the need for Code rules to cover the computerized UT imaging systems that are being utilized by the NDE/ISI industry for examining structurally important plant components. This proposal involves adding a new paragraph to Article 4 of ASME Section V under the new heading "T-435 Computerized Imaging Systems." In addition, a series of Nonmandatory Appendices to Article 4 were recommended to provide the technical detail necessary to describe selected computerized UT imaging systems. Based on SGUT discussion of these proposed new Code rules, the requirements sections (T-435 and its subparagraphs) were significantly expanded including the addition of special calibration blocks for performance evaluation purposes. Concurrently, the appendix material describing the SAFT concept and equipment-related details was reduced in volume by about one-half. This draft now provides a model for use in the development of similar appendices describing other computerized imaging systems.

A letter requesting technical input on computerized UT imaging systems was distributed to the 11 known manufacturers/vendors of such equipment. Good quality input was received from two of these organizations; however, the information received from three others was of limited value in the context of this activity, and the other six manufacturers/vendors provided no input at all. In view of this disappointing response, the SGUT determined that the draft T-435 requirements paragraphs should be expanded and the concept of including system descriptions in a Non-mandatory Appendix was tentatively deferred. Cognizant SGUT task members accepted assignments to prepare/expand specific paragraphs based on the PNL-proposed T-435 requirements.

Information for use in value-impact assessments was developed for submittal to the CRGR. Value-impact assessment information was developed for the new Appendix VII on training and qualification of UT personnel; the new Appendix VIII for performance demonstration of UT/ISI systems; and the 1988 Addenda changes to Table IWB-2500-1, Examination Category B-A, Reactor Pressure Vessel Shell Welds.

The EPRI NDE Center continues to assist the Industry Ad Hoc Committee on Implementation of Appendix VIII. This includes logistics support, technical advice, sample design, and cost estimates for specimen fabrication and program administration. The Ad Hoc Committee on Appendix VIII Implementation held an information meeting September 18, 1990 to update industry on the current status of Ad Hoc Committee activities. At that meeting, utilities representing 87 nuclear units had indicated a willingness to participate in funding the initial setup of the Appendix VIII program, eight had declined to participate, and the remainder were undecided.

The Section V Subgroup on Acoustic Emission (SGAE) has decided to prepare a new Section XI Article on In Situ Monitoring of Pressure Vessels. A proposed scope has been drafted, and the Chairman's stated intent is to base this new Article on the recently approved Section XI Code Case (N-471). The SC-V SGAE is also attempting to develop a new Code Case for Section VIII applications; however, Section VIII has exhibited limited interest in this activity to date.

2.4 Future Work

Minutes for all ASME Section XI SGNDE meetings are assembled for distribution to the approximately 65 recipients on the SGNDE mailing list. Work continues on drafting generic sizing requirements for Supplement 12 of Appendix I, and on new Supplements to Appendix VIII to address cast stainless steel components, dissimilar metal welds, and pre-examination surface conditioning. Future Section XI meetings will be held December 10-14, 1990 in Anaheim, CA; February 4-7, 1991 in San Diego, CA; May 20-24, 1991 in Orlando, FL; August 26-29, 1991 in Pittsburgh, PA; and November 18-21, 1991 in Anaheim, CA.

The proposed new Section V requirements for computerized UT imaging systems will be revised to incorporate SGUT comments. When approved by the SGUT, this document will be submitted for Section V approval.

3.0 Pressure Vessel Inspection

3.1 Analysis of PISC-II Data

3.1.1 Summary

The objective of this task is to determine the capability of U.S. ultrasonic inservice inspection of reactor pressure vessels. This objective is to be accomplished by utilizing data from PISC-II round robin trials, modelling and limited experimental work to supplement areas not adequately addressed by modelling or round robin trials. Comments from the NRC were addressed and a revised document was resubmitted to NRC in June 1990. Additional comments have been received from the June draft, and these comments are being addressed. A final version of this report will be submitted to the NRC program manager.

3.1.2 Introduction

The pressure vessel inspection task is divided into three subtasks which are:

- **PISC II Re-analysis** - The initial effort in this task is an analysis of data gathered during the PISC-II round robin trials. Ultrasonic inspection data was gathered on four heavy section steel components which included two plates and two nozzles configurations. A total of 45 teams from the Common Market, Japan, and the United States participated in the round robins.

- **RPV Research** - The focus of this activity is to track the work being performed under the PISC III program. This means to ensure that the PISC III work of the Action 2 on Full Scale Vessel Tests (FSV) and the Action 3 on Nozzles and Dissimilar Metal Welds provides useful information for conditions and practices in the USA. To track the PISC III work and relay points of interest and concern to the NRC that may arise from the analysis of the newly created and evolving data base.

3.1.3 Status of Work Performed

A summary of the work performed for each subtask is provided below.

3.1.3.1 PISC II Re-analysis

A summary of the analysis was provided in the last semi-annual report (NUREG/CR-4469, Vol.11). During this time period, PNL has received comments from the NRC project manager and is incorporating the comments in the report. The comments made by the NRC project manager were:

- The limitations that exist with the PISC-II data base must be more clearly delineated in the report. Examples would include, no false call information and problems associated with "cleaned data."

- The sizing analysis in the report uses categories that do not clearly describe the sizing techniques used by teams in the trials. More clearly defined sizing categories must be described in the report.

- The use of length as a major variable in Probability of Detection (POD) analysis should be de-emphasized and the analysis should concentrate on POD vs Depth.

- If the data allows, the report should include a section that examines individual technique capability.

Comments from the NRC were addressed and a revised document was transmitted to the NRC in June of 1990. Additional comments have been received from the June draft, these comments are being addressed and a finalized version of the report will be submitted to the NRC.

3.1.3.2 RPV Research

The objective of this work is to track the work that is currently being performed under the PISC III program. Of particular interest is the work being conducted in Actions 2 and 3. These actions will provide useful information concerning the capability to inspect nozzles and dissimilar metal welds and begin to address some aspects of the ability of advanced techniques to accurately size flaws and some aspects of the reliability to inspect actual vessels. The initial results from these studies will begin to be made available to the PISC III Management Board in late 1991. At this time, there are no results available; but these studies should provide some very good and useful data bases and conclusions in the near future.

Based on the results and data bases from the PISC III program, deficiencies will be identified and a program to provide the necessary supplemental information will be developed and implemented. But at this time, the activity is focused on tracking the PISC III program and the results being developed in this program.

3.2 Equipment Interaction Matrix

3.2.1 Summary

The objective of this work is to evaluate the effects of frequency domain equipment interactions and determine tolerance values for improving ultrasonic inspection reliability. An analysis is being performed to evaluate frequency domain effects using computer models to calculate the flaw transfer function, and experimental measurements to verify calculated values and determine if various extraneous effects are important.

The primary purpose of this work was to apply an analysis method developed for thin steel sections to thicker flat sections and to analyze experimental studies. Also emphasized, were efforts devoted to presenting results obtained in the previous reporting period. Related activities included:

- Model calculations were performed to determine the sensitivity of 45° and 60° SV pulse-echo inspection results to changes in equipment parameters when thick (6 to 12 inch) steel sections are examined. The results were essentially the same as those found for the thin section inspection model study. The ASME Code requirements for equipment center frequency were again found inadequate, just as was found for thin sections.
- Results from an experiment performed, in the previous reporting period, to measure the ultrasonic equipment parameter sensitivity and to test model predictions as they effect ASME code requirements were analyzed. These results were consistent with the model, but were not as sensitive to center frequency changes as predicted by the model.
- The most significant results from equipment interaction studies on thin, flat steel sections were presented at The Review of Progress in Quantitative Nondestructive evaluation Conference in July 1990.
- The model's performance was improved by modifying its computer software to run 10 times faster, and made it more useful by backing-up all related software on disks and documented each element of the program.
- Completed a draft version of the NUREG topical report on the work performed to date.
- Reviewed PISC III modeling paper and made recommendations.
- A presentation was given to ASME Section XI Code committees, and a paper was written for the 10th International Conference on NDE in The Nuclear Industry.

3.2.2 Introduction

The goal of this work is to define operating tolerance requirements for UT/ISI equipment that minimize the effects of frequency domain interactions, thus, improving ISI reliability. This work will determine the acceptability of equipment specifications in ASME Code Case N-409-1. The current specifications are based on engineering judgement, rather than an analytical foundation. The Interaction Matrix Study will provide this technical foundation. Both thin (piping) and thick steel sections (pressure vessels) are being evaluated.

The following work was completed during previous reporting periods:

- Mathematical models were developed for UT/ISI equipment. This work was presented for peer review (Mart and Doctor 1986).
- A mathematical model was developed to calculate the transfer functions (frequency responses) of specular reflection from smooth planar defects, and the model was used to identify worst-case defects for frequency domain equipment interac-

tions. A paper on the model was presented at the 1988 conference (Green and Mart 1989).

- Equipment bandwidth and center frequency sensitivity studies were performed for 45° SV inspection of thin sections (piping) using calculated worst-case flaw transfer functions. The model indicated that the ASME Code Case bandwidth tolerance of ±10% is sufficient to ensure reliable inspection, but the center frequency tolerance is not adequate to ensure reliable inspection of certain calculated worst-case flaws with narrow band UT/ISI systems.

- The first draft of the milestone NUREG report entitled *The Interaction Matrix Study: Models and Equipment Sensitivity Studies for the Ultrasonic Inspection of Thin Wall Steel Piping* was reviewed by the PNL program manager. A suggestion was made that the interaction matrix study be extended to include 60° SV pulse-echo inspection.

- 60° SV model calculations were made to determine the difference in sensitivity between 45° and 60° SV pulse-echo inspection equipment parameters. The 45° SV inspection was found to be slightly more sensitive to changes in equipment parameters. However, the model indicated (in either case) that the ASME Code requirements for equipment center frequency are inadequate.

- An experiment was performed to measure the ultrasonic equipment parameter sensitivity and to test model predictions for worst-case defects per the model.

3.2.3 Status of Work Performed

The following work was completed during this reporting period:

- Model calculations were performed to determine the sensitivity of 45° and 60° SV pulse-echo inspection results to changes in equipment parameters when thick (6 to 12 inch) steel sections are examined. The results were essentially the same as those found for thin section inspection model study. The ASME Code requirements for equipment center frequency were again found inadequate, just as found for thin sections.

- Results from an experiment performed, in the previous reporting period, to measure the ultrasonic equipment parameter sensitivity and to test model predictions as they affect ASME code requirements were analyzed. These results are consistent with the model, but are not as sensitive to center frequency changes as predicted by the model.

- Presented the most significant facts resulting from equipment interaction studies on thin, flat steel sections at The Review of Progress in Quantitative Nondestructive Evaluation Conference in July 1990.

Results from an experiment performed, in the previous reporting period, to measure the ultrasonic equipment parameter sensitivity and to test model predictions as they affect ASME Code requirements were analyzed. These results are consistent with the model, but are not as sensitive to center frequency changes as predicted by the model. This is due to necessary simplifications that are built into the model to obtain reasonable run-times, on presently available computer systems. Therefore, the laboratory measurements show the model capable of pinpointing worst-case flaws that can then be examined using the same equipment found in field applications, for inspecting a nuclear power plant.

Model calculations were performed to determine the sensitivity of 45° and 60° SV pulse-echo inspection results to changes in equipment parameters when thick (6 to 12 inch), flat steel sections are examined. The results were essentially the same as those found for the thin section inspection model study. The model accounted for the greater attenuation a signal must experience when passing through a thicker material. The ASME code requirements for equipment center frequency were again found inadequate, just as was found for thin sections.

The most significant facts resulting from equipment interaction studies on thin, flat steel sections were presented at The Review of Progress in Quantitative Nondestructive Evaluation Conference in July 1990. This work was entitled "The Effect of Equipment Bandwidth and Center Frequency changes on Ultrasonic Inspection

Reliability: Are Model Results Too Conservative?". A summary will appear in *Review c' Progress in Quantitative Nondestructive Evaluation*, Vol. 10B.

3.2.4 Future Work

The following work remains to be completed:

- Expand the flaw model to include curved sections (nozzles) and perform equipment parameter sensitivity studies for thick sections (reactor pressure vessels).
- Verify curved surface model calculations experimentally.
- Report findings from curved section study to ASME Code.
- Extrapolate results from curved surface study to all important surface geometries.

4.0 New Inspection Criteria Task

4.1 Summary

Work continued on assessments of the adequacy of existing ASME Code requirements for ISI and on developing technical bases for improved ISI requirements that will contribute to safe nuclear power plant operation. Development of a comprehensive probabilistic approach for improved inspection requirements has moved forward. A major focus of this effort has continued to be participation in an ASME Research Task Force on Risk-Based Inspection Guidelines.

Calculations during this reporting period have applied probabilistic risk assessment (PRA) to establish inspection priorities for pressure boundary systems and components. Plant-specific PRA studies have been performed for the Surry Unit 1 Nuclear Power Station. This study has been performed with the cooperation of Virginia Electric Power Company, and has involved plant visits for system walkdowns and detailed interviews with plant staff. Estimates of failure probabilities have beet. an important input to the risk-based calculations for Surry-1. An expert judgement elicitation was performed in May of 1990 to estimate rupture probabilities for components in four critical systems (reactor pressure vessel, reactor coolant system, low pressure injection system and auxiliary feedwater system).

4.2 Introduction

This task is directed to the development of improved inservice inspection (ISI) criteria using risk-based methods, with the long-range goal to propose changes for consideration by ASME Section XI. These improved criteria will help to establish priorities for selecting systems, components and structural elements for inspection, and will help to determine the extent, frequency, and method of examination. The objective is to ensure that ISI programs ensure a suitably low failure probability, and thus contribute in an effective manner to safe nuclear power plant operation.

In past work, we have reviewed and evaluated various concepts for probabilistic inspection criteria, and have interacted with other industry efforts, notably through a newly organized ASME Research Task Force on Risk-Based Inspection Guidelines. During FY89 we completed pilot applications of PRA methods to the inspection of piping, vessels, and related components for a sample of eight representative nuclear power plants (Surry-1, Zion-1, Sequoyah-1, Oconee-3, Crystal River-3, Calvert Cliffs-1, Peach Bottom-2 and Grand Gulf-1). The results of this study can be found in Vo et. al. 1990. In summary, the results provide generic insights that could be extrapolated from the eight plants to specific classes of light water reactors. While a few exceptions are noted, the PRA-based priorities for inspection of systems were generally correlated with current ASME Section XI requirements for Class 1, 2, and 3 systems.

Work during this reporting period addressed inspection priorities at the more detailed component level, and focused on plant-specific calculations for the Surry Unit 1 Nuclear Power Station. Results of this work are described below.

4.3 Status of Work Performed

4.3.1 ASME Task Force on Risk-Based Inspection Guidelines

During this reporting period we have continued to develop approaches for risk-based inspection requirements. Activities in this area have involved Pacific Northwest Laboratory (PNL) participation on a special ASME Research Task Force on Risk-Based Inspection Guidelines, with Dr. F. A. Simonen and Dr. B. F. Gore serving as members of the Task Force and Mr. T. V. Vo participating as an honorary member. The ASME group has been identified by PNL as an effective route to achieve long-range goals for improved inspection criteria. The initial focus of the ASME Task Force has been on nuclear power applications, and on the development of practical recommendations for use of risk-based methods that can be recommended for consideration by ASME Section XI.

There were three meetings of the ASME Research Task Force during this reporting period as follows:

- December 12-13, 1989 at San Francisco, California
- May 16-17, 1990 at Nashville, Tennessee
- August 29-30, 1990 at Mystic, Connecticut.

The Phase I work of the Task Force has produced a general document that recommends and describes appropriate methods for establishing inspection guidelines using risk-based approaches for any facility or structural system. A final draft of this document was distributed for peer review in August of 1990, and was published by ASME during the first quarter of 1991.

Preparation of a second document on the special topic of nuclear power applications was started during this reporting period (Volume 2 - Part 1). This document will recommend and describe specific methods to be used in developing risk-based inspection plans for nuclear power facilities. Publication is scheduled for the later part of 1991.

Future efforts of the ASME Research Task Force will apply the recommended risk-based methodologies to develop improved inspection programs for nuclear power plant components (Volume 2 - Part 2). This work will be summarized in a document that is scheduled for publication in the later part of 1993. This document will make recommendations for consideration by ASME Section XI.

4.3.2 Plant Specific PRA Application to Surry-1

Work continued on a major effort that involves the application of existing probabilistic risk assessments (PRA) to establish inspection priorities for pressure boundary systems and components. A pilot application of PRA methods to the Surry-1 plant was begun during this reporting period.

The Surry-1 work applied a methodology (Vo et al. 1989) that uses the results of PRA's in combination with the techniques of failure modes and effects analysis (FMEA) to identify and prioritize the most risk-important systems [illegible] components [illegible] nuclear power plants. The spe[illegible] selected for initial analysis were the [illegible] vessel (RPV), the reactor coolant (RCS), [illegible]ssure injection (LPI) (including the accumulato[illegible]d the auxiliary feedwater (AFW).

Core damage frequency (Level-I PRA) was used in this study as the bottom line risk measure. FMEA results were used to calculate the relative importance of each component within the [illegible] addressed. The calculated importance ind[illegible] the expected consequences of failure of the [illegible] (from the Surry-1 PRA) and the expected pro[illegible]f failure (rupture) of the component. Rupture probability estimates for the Surry-1 components were obtained from an expert judgement elicitation.

Staff from the Virginia Electric Power Company (VEPCO) have been actively participating in the pilot study. It was important to assure that the plant models were as realistic as possible and reflected plant operational practices. Visits were made to the Surry-1 plant for system walkdowns and discussions with plant operational technical staff.

Table 4.1 shows a preliminary list of the relative risk-importance of components within the selected Surry-1 systems (stated as categories of High, Medium, and Low risk-importance). On the basis of core damage frequency, the most risk-important components are those located within the beltline region of the reactor pressure vessel. Relatively "High" rankings were estimated for pipe segments of the AFW system, and also for specific segments within the LPI system. A large number of components associated with redundant flow paths fell into the lowest ranking category (e.g. cross connected lines in the AFW and LPI systems).

In future work the results of Table 4.1 will be further quantified with refined inputs to the FMEA calculations. Indirect effects of component ruptures will incorporated into the evaluation. Additional systems will also be evaluated to give a comprehensive ranking of all the most important components within the Surry-1 plant systems that should be addressed when assigning the priorities for inservice inspection. These risk-based priorities can then be compared with current inservice inspection requirements as specified by Section XI of the American Society of Mechanical Engineers Code. The objective is to identify needed improvements to current ISI plans. These results will be made available to the ASME Research Task Force as recommendations for consideration by ASME Section XI.

4.3.3 Expert Judgement Elicitation for Rupture Probabilities

The risk-based studies of the Surry-1 plant have required estimates of rupture probabilities on a detailed component-by-component level. Because neither sufficient data from operating experience nor detailed frac-

Table [illegible] Risk-Important Components for Selected Systems at Surry-1[(a)]

Ranking[(b)]	Component	System[(c)]
High	Beltline Region	RPV
	Reactor Head Areas including CRDMs and Instrument Lines	RPV
	Steam Generators to Containment Isolation Valves	AFW
	LPI Discharge Headers to Hot Leg and Cold Leg Injection Lines	LPI
	Pressurizer Spray Lines	RCS
Medium	AFW Suction Lines	AFW
	LPI Suction Lines	LPI
	Pressurizer Surge Line	RCS
	Pressurizer Relief Lines	RCS
	Other RPV Components (e.g., studs, flanges, etc.	RPV
	Accumulator Sample/Drain Lines	LPI
	Pipe Segments within the RCS Loops (e.g., loop welds, inst. lines, etc.	RCS
	AFW Pump Discharge Lines	AFW
	LPI Pump Discharge Lines	LPI
Low	Other AFW Components (e.g., cross-connected lines, etc.)	AFW
	Other LPI Components (e.g., cross-connected lines, etc.)	LPI

(a) Based on preliminary estimates using estimated median values.
(b) High component ranking corresponds to core damage frequency of about 1.0E-06 to 1.0E-08 per plant year.
Medium component ranking corresponds to core damage frequency of less than 1.0E-08 to 1.0E-10 per plant year.
Low component ranking corresponds to core damage frequency of less than 1.0E-10 per plant year.
(c) RPV = Reactor Pressure Vessel; AFW = Auxiliary Feedwater; LPI = Low Pressure Injection; RCS = Reactor Coolant System.

ture mechanics analyses were available, an expert judgment elicitation was conducted to estimate the needed rupture probabilities. Historically there have been very few structural failures of the pressure boundary components of interest, thus giving a high level of uncertainty to statistical estimates of failure probabilities. Probabilistic fracture mechanics models have addressed a few components of interest, but these calculations also have a high degree of uncertainty.

To overcome the data deficiency, PNL conducted an expert judgment elicitation meeting on May 8-10, 1990 at Rockville, Maryland to address the issue of failure probabilities. The goal was to obtain numerical estimates for probabilities of catastrophic or disruptive failures in pressure boundary systems and components in pressurized water reactors (PWRs). The systems addressed by the elicitation were the reactor pressure vessel, reactor coolant, low pressure injection (including the accumulators), and auxiliary feedwater.

The expert elicitation was performed using a systematic procedure, which closely followed the approach used for the NUREG-1150 PRAs (NRC 1990 and Wheeler et al. 1989). Obtaining the experts' estimates of rupture probabilities was the primary objective for the elicitation. The selected experts were selected to have knowledge of the subject matter as well as an understanding of rules governing the form in which they were to respond. Thus, the experts had demonstrated expertise by publications, hands-on experience, and managing or performing research in the areas related to the issues. Experts were selected to be versatile enough to be able to address several issues and have sufficient PRA experience to consider how these issues would be used in a risk analysis. Finally, experts were selected to represent as wide a perspective of the issues as possible, and willing to be elicited under the methodology used.

The expert elicitation meeting began by outlining objectives, and by describing the types of information that PNL was seeking from the meeting discussions, followed by a summary of the overall NDE Reliability Program being conducted at PNL. For each system addressed, a formal presentation was provided. The presentations covered technical descriptions, historical component failure mechanisms, elicitation statements, suggested approaches, questionnaire forms, and other material that supported the evaluation of rupture probabilities.

The discussion of rupture probabilities for each system involved the experts, the observers, and project team analysts. All experts were encouraged to participate in the discussions. Knowledge from experts regarding plant design and operation, failure history, material degradation mechanisms, recomposition and aggregation of the data, etc. were brought into the discussions. Since the process was designed to take advantage of the diversity of the knowledge, experts were encouraged to seek their own estimations and no effort was made to seek a consensus among the experts on estimated rupture probabilities.

For most components, the panel members all considered the fact that no ruptures have occurred, but information from the literature and discussion sessions did suggest to them that certain failure modes were possible. Data exist regarding the structural degradation due to erosion/ corrosion, fatigue, etc., but several experts also expressed the view that other mechanisms could contribute to future failures. Each expert then completed questionnaire forms that addressed location-specific rupture probabilities for the systems of interest. These data covered best estimates of probabilities, uncertainty estimates and the rationale for these estimates. Members of the VEPCO staff also participated in the elicitation through a special meeting, and provided additional Surry-1 plant-specific data for consideration by the expert panel.

Following a review of the data provided, the results were compiled into distributions by PNL staff. The distributions determined a "best" estimate for every component present in the data set.

Figure 4.1 shows the failure probability estimates for components within the reactor pressure vessel. The plots present distributions associated with the expert population. Similar plots were produced for all the other selected systems, e.g., reactor coolant, low pressure injection, and auxiliary feedwater systems (not shown). The complete set of plots for all systems indicate estimates of component rupture probabilities that vary between 1.0E-09 and 1.0E-03 failures/year, depending on the systems, components within systems, and component locations. For a given component within a particular system, the quartile range generally represents variations between a factor of 10 to 100, indicating the variation between the experts' estimates.

For the reactor pressure vessel (Figure 4.1), the results indicate that the **highest** failure probability estimates are for those components located within the vessel heads [e.g., instrument lines and control rod drive mechanisms (CRDMs)]. The high estimates for the CRDMs were primarily due to the high vulnerability to leakage believed by the expert panel. The high failure probability estimates for the instrument lines were due to occurrences of thimble tube cracking and leakage which have been reported in these components at Surry and at other operating PWRs.

In summary, the expert elicitation produced a set of component rupture probabilities for various components within selected systems at Surry-1 as a function of location, and failure mechanisms such as stress, thermal stratification, etc. Accomplishments from the expert judgment elicitation greatly enhanced the realism and credibility of the Surry plant analyses. Access to utility staff was found to be a very important factor for elicitation meetings. It is absolutely vital that detailed plant-specific design, configuration, operating experience, etc., be provided and brought into discussion at the meetings to help experts better assess the desired component failure probabilities. Other notable accomplishments gained from the expert judgment meeting included an enhancement of PNL's risk-based methodology in modeling parameter estimates for specific fault events, and guidance for incorporating innovative recovery actions in support of the improved inspection priorities of nuclear power plant components.

4.4 Future Work

Future activities on the New Inspection Criteria Task will include:

- Continuing support of the ASME Research Task Force on Risk-Based Inspection Guidelines.
- Expert elicitation for rupture probabilities on the remaining four systems at Surry-1.
- Complete the ISI prioritization for components at Surry-1.

The long range objective will be to develop improved criteria for inservice inspections (what, where, when, and by what method) using risk-based methods. The pilot calculations serve to demonstrate the feasibility of the proposed approach, and will focus on the component level to establish inspection priorities. Other calculations will be used in the development of risk-based inspection programs for the high priority components. These inspection programs will make use of information on the probabilities and consequences of component failures to assign target values of probabilities that are to be maintained by inservice inspection. Probabilistic fracture mechanics and decision analysis methods will identify inspection strategies that meet criteria for both safety and cost effectiveness. Output from the New Criteria Task will be made available to the ASME Research Task Force on Risk-Based Inspection Guidelines for their use in preparing a document that will recommend risk-based inspection programs for codes and standards consideration.

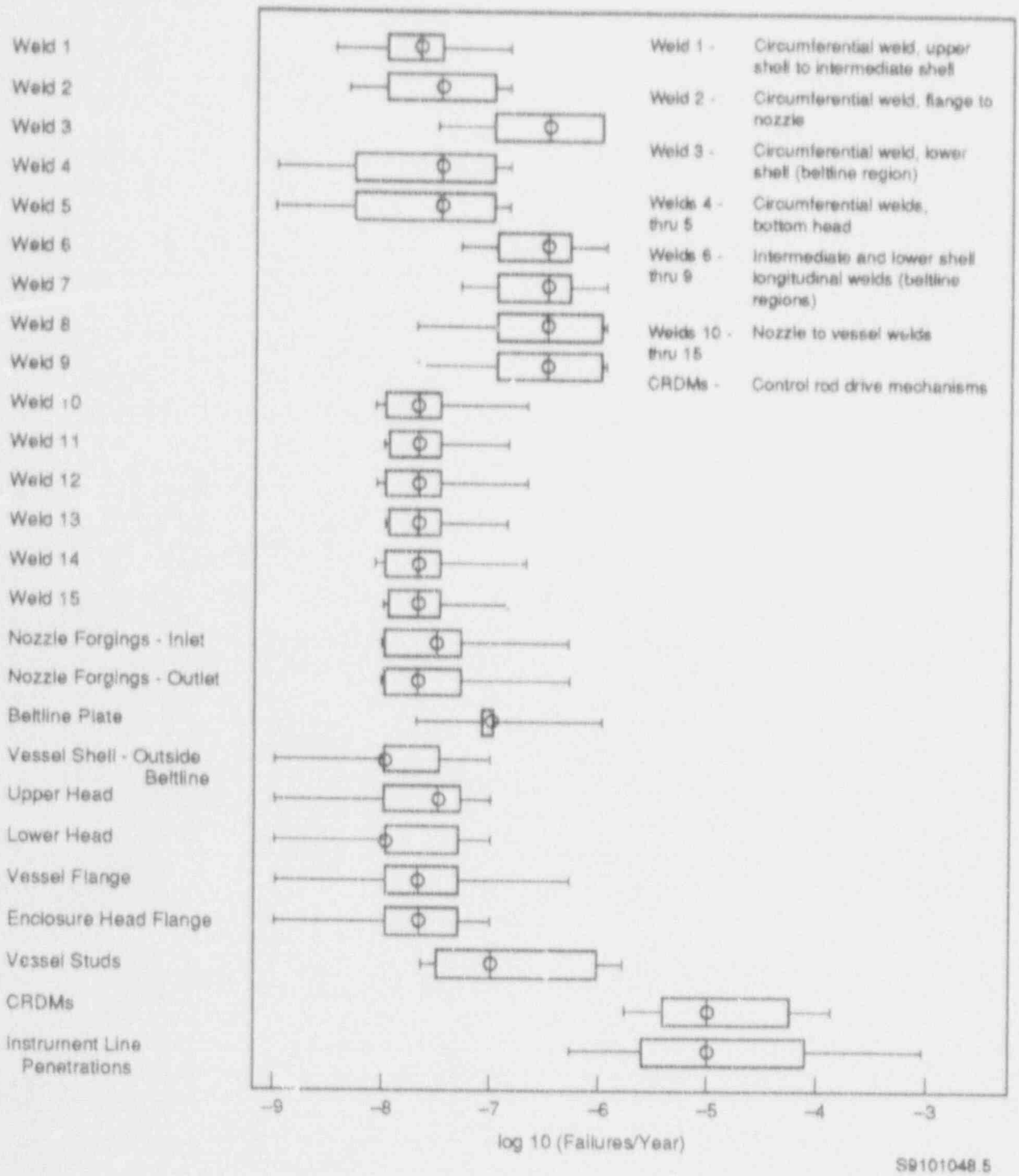

Figure 4.1. Failure Probability Estimates for the Reactor Pressure Vessel Components

5.0 Consult on Field Problems

5.1 Introduction and Summary

The objective of this work is to provide a rapid response to urgent and unexpected problems as they are identified by the Office of Nuclear Regulatory Research (RES). The major activity under this task involved a request to critique EPRI Report NP-6090. This report reviewed the feasibility of using a special ultrasonic method to quantify the degree of corrosion occurring in Mark I containment vessel walls. The PNL review concluded that several problems should be addressed before this technique is used to perform a vessel inspection.

5.2 Status of Work Performed

A critique of EPRI Report NP-6090 was performed in response to a request from the RES program manager of the PNL NDE Reliability Program. The inspection problem defined in the report was to quantify the degree of corrosion occurring in Mark I containment vessel walls. The vessel wall was reported to be a ferritic steel structure approximately 25-mm thick. Standard techniques were stated as not being applicable due to the physical restrictions imposed by 1) a bed of gravel that surrounds the outer surface and 2) a concrete floor within the structure. The purpose of the EPRI report was to determine the feasibility of the proposed technique. Therefore, apparent weaknesses of the technique's reported status should not be viewed as inherent limitations, but more as the limitations as they existed at the end of Phase I development. The critique was performed by M. S. Good of PNL and was provided to the RES program manager in October 1989.

The proposed technique as described in the report used electromagnetic acoustic transducers (EMATs) to transmit and receive horizontally polarized shear waves at 50 kHz. This wave mode was selected since these waves were expected to have low scattering and less attenuation from a bonded concrete floor, surface loading of the gravel, weld crowns, and weld roots. The frequency was selected such that a symmetric wave mode would be generated; i.e., all particle motion throughout the wall thickness would move in phase. The reported criteria for this was that the wave length be greater than 1.6 X the plate thickness. To test the technique, pulse-echo and through-transmission techniques were used on a large flat plate containing two 100-mm long notches with a 45% through-wall depth. One notch had planar sides while the mid-region of the other was extended to make a semi-circular surface.

The PNL review concluded that although technique feasibility was shown, various problems need to be addressed before the technique is used to perform an inspection. These include the following:

1. An accurate description of corrosion as it is expected in the field is needed so that representative flaws can be made in various standards.

2. Flaw detection should be demonstrated for all shapes, sizes, and types of expected corrosion. Although the 1- to 3-m zone under the concrete floor was listed as an area of suspect corrosion, a conservative argument w[illegible] flaws having a low acoustic response are [illegible]e throughout the entire range of 17-m s[illegible]lso be demonstrated.

3. Adequate discriminating ability must be demonstrated between corrosion defects and geometrical reflectors. Synthetic aperture processing was recommended by the EPRI report; however, the use of this technique to improve the ability to discriminate should be demonstrated to be effective and reliable.

4. Scattering and attenuation from objects such as the cement floor were reported by the EPRI report as being small; however, more data are needed to confirm this.

5. The EMATs used in the EPRI study generate forward and backward propagating waves with the latter being a possible noise source with geometric reflectors. Adequate signal-to-noise ratios, means of discriminating these spurious signals from possible flaw signals, or a means of propagating only a forward propagating wave need to be addressed. The EPRI report recommended a phased EMAT which is a state-of-the-art design that generates only a forward propagating wave as a possible solution.

5.3 Future Work

Since this task is consciously designed as a mechanism for responding to unexpected needs and requests from RES, it is not possible to plan or describe the specific work activities that might occur during the next reporting period.

6.0 Piping Inspection Task

This task is designed to address the NDT problems associated with the piping used in light water reactors. The primary thrust of the work has been on wrought and cast stainless steel since these materials are harder to inspect than carbon steel. However, many of the subtasks' results also pertain to carbon steel. The current subtasks are: mini-round robin report, qualification criteria for UT/ISI systems, piping inspection round robin report, cast stainless steel inspection, surface roughness, field pipe characterization, and PISC-III activities.

The work accomplished during this reporting period is summarized in the following paragraphs:

- <u>MRR Report</u> - The Mini-Round Robin (MRR) subtask was conducted to provide an engineering data base for UT/ISI that would help: a) quantify the effect of training and performance demonstration testing that resulted from IEB 83-02, b) quantify the differences in capability between detecting long versus short cracks, and c) quantify the capability of UT/ISI technicians to determine length and depth of intergranular stress corrosion cracks (IGSCC). NUREG/CR-4908 entitled <u>Ultrasonic Inspection Reliability for Intergranular Stress Corrosion Cracks: A Round Robin Study of the Effects of Personnel, Procedures, Equipment, and Crack Characteristics</u> was published in July 1990 to document the work conducted under this subtask.

- <u>Qualification Criteria for UT/ISI Systems</u> - The objective of this subtask was to improve the reliability of UT/ISI through the development of new criteria and requirements for qualifying UT/ISI systems. NUREG/CR-4882 entitled <u>Qualification Process for Ultrasonic Testing on Nuclear Inservice Inspection Applications</u> was published in April 1990.

- <u>Cast Stainless Steel Inspection</u> - The objective of this subtask is to evaluate the effectiveness and reliability of ultrasonic inspection of cast materials within the primary pressure boundary of LWRs. Activities for this work period included macrostructural classification of CCSS material within a PISC block, use of the Rayleigh critical angle technique to characterize CCSS macrostructures, and phase mapping of ultrasonic fields in CCSS.

- <u>Surface Roughness Conditions</u> - The objective of this subtask is to establish specifications such that an effective and reliable ultrasonic inspection is not precluded by the condition of the surface from which the inspection is conducted. Activities included continued refinement of the model by the Center for NDE (CNDE) at Ames Laboratory, and development of better experimental procedures by PNL for obtaining quantitative data for comparing to the model prediction.

- <u>Field Pipe Characterization</u> - The objective of this subtask is to provide pipe weld specimens that can be used for studies to evaluate the effectiveness and reliability of ultrasonic inservice inspection (UT/ISI) performed on BWR piping. The safe-ends that were saved to be given to the PISC-III program have been rejected because of priority and PISC-III funding constraints.

- <u>PISC-III Activities</u>

  This activity involves participation in the PISC-III program to ensure that the work addresses NDE reliability problems for materials and ISI practices on U.S. LWRs. This includes support for the co-leader of Action 4 on Austenitic Steel Tests (AST); providing five safe-ends from the Monticello plant; providing a sector of the Hope Creek reactor pressure vessel containing two recirculation system inlet nozzles; coordination of the inspections to be conducted by U.S. teams on the various actions; input to the studies on reliability and specimens for use in the parametric, capability, and reliability studies of the AST. During this reporting period, tracking and coordination occurred for three assemblies of Action 3, which were inspected by two teams in the U.S. During this reporting period, the efforts continued towards coming up with working solutions to the problem of specimens for the capability studies and working to obtain participation of teams from the USA.

6.1 Mini-Round Robin Report

6.1.1 Introduction

The Mini-Round Robin (MRR) subtask was conducted to provide an engineering data base for UT/ISI that would help:

- quantify the effect of training and performance demonstration testing that resulted from IEB 83-02,
- quantify the differences in capability between detecting long (greater than 3-in.) cracks versus short (less than 2-in.) cracks, and
- quantify the capability of UT/ISI technicians to determine length and depth of intergranular stress corrosion cracks (IGSCC).

[illegible] Status of Work Performed

[illegible] G/CR was prepared to document the work [illegible] under this subtask, and this report was published [illegible] ly 1990. This task is completed and will be [illegible] om future reports.

6.2 Qualification Criteria for UT/ISI Systems

6.2.1 Objective

The objective of this subtask is to improve the reliability of ultrasonic testing/inservice inspection (UT/ISI) through the development of new criteria and requirements for qualifying UT/ISI systems.

6.2.2 Status of Work Performed

Development of criteria and requirements for qualifying UT/ISI systems continued with additional editing of the qualification document (NUREG/CR-4882), which is entitled *Qualification Process for Ultrasonic Testing on Nuclear Inservice Inspection Applications*. This report was published in April 1990.

6.3 Cast Stainless Steel Inspection

6.3.1 Summary

The objective of this task is to evaluate the effectiveness and reliability of ultrasonic inspection of cast materials used within the primary pressure boundary of LWRs. Due to the coarse microstructure of this material, many inspection problems exist and are common to structures such as cladded pipe, inner-surface cladding of pressure vessels, statically cast elbows, statically cast pump bowls, centrifugally cast stainless steel (CCSS) piping, dissimilar metal welds, and weld-overlay-repaired pipe joints. Far-side weld inspection is an inspection technique included in the work scope since the ultrasonic field is passes through weld material.

CCSS piping is used in the primary reactor coolant loop piping of 27 pressurized water reactors (PWRs) manufactured by the Westinghouse Electric Corporation. However, CCSS inspection procedures continue to perform unsatisfactorily due to the coarse macrostructure that characterizes this material. The major macrostructural classifications are a columnar, and equiaxed, and a mixed columnar-equiaxed macrostructure of which the majority of field material is believed to be the latter.

6.3.2 Status of Work Performed

Activities for this work period included macrostructural classification of CCSS material within a PISC (PISC denotes the international Programme for Inspection of Steel Components) block, use of the Rayleigh critical angle technique to characterize CCSS macrostructure, and phase mapping of ultrasonic fields in CCSS.

An opportunity became available to classify CCSS macrostructure contained within the PISC-III Action 3 concerning Nozzles and Dissimilar Metal Welds (NDW) Assembly 25. The block was in route from Southwest Research Institute (SwRI) in San Antonio, Texas to France and a time slot of several days existed when testing could be performed. Personnel and instrumentation were transported to SwRI since it was more economical and timely than shipping the block to PNL. In preparation for the trip, personnel at PNL attempted the techniques defined by Kupperman, et. al (1987) for ultrasonically characterizing cast stainless steel. A letter report of the results was drafted and sent to the NRC program monitor. The preliminary conclusion was that the CCSS material was a columnar macrostructure. Details will be reported after the report has been finalized.

Work began January 1990 to evaluate the Rayleigh critical angle technique as a means of classifying the macrostructure of cast stainless steel. An existing ultrasonic scanning system was modified to permit amplitude and phase measurement. A pitch-catch configuration was employed with the transmitter and receiver inclined at the same angle, the focal spots overlapping at the material surface being insonified and the receiver positioned to receive the wave reflected by the front surface. The selected angle was the critical angle typical of the material being investigated. The premise was that each different grouping of anisotropic grains were characterized by wave velocity and that either amplitude or phase images would correlate with material-property changes.

Preliminary critical angle work regarding the characterization of CCSS was completed. Multi-frequencies were used to analyze anterior layers of different thicknesses. Test results were encouraging in that images displaying texture were acquired from samples having unique macrostructures. C-scan images were made at 1.0, 0.5, and 0.25 MHz for each of the three samples. The lower frequency was set by existing equipment limitations but was considered adequate to assess the technique. The wave velocity of the near-surface material modulated the amplitude of the received signal. Since penetration is inversely proportional to frequency, changes in wave velocity and, hence, macrostructure were detected for layer thicknesses of 3, 6, and 13 mm. The three samples were a pure equiaxed, a pure columnar, and a layered columnar-equiaxed macrostructure. The mixed-mode sample had a columnar outer surface that transformed into an equiaxed layer at a depth of about 10 mm. Detection of this transformation was considered to be a primary goal since it would demonstrate that the technique is capable of detecting changes in macrostructure as a function of depth. The preliminary conclusion was that encouraging results have been obtained; however, insufficient data exists to state that macrostructural classification of the entire pipe wall is possible.

Work began December 1989 on the two-dimensional mapping of phase. The mapping of amplitude has been performed on different macrostructures of cast material, and it was also of interest to extend this capability to include phase. This data was thought to be useful in validating models that are being refined to predict ultrasonic fields in solids, compensating for phase distortion when imaging reflectors, and detecting flaws by detecting phase shifts or interference between the primary wave front and a flaw. A modified Holosonics system was used to collect data on CCSS material of pure equiaxed and pure columnar macrostructure. A paper was drafted and submitted to the NRC for approval prior to submission to the proceedings of the annual conference "Review of Progress in Quantitative Nondestructive Evaluation."

The preliminary conclusion of phase mapping in CCSS was that the fringe pattern for longitudinal waves at 1 and 2 MHz displayed significantly less distortion than was expected and also less than in the amplitude mapping. This reduced sensitivity to material macrostructure may facilitate the use of phase data for flaw detection and enhance imaging by compensating for phase distortion.

6.3.3 Future Work

CCSS work will focus on collecting the pertinent information concerning CCSS and presenting this at a workshop for NRC personnel. This will include the CCSSRRT, selective frequency filtering of ultrasonic signals for CCSS macrostructures, ultrasonic field distortion, and ultrasonic attenuation. Additional critical angle work will be performed to determine how this technique might be implemented in the field. Specifically this would consist of implementing much lower frequencies (e.g., 30 to 250 kHz) and classifying each layer as penetration is increased. The low frequency is required so that penetration to the inner diametrical surface is accomplished. Prior to performing experimental work, an analysis will be performed to assure that successive layers in depth can be classified. CCSS work will also continue to document macrostructures, acquire ultrasonic attenuation measurements from the respective macrostructures, and acquire ultrasonic field maps from complex material macrostructures. However these later efforts will be done at a lower priority as funds are available.

Far-side inspection and dissimilar metal welds work will include sample acquisition and metallography, and the acquisition of ultrasonic field maps to document field distortion.

6.4 Surface Roughness Conditions

6.4.1 Summary

The objective for this work was to establish specifications such that an effective and reliable ultrasonic inspection is not prevented by the condition of the inspection surface. Past efforts included an attempt to quantify the effect produced by an outer surface irregularity. This approach was then redefined to cooperate with EPRI in establishing a mathematical model to be used as an engineering tool for deriving guidelines for surface specifications. Under the auspices of the cooperative agreement, the Center for NDE (CNDE) at Ames Laboratory with EPRI funding was assigned that task of refining an existing, isotropic model; and PNL with NRC funding was assigned the task of acquiring experimental data to support model refinement and validating the model.

6.4.2 Status of Work Performed

Activities for the past work period included continued refinement of the model by CNDE, and development of better experimental procedures by PNL for obtaining quantitative data for comparison with the model predictions. Delays occurred in model refinement due to funding constraints. PNL also received a normal incident, shear-wave, electromagnetic-acoustic transducer (EMAT) for measuring shear-wave directivity of microprobes.

The EMAT was designed for use on the curved surface of a cylinder. Directivity was measured by placing a microprobe on the planar surface of a half-cylinder of steel and receiving the shear wave transmitted by the EMAT. Thus, directivity was determined by the amplitude response of the microprobe as the EMAT was moved from -90° to 90°. The amplitude, however, was dependent on the rotational direction of the EMAT. A possible explanation was that the magnets of the EMAT leave a residual field in the steel and that this affected the succeeding measurement.

6.4.3 Future Work

The schedule of model refinement, delivery of the model to PNL, model validation by PNL, and use of the model to determine code recommendations for surface condition is being redefined. If a reasonable schedule cannot be established, then a simpler two-dimensional model already at PNL might be used instead of the three dimensional model that would have been received from CNDE.

6.5 Field Pipe Characterization

6.5.1 Introduction

The objective of this subtask is to provide pipe weld specimens that can be used to help determine the effectiveness and reliability of ultrasonic inservice inspection (UT/ISI) that is performed on BWR piping. This goal will be accomplished by supporting PNL laboratory studies and providing specimens that will be used in other work such as PISC III.

6.5.2 Status of Work Performed

The focus of the work this year has been in the preparation of the documentation for the shipment of the 5 safe-ends that had been removed from the Monticello Nuclear Power Station. After a long and drawn out review of data and the unpackaging of the specimen for making detailed drawings, taking of photographs and the radiation survey of one of the safe-ends, the complete package was submitted to PISC III program. After a review process by PISC III representatives, it was decided that the priorities and the funding for the evaluation of removed-from-service components would no longer be supported by the PISC III program. As a result, these specimen will have to be buried. The new rules and regulations from the Environmental Protection Agency make this process more difficult and confusing because the rules are not clearly stated and there seems to be a wide range of interpretations of how to meet the requirements.

A complete package of information on the specimen that will be used in the wrought stainless steel reliability test was assembled and provided to the staff at the Joint Research Centre in Ispra, Italy. The stands to be used for supporting the specimen during inspection were fabricated and tested to insure that they would work properly.

6.5.3 Future Work

Proceed with determining what must be done to meet requirements in order to bury the five safe-ends.

Prepare the specimen for the wrought stainless steel reliability test for shipment to the Joint Research Centre in Ispra, Italy.

6.6 PISC-III Activities

6.6.1 Introduction

The objective of this subtask is to contribute to the international Programme for the Inspection of Steel Components III (PISC III) to facilitate current studies on the reliability, capability, and parametric analysis of NDE techniques, procedures, and applications. This includes full-scale vessel testing; piping inspections; and human reliability, real components, nozzles and dissimilar metal welds, and modeling studies on ultrasonic interactions. These data will be used in quantifying the inspection reliability of ultrasonic procedures and the sources and extent of errors impacting reliability.

6.6.2 Status of Work Performed

The primary areas in which PNL participated include Action No. 1 on Real Contaminated Structures Tests (RCS), Action No. 2 on Full-Scale Vessel Tests (FSV), Action No. 3 on Nozzles and Dissimilar Metals Welds (NDM), Action No. 4 on Round-Robin Tests on Austenitic Steels (AST), Action No. 6 on Ultrasonic Testing Modeling (MOD), and Action No. 7 on Human Reliability Exercises (REL). These actions are being followed to ensure that conditions, materials, and practices in the U.S. are being included in the work so that the results are transferable to the U.S.

As was noted in the CCSS work, PNL performed some inspections on the Action 3 Assembly number 25 to see if it is possible to determine the macrostructure of the material through use of acoustic measurements in a blind test. This work was performed and reported to the NRC and to the PISC III program. This work will be reported in the next semi-annual report.

This task supported 2 staff from PNL to attend Management Board meetings during the course of the year. In addition, this task supported the technical exchange needed to aid in guiding the implementation of the AST program. The key here was the acceptability of the design for the cast-to-cast specimen set. The final design is a full pipe that will be made from welded segments that will permit inexpensive flaws to be introduced into the segments. The materials that were obtained from the European source provided a good range of macrostructures for use in this study. It is planned that the assembly for this study will be ready for circulation by the middle of the fiscal year of 1992 and also for the assembly for the cast-to-wrought study.

6.6.3 Future Work

This will involve supporting and coordinating teams from the USA to participate in the AST studies, supporting data analysis, in the report writing, and attendance at action and management board meetings.

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The Evaluation and Improvement of NDE Reliability for Inservice Inspection of Light Water Reactors (NDE Reliability) Program at the Pacific Northwest Laboratory was established by the Nuclear Regulatory Commission to determine the reliability of current inservice inspection (ISI) techniques and to develop recommendations that will ensure a suitably high inspection reliability. The objectives of this program include determining the reliability of ISI performed on the primary systems of commercial light-water reactors (LWRs); using probabilistic fracture mechanics analysis to determine the impact of NDE unreliability on system safety; and evaluating reliability improvements that can be achieved with improved and advanced technology. A final objective is to formulate recommended revisions to ASME Code and Regulatory requirements, based on material properties, service conditions, and NDE uncertainties. The program scope is limited to ISI of the primary systems including the piping, vessel, and other components inspected in accordance with Section XI of the ASME Code. This is a progress report covering the programmatic work from October 1989 through September 1990.

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