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

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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 April 1991 through September 1991.

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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. 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). Minutes of the ASME Section XI Subgroup on Nondestructive Examination (SGNDE) meetings held February 1991 in San Diego, CA, May 1991 in Orlando, FL, and August 1991 in Pittsburgh, PA were prepared and distributed. Meetings of the Section V Subgroups on Ultrasonic Testing (SGUT) and on Acoustic Emission (SGAE) were also attended during ASME Section V meetings held in May 1991 and September 1991. The previous Ad Hoc Industry Task Group on Appendix VIII Implementation has been renamed "Performance Demonstration Implementation (PDI)." A new Supplement 12 for Appendix I "Flaw Sizing Requirements" was approved for publication in the 1991 Addenda. Significantly, the PNL-prepared new Paragraph T-435 (plus appendices) on Computerized UT Imaging Techniques was approved by the SGUT for submittal to Section V. The activity to develop performance demonstration requirements for eddy current data analysts is continuing.

- <u>Pressure Vessel Activities</u>

  <u>Re-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 and these are being addressed prior to submitting this report to the NRC for publication.

  <u>RPV Research</u>. The objective of this subtask is to track work currently being performed under the PISC-III program; particularly work conducted under Actions 2 and 3. These actions provide useful information concerning the capability to inspect nozzles and dissimilar metal welds and will begin to address techniques for accurate flaw sizing as well as the reliability of vessel inspections. At this time, no results are available but these

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

studies are expected to provide useful data bases and conclusions in the near future. Initial results will be made available to the PISC-III Management Board in late 1991. In addition, a new activity was started to evaluate the status of the technology that the industry is developing for the inspection of the boiling water reactor (BWR) reactor pressure vessels. During this reporting period PNL staff attended a workshop on this topic that was conducted by EPRI which brought together all companies involved with the developed of this technology.

License Renewal. The ASME Section XI rules and procedures for flaw evaluation were examined under this subtask. The Section XI procedures classify flaws as linear, laminar, or planar, and equivalent flaw sizes are determined in accordance with Subsection IWA. The pass/fail criteria for flaws are found in Subsection IWB-3500. For flaws that fail the simple criteria under IWB-3500, a more detailed fracture mechanics evaluation may be conducted in accordance with Appendix A to Section XI. Appendix A utilizes stress field data at the flaw location, measured flaw size, and estimated material properties to evaluate flaw acceptability.

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. A computer model is being used to calculate flaw transfer functions and frequency domain effects that occur due to the interaction between inspection equipment and worst-case flaws. Calculated values are also being compared with experimental measurements to determine if other measurement-related effects, not programmed into the model, are significant enough for inclusion in the model.

During this reporting period, work was performed to model equipment interactions with worst-case flaws under curved surfaces, similar to the way equipment interactions with worst-case flaws under flat surfaces was previously modeled. Related activities included a) ray tracing analysis combined with other calculations showed the different effects a model must account for when the surface to be inspected is curved, and b) appropriate computer hardware and software was installed to manage the increased complexity associated with extending the model to predict maximum allowable equipment tolerances.

- 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 high nuclear power plant component structural integrity. 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. During this reporting period the ASME Task Force accomplished its first major milestone with publication of the ASME report "Risk-Based Inspection - Development of Guidelines - Volume 1 General Document" (ASME, 1991). A second document specifically directed to light water reactor nuclear power plant components is completed in draft form, and is undergoing revisions suggested by peer reviewers.

Calculations have applied probabilistic risk assessment (PRA) to establish inspection priorities for pressure boundary systems and components. Plant-specific risk-based studies have been conducted for the Surry Unit 1 Nuclear Power Station, with the cooperation of Virginia Electric Power Company. Estimates of failure probabilities from an expert judgement elicitation performed in May of 1990 have been used to rank inspection priorities for components in four critical systems (reactor pressure vessel, reactor coolant system, low pressure injection system, and auxiliary feedwater system). A final ranking of components in these four systems was completed during this reporting period.

- 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 Regulatory Research (RES). Efforts continued to support NRC staff in the application, validation, and modification of the VISA-II fracture mechanics code for predicting the contributions of undetected flaws to the failure of reactor pressure vessels under conditions of pressurized thermal shock accidents. Recent work to validate the VISA-II code revealed no significant errors, although some corrections were made to enhance the overall utility of this code. New features include modeling of residual stresses, modeling of underclad cracks, programming the chemistry factor, and improved plastic instability predictions. An updated version of the VISA-II code with documentation of all corrections and enhancements was prepared and made available to the NRC staff.

During this period another request was made by RES to provide support in the development of a technical position regarding industries interest in digitizing radiographs. PNL provided support in the development of performance demonstration requirements for personnel that would be involved in this process.

- 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: cast stainless steel inspection, surface roughness conditions, field pipe characterization, and PISC-III activities.

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 a workshop conducted for NRC Regional personnel on ultrasonic testing/inservice inspection of cast material and preparation of a paper documenting an imaging technique to characterize localized changes in microstructure. Input for a document describing a UKAEA/NRC cooperative effort on cast materials was drafted, a strategy change to redirect this task to employ techniques inherently insensitive to microstructural changes rather than specific classifications of microstructures was prepared for the effort for the next fiscal year, and Dr. J. L. Rose at Drexel University was visited.

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. Significant progress was achieved by transferring the first version of the CNDE model to PNL and comparing model predictions with experimental data. Also, the responsibility for model development at CNDE was transferred from Mr. Ali Minachi to Dr. Isaac Yalda. R. B. Thompson continues in his role of supervising the model refinement work at CNDE at the Iowa State University.

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. Documentation of the five safe-ends removed from the Monticello nuclear power station are no longer needed for any programmatic work. Processes to dispose of these samples are under review to meet the new requirements for disposal required by state and federal agencies.

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 major efforts focused on trying to get participation of teams from the USA in the capability studies.

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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.

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 and V activities continued toward achieving Code acceptance of NRC-funded PNL research results to improve the reliability of nondestructive examination/inservice inspection (NDE/ISI). Minutes of the ASME Section XI Subgroup on Nondestructive Examination (SGNDE) meetings held February 1991 in San Diego, CA, May 1991 in Orlando, FL, and August 1991 in Pittsburgh, PA were prepared and distributed. Meetings of the Section V Subgroups on Ultrasonic Testing (SGUT) and on Acoustic Emission (SGAE) were also attended during ASME Section V meetings held in May 1991 and September 1991. The previous Ad Hoc Industry Task Group on Appendix VIII Implementation has been renamed "Performance Demonstration Implementation (PDI)." A new Supplement 12 for Appendix I "Flaw Sizing Requirements" was approved for publication in the 1991 Addenda. Significantly, the PNL-prepared new Paragraph T-435 (plus appendices) on Computerized UT Imaging Techniques was approved by the SGUT for submittal to Section V. The activity to develop performance demonstration requirements for eddy current data analysts is continuing.

2.2 Introduction

One objective of this task is to develop and/or evaluate new criteria and requirements for achieving acceptable and reliable ultrasonic testing/inservice inspection (UT/ISI). 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 (ASME 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 Regulatory Guide input as a back-up approach.

To implement this goal, PNL staff members are active participants in Section XI and Section V ASME Code committee activities. PNL staff members serve as secretary of the SC-XI SGNDE, chair a special SC-XI task group to develop acoustic emission requirements, and serve as members of the SC-XI Working Group on Surface Examination and Personnel Qualification (WG-S) and the SC-XI Working Group on Volumetric Examination and Procedure Qualification (WG-V). In addition, PNL personnel serve as members of the SC-V Subgroup on Ultrasonic Testing (SGUT) and the SC-V Subgroup on Acoustic Emission (SGAE). Administrative assistance is also provided in support of related efforts to achieve ASME Code acceptance of new personnel qualification (PQ) and performance demonstration (PD) requirements for eddy current equipment operators and data analysts, adoption of national standard requirements for personnel qualification, and a new SC-V activity to develop requirements and criteria for computerized UT imaging techniques.

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 results to improve the reliability of NDE/ISI. During this reporting period, meetings of the ASME Section XI Subcommittee (including relevant Subgroup and Working Group meetings) were attended May 20-23, 1991 in Orlando, FL and August 26-29, 1991 in Pittsburgh, PA. In addition, meetings of the Section V Subcommittee on Nondestructive Examination (specifically the Subgroup on Ultrasonic Testing, the Subgroup on Acoustic Emission, and the Subgroup on General Requirements/Surface Examination) were attended May 20-23, 1991 in Orlando, FL and September 9-11, 1991 in New York City, NY.

Agendas and minutes of the SGNDE meetings held in conjunction with the Section XI Subcommittee meetings were prepared and distributed. One PNL representative serves as Secretary of the SC-XI SGNDE, as a member of the Working Group on Surface Examination and Personnel Qualification, and as a member of the SC-V SGUT and SGAE. Another PNL representative chairs a special task group to develop acoustic emission requirements and serves as a member of the Working Group on Volumetric Examination and Procedure Qualification.

The SGNDE and SC-XI sequentially approved a proposed action to delete Appendix VI (UT Examination of Bolting) from Section XI. Appendix VI is no longer needed since the recently available Appendix VIII adequately covers performance demonstrations for UT of bolting.

A detailed review of the new ASNT PQ Standard continued at the working group level toward possible adoption by Section XI as Code rules. This action complements a recommendation that the ASME Code stay with the 1984 Edition of SNT-TC-1A, at least for the time being, rather than adopt the 1988 Edition. This recommendation was made by a Main Committee task group that was charged to consider various NDE personnel qualification proposals, and is consistent with the position taken by the PNL representatives over the past three years.

During a joint SC-V SGUT and SC-XI WG-V meeting, a PNL restructured draft of T-435 entitled "Proposed Revisions to ASME Section V, Article 4 on Computerized UT Imaging Systems" was adopted at the Working Group level. Subsequently, the T-435 document was "decommercialized" and condensed for submittal to the SGUT. During the SC-V meetings in September 1991, the SGUT unanimously approved the PNL-prepared T-435 document for subsequent placement on the Section V agenda for the November 1991 meeting.

The previous Ad Hoc Industry Task Group on Appendix VIII Implementation has been renamed "Performance Demonstration Implementation (PDI)." PDI is coordinating the development and acquisition of performance demonstration specimens for reactor piping, pressure vessels, and bolting. It was reported that 96 of 112 nuclear power stations are now participating in the Appendix VIII implementation activity.

During the May SC-XI meeting, both the SGNDE Chairman and the SC-XI Chairman stated that they expected the Special Working Group on Eddy Current Testing (SWGET) to develop performance demonstration criteria and requirements that are similar in approach and philosophy to the UT requirements and criteria that were developed and published in Appendix VIII. In response to this direction, a revised scope of the new ET document was disseminated. During the August SC-XI meeting, the SWGET reached a non-unanimous consensus on a mandatory scope for this appendix. Discussion focused on a proposed sampling plan, specimen set criteria, the consequences of different pass/fail criteria for the ET and UT methods, and other open issues.

A new Supplement 12 for Appendix I "Flaw Sizing Requirements" was approved for publication in the 1991 Addenda, and the ASME Code rules for limited certification of NDE personnel were clarified via an implementing Code Case that was sequentially approved by the SGNDE and SC-XI. Alternative visual acuity requirements for NDE personnel were approved for publication in the 1991 Addenda.

2.4 Future Work

Minutes for the ASME Section XI SGNDE meeting held August 1991 in Pittsburgh, PA will be finalized for distribution to the approximately 65 recipients. Administrative support will continue for the activity to develop qualification requirements for the eddy current personnel, equipment, and procedures used for ET/ISI of steam generator tubes.

The proposed new Section V requirements for computerized UT imaging systems will be further refined and submitted for Section V approval. Comments arising from the Section V review will be accommodated, and this document will be submitted for Main Committee consideration.

In our role as SGNDE Secretary, PNL prepares and distributes the agendas and minutes for all SGNDE meetings. These are drafted immediately following each meeting, and are then finalized and distributed to the approximately 65 recipients on the SGNDE mailing list 4-6 weeks prior to each Section XI meeting. Future Section XI meetings will be held November 1991 in Anaheim, CA, February 1992 in Atlanta, GA, and May 1992 in San Antonio, TX. Future Section V meetings will be held February 1992 in New York City, NY and May 1992 in San Antonio, TX in conjunction with the other Code committees.

3.0 Pressure Vessel Inspection

3.1 Pressure Vessel Inspection

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. Additional comments have been received from the draft report on the re-analysis of the PISC-II data, 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 effort in this task is a re-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 is intended 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. This subtask will also 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. In addition, a new activity was started to evaluate the status of the technology that the industry is developing for the inspection of the boiling water reactor (BWR) reactor pressure vessels. During this reporting period PNL staff attended a workshop on this topic that was conducted by EPRI which brought together all companies involved with the developed of this technology.

- **License Renewal** - The License Renewal Task was initiated in April 1991. The objective of this task is to determine inservice inspection (ISI) requirements for license renewal. Initial efforts have focussed on reviewing the technical bases for existing reactor pressure vessel (RPV) flaw acceptance criteria. The RPV was selected for initial study since it was judged to be the most important component unlikely to be replaced and is the one component most susceptible to radiation damage.

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 semi-annual report NUREG/CR-4469, Vol. 11. No work was performed during this time period, although we have now received all NRC comments on the draft report.

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. At this time, there are no results available; although these studies should provide some useful data bases and conclusions in the near future. The initial results from these studies will begin to be made available to the PISC III Management Board in late 1991.

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. At this time, this subtask is focused on tracking the relevant PISC III program activities and the results being developed.

During this reporting period two activities occurred that addressed the issue of the problem of the inspection of BWR reactor pressure vessels. These RPVs were not designed for inspectability and consequently, they represent a significant challenge from an access standpoint. In addition, the condition of the surfaces and particularly the inside surface is not well know for most of these RPVs. On the inside there is the upper core shroud that inspection devices must be maneuvered around and then there is the constraint of the jet pumps and the limited access between and behind these pumps that must be overcome in order to be able to inspect the needed volumes of RPV material.

The first activity was witnessing a vessel examination that was being conducted at the Monticello nuclear power plant. The equipment was quite well designed and was based on designs that had been developed for other inspections. Some complications and problems were encountered during the examination, resulting in fewer inspections being performed than had been planned. The significance of the inspection was that there appeared to be no major problem from the standpoint of being able to get a transducer and scanning head into the inspection zone. The only problem that appeared was dealing with knowing where the transducer is at all times and confirming that things such as rough surfaces do not cause the transducer to hang up and move in jumps rather than a smooth step-like fashion.

The second activity was attendance at a workshop entitled "BWR Reactor Pressure Vessel & Internals - Inspection and Repair Workshop" conducted by EPRI from July 16 through July 18, 1991. This was a good workshop and it showcased technology that had been developed for other applications that appears to be modifiable for application to the BWR RPV problem. The technology appears to have the necessary performance but now it needs to be proven and shown to be reliable for this application.

3.1.3.3 License Renewal

Work under this subtask examined the rules and procedures for flaw evaluation presented in ASME Section XI, Division I for the welds in RPVs. The most relevant portions of the Code are discussed below.

Subsection IWA describes methods for determining if flaws of various sizes and shapes are linear, laminar, or planar. Furthermore, equivalent sizes (length for linear flaws, and ellipse dimensions for planar and laminar flaws) are determined in accordance with this subsection. A pass/fail evaluation of these flaws is then conducted, starting with Subsection IWB-3500. If a flaw fails to pass under IWB-3500, a more detailed fracture mechanics evaluation (Appendix A of Section XI) may be utilized to see if the flaw is acceptable. Thus, IWB-3500 might be considered as an initial screening process to accept those flaws with relatively large safety margins, thus avoiding the effort of a more detailed evaluation.

Planar flaws are evaluated with the use of the flaw acceptance standards of Table IWB-3510-1. This table gives allowable equivalent flaw sizes as a function of ellipse aspect ratio and vessel wall thickness. The surface "reference flaw" size was selected to be one tenth as large as the so called maximum postulated defect ($a/l=1/6$, $a=t/4$ when a = through-wall extent, l = length, and t = wall thickness) of Appendix G, Section III. This gives a suitable safety factor for stress. Sizes of other elliptical surface flaws (of different aspect ratios) were then computed so that they would have the same safety factor as the surface reference flaw.

The tables in IWB-3500 for flaw evaluation do not directly involve the use of the stress level at the flaw location. Flaw sizes allow for adequate safety margins if stress levels are at the upper limits of Section III allowables. In addition, it is stated that the tables may be used only if minimum fracture toughness can be assured. This is mandated in several places in IWB-3500. The first is Subarticle IWB-3410-1 which states that the material must conform to the requirements of Section III, Division I Appendices with respect to selection of a maximum postulated defect. In addition, the flaw size tables of IWB-3500 state that the material must meet the fracture toughness requirements of NB-2331 and G-2110(b), Section III. Subarticle NB-2331 contains procedures for determining the reference nil ductility temperature (RT_{NDT}). This involves performing Charpy V-notch tests and obtaining results of at least 50 ft-lbs and at least 0.035 in. of lateral expansion.

If a detected flaw does not pass the acceptance standards tables of IWB-3500, then acceptability can still be achieved by performing detailed fracture mechanics analyses as described in Appendix A of Section XI. Appendix A utilizes stress field data at the flaw location, measured flaw size, and estimated material properties (including irradiation effects) and computes the stress intensity factor of the flaw in question, the critical flaw size, and the material fracture toughness. The RT_{NDT} of Section III, NB-2231 is used for the fracture toughness evaluation. Consideration is given to both normal and faulted loading conditions. For normal loading conditions, the safety factor is 3 for stress and 10 for flaw size -- approximately the same values used for the IWB-3500 tables if the stress level is at the maximum allowable for Section III.

3.1.4 Future Work

- **PISC II Re-analysis** - NRC comments received on the June 1990 draft are being accommodated, and a final version of this report will be submitted to the NRC for publication.
- **RPV Research** - The work planned for the next reporting period is to track the PISC III work and to write a status report on the BWR RPV inspectability review that is underway.
- **License Renewal** - An evaluation will be conducted to determine if the flaw acceptance criteria, summarized above, adequately consider the impact of degraded material properties. The goal is to determine if adequate NDE reliability exists for flaw detection and sizing after considering the effect of material property degradation on the locations, types, and sizes of flaws that may be of concern from a license renewal perspective.

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 equipment operating parameters for improving ultrasonic inspection reliability. A computer model is being used to calculate flaw transfer functions and frequency domain effects that occur due to the interaction between inspection equipment and worst-case flaws. Calculated values are also being compared with experimental measurements to determine if other measurement related effects, not programmed into the model, are significant enough for inclusion in the model.

During this reporting period, work was performed to model equipment interactions with worst-case flaws under curved surfaces, similar to the way equipment interactions with worst-case flaws under flat surfaces were previously modeled. Related activities included:

- Ray tracing analysis combined with other calculations showed the different effects that a model must account for when the surface to be inspected takes on a finite radius of curvature.
- Appropriate computer hardware and software was installed to manage the increased complexity associated with extending the UT equipment interaction model to predict maximum allowable equipment tolerances, for reliably detecting a worst-case flaw beneath a curved surface.

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 Section XI Code, Appendix VIII. The current specifications are based on engineering judgment and lack a solid analytical foundation. The Interaction Matrix Study will provide this foundation. Both thin sections (piping) and thick sections (pressure vessels) are being evaluated.

The following work was completed during previous reporting periods:

- Mathematical models were developed for UT/ISI equipment.
- 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 interactions.

- 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 ASME Code center frequency tolerance of ±10% is not adequate to ensure reliable inspection of certain calculated worst-case flaws with narrow band UT/ISI systems.

- The first draft of the topical NUREG/CR report entitled *The Interaction Matrix Study: Models and Equipment Sensitivity Studies for the Ultrasonic Inspection of Thin Wall Steel Piping* was reviewed by PNL staff. 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.

- 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. The ASME Code requirements for equipment center frequency were again found inadequate, just as for thin sections.

- Results from an experiment performed, in a 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 are consistent with the model, but are not as sensitive to center frequency changes as predicted by the model.

3.2.3 Status of Work Performed

The following work was completed during this reporting period:

- A ray tracing analysis was combined with other calculations to provide an overview of the different effects that a model must account for when the surface to be inspected takes on a finite radius of curvature.

- Computer hardware and software were upgraded to manage the increased complexity associated with extending the UT equipment interaction model to predict maximum allowable equipment tolerances, for reliably detecting a worst-case flaw beneath a curved surface.

Calculations that provide insight into the inspection modeling required for addressing curvature effects were investigated. These calculations showed that inspection from the O.D. will result in an altered echodynamic curve. This curve will be greatly shortened because the ray tracing shows that the scattered rays will not strike at the planned 45°, but will occur at different angles and lead to energy going toward areas other than where the transmitting transducer may be located. This form of modeling was found useful in obtaining rough overviews of the type of interactions that occur when inspecting surfaces with finite radii of curvature. These calculations also pointed out that there should not be a deterioration in a corner response, so detection performance should remain unaltered. More detailed studies are planned to determine whether this first approximation is accurately reflecting the effects of curvature.

The Applied Physics Center provided (at no cost to the NRC) a SPARK station 2, Sun workstation to the NDE group for use in ultrasonic modeling studies. The Sun workstation was connected to Ethernet for communication with other computers, printers, and data transfer. Software used in performing modeling studies on UNIX-based workstations was installed and tested on the Sun. This new workstation was found to compile and execute programs 20 to 30 times faster than previously found on the VAX-11/780. Speed improvements of this magnitude will become essential as larger programs associated with acoustically modeling curved surface effects are developed. There were also cost savings in moving the model from the VAX to the Sun, as the VAX aged, maintenance costs started to reach unacceptable levels.

3.2.4 Future Work

The following work remains to be completed:

- Test computer code developed on the VAX on the Sun and verify results.
- Expand the flaw model to include curved sections (reactor pressure vessel shell and the nozzles) and perform experiments to validate these calculations.
- Report findings and recommendations from the curved section study to ASME Code Section XI.
- Extrapolate results from curved surface study to all important surface geometries.
- Write a topical report documenting all work on thick sections (flat and curved).

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 assure high nuclear power plant component structural integrity. 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. During this reporting period the ASME Task Force accomplished its first major milestone with publication of the ASME report "Risk-Based Inspection - Development of Guidelines - Volume 1 General Document" (ASME, 1991). A second document specifically directed to light water reactor nuclear power plant components is completed in draft form, and is undergoing revisions suggested by peer reviewers.

Calculations have applied probabilistic risk assessment (PRA) to establish inspection priorities for pressure boundary systems and components. Plant-specific risk-based studies have been conducted for the Surry Unit 1 Nuclear Power Station, with the cooperation of Virginia Electric Power Company. Estimates of failure probabilities from an expert judgement elicitation performed in May of 1990 have been used to rank inspection priorities for components in four critical systems (reactor pressure vessel, reactor coolant system, low pressure injection system, and auxiliary feedwater system). A final ranking of components in these four systems was completed during this reporting period.

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 the 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 provided generic insights that could be extrapolated from the eight plants to specific classes of light water reactors. With a few exceptions, the PRA-based priorities for inspection of systems generally correlated with current ASME Section XI requirements for Class 1, 2, and 3 systems.

In FY90 and FY91 our work has addressed inspection priorities at a more detailed component level, and focused on plant-specific calculations for the Surry Unit 1 Nuclear Power Station. During this reporting period we have continued with the Surry-1 studies. Evaluations for the initial four systems previously selected for detailed study were completed, and final results for these systems are described below. Work continues on additional Surry-1 systems, and future results will give a complete picture of the most risk significant components within the plant.

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 PNL participation on a special ASME Research Task Force on Risk-Based Inspection Guidelines. This ASME Task Force has been identified as an effective route to achieve long-range goals for improved inspection criteria. The ASME Task Force has had a strong focus on nuclear power applications, with the goal of developing recommendations to ASME Section XI on the use of risk-based methods to develop improved inspection programs.

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

- May 14-16, 1991 at Clearwater, Florida
- August 20-22, 1991 at Pittsburgh, Pennsylvania

In addition there was an informal meeting on June 25, 1991 of key members of the task force at the ASME Pressure Vessel and Piping Conference. This working session was to discuss plans for probabilistic fracture mechanics calculations to support future development of inservice inspection programs. During this session O.J.V. Chapman (Rolls-Royce and Associates Limited, UK), B.A. Bishop (Westinghouse Electric), and D.O. Harris (Failure Analysis Associates) demonstrated computer software developed by their respective organizations, and also described examples of how ISI effectiveness has been included in their structural reliability calculations.

The Phase I work of the ASME Task Force has produced a general document (ASME 1991) that recommends and describes appropriate methods for establishing inspection guidelines using risk-based approaches for any facility or structural system. This document was subject to extensive independent peer review prior to publication, and was published by ASME during June of 1991 and is planned to be published by the NRC in FY92 as NUREG/GR-0005, Vol. 1.

Work on a second document (Volume 2 - Part 1) on the special topic of nuclear power applications continued during this reporting period. This document recommends and describes specific methods to be used in developing risk-based inspection plans for nuclear power facilities. Incorporation of recommendations from independent peer reviewers and final editing of the report is in progress. Publication by ASME is expected during 1992.

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 which is scheduled for publication in the later part of 1993. The document will make specific recommendations for consideration by ASME Section XI.

4.3.2 Plant Specific PRA Application to Surry-1

A major part of the new inspection criteria task involves the application of existing probabilistic risk assessments (PRA) to establish inspection priorities for pressure boundary systems and components. During this reporting period a pilot application of PRA methods to the Surry-1 plant continued, including completion of risk-based rankings for components in four of the most safety significant systems.

The Surry-1 work has 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 and components at nuclear power plants. The specific systems initially selected for analysis were the reactor pressure vessel, the reactor coolant, the low pressure injection (including the accumulators), and the auxiliary feedwater. Efforts during this reporting period have been expanded to address other systems including the high pressure injection, main feedwater, service water, component cooling water, main steam, condensate, and residual heat removal systems.

Core damage frequency (Level-I PRA) has been used in this study as the bottom line risk measure. FMEA results are applied to calculate the relative importance of each component within the systems being addressed. The calculated importance measures reflect the expected consequences of failure of the component (from the Surry-1 PRA) and the expected probability of failure (rupture) of the component. Estimates of rupture probabilities for the Surry-1 components have been obtained from an expert judgement elicitation.

Staff from the Virginia Electric Power Company (VEPCO) have been actively participating in the pilot study. This participation is important to assure that the plant models are as realistic as possible and reflect plant operational practices. Data from prior visits to the Surry-1 plant site were incorporated into the FMEA calculations during this reporting period, and telephone contacts continued with plant operational technical staff to further resolve details of plant design and operation.

Table 4.1 summarizes the risk-based rankings of components within the four selected systems at the Surry-1 plant as established during the current reporting period. These rankings are based on contributions of component ruptures to core damage frequency. The rupture frequencies as listed in Table 4.1 were obtained from the 1990 expert elicitation workshop whereas the conditional core damage frequencies (given rupture) were based on information extracted from the Surry-1 PRA. Figure 4.1 provides a plot of the core damage contributions with uncertainties in the estimates of rupture frequencies indicated by upper- and lower-bound values.

The contributions of individual component failures range widely from 1.6E-06 to 1.6E-14 per plant year. The cumulative risk contribution for all the listed components is about 2.1E-06 per plant year, which corresponds to an acceptably low fraction (about 5%) of the total Surry-1 PRA risk. The total risk from component ruptures is dominated by failures of the reactor pressure vessel (86%). This risk is followed by the low-pressure injection system components (10%) and then other components within the auxiliary feedwater and reactor coolant system (4%).

Sensitivity analyses were performed to address the contributions to core damage frequency due to indirect effects of component failures (e.g., contributions by failures due to pipe whip, jet impingement effects, failures of vital electrical buses, etc.). The overall contributions of indirect effects was found to be negligible (less than 2%).

In future work the results of Table 4.1 will be expanded in scope to cover components of all other significant pressure boundary systems at the Surry-1 plant. This will result in a comprehensive ranking of all the most important components within the Surry-1 plant systems that should be addressed when assigning priorities for inservice inspection. These risk-based priorities can then be compared with current inservice inspection requirements as specified by Section XI of the ASME 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 fracture mechanics analyses are available, the expert judgment elicitation process was selected as a method to estimate needed rupture probabilities.

PNL conducted the first of two expert judgment elicitation meetings on May 8-10, 1990 at Rockville, Maryland to address the issue of failure probabilities for Surry-1 components. The systems addressed by the first 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). During the previous reporting period the results of the elicitation were compiled to determined a "best" estimate of rupture probability for every component listed in the risk-based rankings shown in Figure 4.1 of this report.

During this reporting period we started to prepare for a second expert elicitation workshop scheduled for February 1992. This future workshop will again address Surry-1 components and will include the high pressure injection system, the residual heat removal system, the component cooling water system, the service water system, and the power conversion system.

4.3.4 Structural Reliability Calculations

The long range objective of the new criteria task is to develop improved inservice inspection plans (what, where, when, and by what method) using risk-based approaches. Following the guidelines given in the Volume 1 document issued by the ASME Research Task Force on Risk-Based Inspection, probabilistic fracture mechanics and decision risk analysis methods will be used to identify inspection strategies that meet criteria for both safety and cost effectiveness.

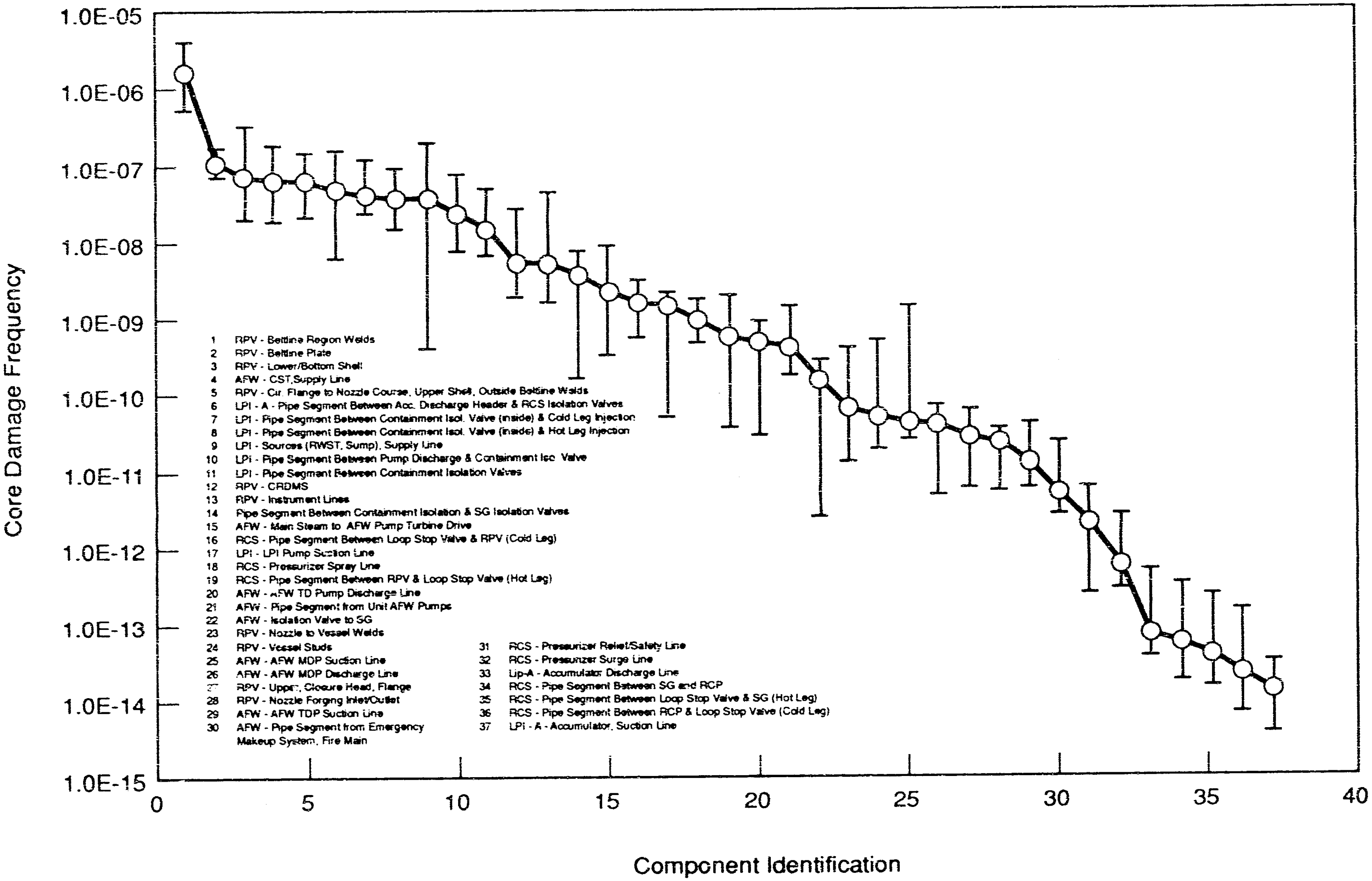

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Figure 4.1. Risk Contributions for Surry Components

Table 4.1. Risk Importance Parameters for Components at Selected Systems at Surry-1[(a)]

System[(b)]	Component[(c)]	Rank	Conditional Core Damage Frequency Given Rupture	Rupture Frequency	Core Damage Frequency
RPV	Beltline Region Welds	1	1.0	1.58E-06	1.58E-06
RPV	Beltline Plate	2	1.0	1.00E-07	1.00E-07
RPV	Lower/Bottom Shell	3	1.0	7.32E-08	7.32E-08
AFW	CST, Supply Line	4	1.7E-02	4.03E-06	6.86E-08
RPV	Cir. Flange to Nozzle Course Upper Shell, Outside Beltline Welds	5	1.0	6.16E-08	6.16E-08
LPI-A	Pipe Segment Between Acc. Discharge Header and RCS Isolation Valves	6	1.8E-02	2.59E-06	4.67E-08
LPI	Pipe Segment Between Containment Isol. Valve (inside) and Cold Leg Injection	7	3.2E-02	1.30E-06	4.16E-08
LPI	Pipe Segment Between Containment Isol. Valve (inside) and Hot Leg Injection	8	3.2E-02	1.19E-06	3.80E-08
LPI	LPI Sources (RWST, Sump), Supply Line	9	3.64E-02	1.00E-06	3.64E-08
LPI	Pipe Segment Between Pump Discharge and Containment Isolation Valve	10	3.2E-02	8.63E-07	2.76E-08
LPI	Pipe Segment Between Containment Isolation Valves	11	1.6E-02	9.13E-07	1.46E-08
RPV	CRDMs	12	5.0E-04	1.00E-05	5.00E-09
RPV	Instrument Lines	13	5.0E-04	1.00E-05	5.00E-09
AFW	Pipe Segment Between Containment Isolation and SG Isolation Valves	14	8.49E-05	3.92E-05	3.33E-09
AFW	Main Steam to AFW Pump Turbine Drive	15	1.64E-04	1.28E-05	2.10E-09

System(b)	Component(c)	Rank	Conditional Core Damage Frequency Given Rupture	Rupture Frequency	Core Damage Frequency
RCS	Pipe Segment Between Loop Stop Valve and RPV (Cold Leg)	16	1.13E-02	1.42E-07	1.60E-09
LPI	LPI Pump Suction Line	17	1.36E-03	1.10E-06	1.50E-09
RCS	Pressurizer Spray Line	18	1.0E-04	1.00E-05	1.00E-09
RCS	Pipe Segment Between RPV and Loop Stop Valve (Hot Leg)	19	2.86E-03	2.00E-07	5.72E-10
AFW	AFW TD Pump Discharge Line	20	5.2E-05	1.02E-05	5.26E-10
AFW	Pipe Segment from Unit 2 AFW Pumps	21	1.4E-04	2.98E-06	4.18E-10
AFW	AFW Isolation Valve to SG	22	2.46E-06	6.51E-05	1.60E-10
RPV	Nozzle to Vessel Welds	23	3.0E-03	2.00E-08	6.00E-11
RPV	Vessel Studs	24	5.0E-04	1.00E-07	5.00E-11
AFW	AFW MDP Suction Line	25	1.2E-05	3.55E-06	4.27E-11
AFW	AFW MDP Discharge Line	26	1.65E-05	2.39E-06	3.95E-11
RPV	Upper, Closure Head, Flange	27	1.79E-03	2.00E-08	3.58E-11
RPV	Nozzle Forging Inlet/Outlet	28	1.25E-03	2.00E-08	2.50E-11
AFW	AFW TDP Suction Line	29	2.47E-06	6.12E-06	1.51E-11
AFW	Pipe Segment from Emergency Makeup System, Fire Main	30	3.9E-06	1.46E-06	5.71E-12
RCS	Pressurizer Relief/Safety Line	31	3.53E-07	6.41E-06	2.26E-12
RCS	Pressurizer Surge Line	32	1.5E-06	6.1E-07	9.15E-13
LPI-A	Accumulator Discharge Line	33	3.5E-08	2.62E-06	9.09E-14
RCS	Pipe Segment Between SG and RCP	34	3.05E-07	2.0E-07	6.10E-14
RCS	Pipe Segment Between Loop Stop Valve and SG (Hot Leg)	35	3.05E-07	1.41E-07	4.30E-14

System[b]	Component[c]	Rank	Conditional Core Damage Frequency Given Rupture	Rupture Frequency	Core Damage Frequency
RCS	Pipe Segment Between RCP and Loop Stop Valve (Cold Leg)	36	3.05E-07	7.75E-08	2.36E-14
LPI-A	Accumulator, Suction Line	37	3.5E-08	4.57E-07	1.60E-14

(a) Based on the estimated median valves.
(b) RPV = Reactor Pressure Vessel; AFW = Auxiliary Feedwater; LPI = Low Pressure Injection; LPI-A = Low Pressure Injection-Accumulator; RCS = Reactor Coolant System
(c) CST = Condensate Storage Tank; RCS = Reactor Coolant System; RWST = Refueling Water Storage Tank; CRDM = Control Rod Drive Mechanism; TD = Turbine Driven; SG = Steam Generator; MDP = Motor Driven Pump; TDP = Turbine Driven Pump; RCS = Reactor Coolant Pump

To prepare for future probabilistic fracture mechanics calculations we organized an informal meeting on June 25, 1991 in San Diego, California with key members of the Research Task Force on Risk-Based Inspection. This meeting was scheduled during the ASME Pressure Vessel and Piping Conference. During the discussions O.J.V. Chapman (Rolls-Royce and Associates Limited, UK), B.A. Bishop (Westinghouse Electric), and D.O. Harris (Failure Analysis Associates) demonstrated computer software developed by their respective organizations, and also described examples of how ISI effectiveness has been included in their structural reliability calculations.

The PRAISE code for predicting leak and rupture probabilities has been obtained and made operational at PNL. This code predicts rupture and leak probabilities for reactor piping, and includes the effects of inservice inspection on piping reliability. In obtaining the PRAISE code and becoming knowledgeable with its application, we have been interacting with research staff at Lawrence Livermore National Laboratory, Failure Analysis Associates, and Idaho National Engineering Laboratory. A subcontract is being placed with Failure Analysis Associates (the developer of the PRAISE code) so they can train us to better use the code and assist us in making appropriate modifications.

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 systems at Surry-1.
- Begin probabilistic fracture mechanics calculations to evaluate the effects of inservice inspection.
- Continue with the ISI prioritization for components at Surry-1.

The continuing pilot calculations serve to focus inspections on the high risk components. Structural reliability calculations will be used in the development of risk-based inspection programs for the high priority components. Part of the evaluations will be 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.

5.0 Consult on Field Problems

5.1 Summary

Efforts continued to support NRC staff in the application, validation, and modification of the VISA-II fracture mechanics code for predicting the contributions of undetected flaws to the failure of reactor pressure vessels under conditions of pressurized thermal shock accidents. Recent work to validate the VISA-II code revealed no significant errors, although some corrections were made to enhance the overall utility of this code. New features include modeling of residual stresses, modeling of underclad cracks, programming the chemistry factor, and improved plastic instability predictions. An updated version of the VISA-II code with documentation of all corrections and enhancements was prepared and made available to the NRC staff.

During this period another request was made by RES to provide support in the development of a technical position regarding industries interest in digitizing radiographs. PNL provided support in the development of performance demonstration requirements for personnel that would be involved in this process.

5.2 Introduction

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).

5.3 Status of Work Performed

5.3.1 VISA-II Code

We continued to support NRC staff in the application, validation and modification of the VISA-II fracture mechanics code. This code predicts the contributions of undetected flaws to the failure of reactor pressure vessels under conditions of pressurized thermal shock accidents, and is currently in use by NRC staff and others for plant specific vessel evaluations.

Recent efforts have been successful in further validating the VISA-II code. No significant errors in VISA-II that could cause serious errors in calculated failure probabilities have been found. Some corrections have been made to the output summary table that lists certain details of initiation and arrest events, and to avoid occasional inconsistent output data. The procedure for counting multiple initiation events has been corrected to properly deal with severely embrittled vessels for which more than one flaw may initiate. The potential impacts of the code corrections have been addressed and reported to NRC staff.

Certain features of the VISA-II code had not been extensively tested when the code was originally developed. Additional testing has further validated the correct implementation of these features into the code. Some key features of interest were the simulation of warm pre-stress, stress intensity factor solutions for finite length flaws, and the simulation of fracture toughness for crack arrest predictions.

New and enhanced features have been added to the VISA-II code. These include the modeling of residual stresses, modeling of underclad cracks, programming the table version of the Regulatory Guide 1.99 Rev 2 chemistry factor, and consideration of pressure on crack faces in plastic instability predictions.

A new version of the VISA-II code with documentation of all corrections and enhancements has been prepared and made available to NRC staff. We believe that confidence in VISA-II has been significantly increased by recent PNL efforts, and that the new version of the code will prove to be useful for future plant specific evaluations of pressurized thermal shock.

5.3.2 Performance Demonstration for Digitizing Radiographs

There is a lot of interest by industry to digitize their radiographic film and store the digital information on a high density media such as an optical disk. There are attractive reasons for industry to do this since the digital information will not degrade and there are economical cost incentives that also make it attractive. However, the digital storage media does degrade but at a much slower rate than film. Industry also wants to be able to recycle their radiographs after they are digitized which means that the original source of information will be destroyed and only the digital information will exist.

However, from the standpoint of the NRC there are technical issues of concern.

PNL became involved because the skills required to interpret digitized radiographic data on a video display is quite different from those required for reading a film. The bottom line is that it was felt that a performance demonstration would result in providing evidence that interpreters could equally well arrive at the same decision whether they were using film or digitized data. It needs to be noted that video displays can only display at most 8 bits of grey scale while a film has 12 bits of information. The film interpreter will use a control on his light source to adjust for film density variations and the interpreter of video displayed data must use other controls to accomplish the same compensation.

PNL staff provided input on the development of performance demonstration requirements that became part of a document that was being developed by the NRC on this topic.

5.4 Future Work

During the next reporting period we will continue to support NRC staff in the application, validation, and modification of the VISA-II fracture mechanics code. This will include partipation in a NRC/EPRI meeting on PTS Analyses and Modeling Assumptions at Anaheim, California on November 21, 1991 and a presentation at this meeting on VISA-II Structure and Modeling Assumptions.

Provide continued support to the NRC while the technical position for the digitation of radiographic film is being finalized.

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: 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:

- 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 a workshop conducted for NRC Regional personnel on ultrasonic testing/inservice inspection of cast material and submission of a paper documenting an imaging technique to characterize localized changes in microstructure. Input for a document describing a UKAEA/NRC cooperative effort on cast materials was drafted, a strategy change to employ techniques inherently insensitive to microstructural changes rather than specific classifications of microstructures was prepared for the effort for the next fiscal year, and Dr. J. L. Rose at Drexel University was visited.

- Surface Roughness Conditions - 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. Significant progress was achieved by transferring the first version of the CNDE model to PNL and comparing model predictions with experimental data. Also, the responsibility for model development at CNDE was transferred from Mr. Ali Minachi to Dr. Isaac Yalda. R. B. Thompson continues in his role of supervising model refinement at CNDE at the Iowa State University.

- 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. Documentation of the five safe-ends removed from the Monticello nuclear power station are no longer needed for any programmatic work. Processes to dispose of these samples are under review to meet the new requirements for disposal required by state and federal agencies.

- PISC-III Activities - 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 major efforts focused on trying to get participation of teams from the USA in the capability studies.

6.1 Cast Stainless Steel Inspection

6.1.1 Introduction

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 passed through weld material. CCSS is an area which received attention this reporting period as well as a workshop to NRC personnel on the ISI of cast 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 microstructure that characterizes this material. The major microstructural classifications are columnar, equiaxed, and a mixed columnar-equiaxed microstructure of which the majority of field material is believed to be the latter.

6.1.2 Status of Work Performed

Activities during this reporting period included the following: participation in a workshop to NRC Region personnel concerning problems relating to the ISI of cast material, submission of a paper to the NRC Program Manager documenting an imaging technique to characterize localized changes in microstructure, the draft input of a cast-material cooperative effort between UKAEA and NRC was prepared, a strategy change to employ techniques inherently insensitive to microstructural changes rather than specific classifications of microstructures was prepared for next fiscal year, and a visit to Dr. J. L. Rose of Drexel University.

A NRC workshop for Region personnel was conducted at PNL during April 1991. The workshop had a session on CCSS inspectability and consisted of a morning and two laboratory demonstrations/experiments which encouraged hands-on experience with ISI of CCSS. The presentation focused on the anisotropy of the austenitic grains; differences of microstructure between cast stainless steel, wrought stainless steel, and low alloy steels; why false signals originate from the grain boundaries, results from CCSSRRT, selective frequency filtering of ultrasonic signals by CCSS microstructures, ultrasonic field distortion, and ultrasonic attenuation. The demonstrations/experiments then re-enforced points made during the lecture by having groups classify microstructure by ultrasonic techniques and attempt to image defects in CCSS pipe sections.

A paper entitled "Application of Critical Angle Imaging to the Characterization of Cast Stainless Steels" was approved by the NRC Program Manager for submittal to the annual "Review of Progress in Nondestructive Evaluation" conference to be published in the proceedings. Dr. Hermann Wustenburg of BAM, Germany also visited PNL prior to the conference and was presented a summary of the cast material task. He was favorably impressed with the combined usage of critical angle measurements and imaging. His opinion was that the spatial relationship displayed in an image format would greatly facilitate the process of microstructural classification. A draft NUREG/CR report on critical angle imaging was submitted to the NRC Program Manager during the last reporting period. PNL is awaiting NRC guidance concerning future actions relating to this report draft.

A draft of a cooperative effort between UKAEA and NRC was prepared and sent to the NRC Program Manager as reported in the previous semi-annual report. During this reporting period the draft was forwarded by the NRC Program Manager to Mr. William Gardener of UKAEA as a reviewer. A related activity was sending some micrographs of CCSS macrostructures to Dr. Andrew Temple of Harwell, per his request. These were to serve as a guide for microstructures that the UKAEA needed to consider in their model development. PNL is awaiting comments from the NRC concerning the cooperative draft before initiating further activity. Several revisions of the work plan are expected before mutual approval by the NRC, PNL, and the UKAEA.

An April 1991 program review was performed by the NRC Program Manager. During the review several issues were discussed of which one was the strategy of classifying microstructure versus development of techniques which were less sensitive to changes in microstructure. Since little evidence existed for successful inspections of selected microstructures, a strategy change was agreed upon in which PNL would address techniques that would attempt to lower sensitivity to the microstructure of cast material; e.g., low frequency L-waves, pitch-catch arrangements with detection by means of the flaw blocking sound transmission, etc.

A visit was made with Dr. Joseph L. Rose at Drexel University to discuss the use of Lamb waves to inspect steam generator tubing and modification of the technique for low frequency inspection of CCSS. Prior to the visit, papers by Dr. Rose were reviewed concerning his theoretical studies and their potential in CCSS microstructure characterization. The purpose was in part to evaluate the use of a consultant to mentor PNL in the use of Lamb wave inspection where wave lengths are on the order of a pipe wall thickness (i.e., 6 centimeters).

6.1.3 Future Work

CCSS work will focus on implementing techniques that will be less sensitive to the problems caused by coarse microstructure; e.g., use of much lower frequencies such as 100 kHz, pitch-catch geometries where large flaws would be detected by blocking signal transmission, and techniques that could automatically adapt to the phase distortion due to the microstructure, etc.

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. These efforts, however, will be done at a lower priority due to limited funds.

6.2 Surface Roughness Conditions

6.2.1 Introduction

The objective of this task 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.

Past efforts included an empirical attempt to quantify the effect produced by an outer surface irregularity to understand the severity of the problem. This approach was then redefined to take advantage of an ultrasonic model developed under EPRI funding at the Center for NDE (CNDE) at Ames Laboratory. A cooperative effort between the NRC and EPRI was then approved. Under the auspices of the cooperative agreement, the 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.

Significant progress was made this reporting period by means of transferring the first version of the CNDE model to PNL and comparing between model prediction and experimental data. Another significant event was the transfer of responsibility of model development at CNDE to Dr. Isaac Yalda from Mr. Ali Minachi. (Dr. Isaac Yalda has assumed the responsibility of model development since Mr. Ali Minachi has shifted his focus to completing his Ph.D. thesis.) Dr. R. B. Thompson continued in his role of supervising model refinement at CNDE at Iowa State University (ISU).

6.2.2 Status of Work Performed

Work included PNL preparation for receiving the CNDE model, continued refinement of the model at CNDE, a joint paper by CNDE and PNL, and the recognition of discrepancies between model and experiment.

A simple modelling exercise was performed by PNL in preparation for receiving the CNDE model. The purpose of this exercise was to gain insight into the use of models so that better use could be made of the isotropic model when delivered to PNL. Activity began in July 1990 and was completed with a letter report as a deliverable to the NRC Program Manager (see Appendix A). The main conclusions were:

1. The effect of transducer tilt as the search unit is displaced across a sufficiently wavy or rough surface will result in some volumetric regions not being scanned.
2. A crack response may be mis-interpreted as being further away than it actually is and thereby make it harder to distinguish it from the weld root signal.
3. A crack may be systematically undersized due to the displacement error as referenced in the earlier conclusion.

Delays have occurred at CNDE in past reporting periods due to funding cuts at EPRI. At the beginning of this reporting period, Dr. Thompson reaffirmed that CNDE was committed to completing the isotropic model and recommended that a favorable comparison between model prediction and experiment be done prior to transferring the model to PNL. In anticipation of receiving the model, PNL proceeded with ordering Fortran 77 to implement the model on a Sun Sparc-Station 2 and PV-wave to graphically display model predictions.

Dr. R. B. Thompson, Mr. Ali Minachi, and Dr. Isaac Yalda of the CNDE have been refining the model while PNL staff have been refining procedures for acquiring experimental data. PNL collected 0°, L-wave field maps that passed over a 1.5-mm step discontinuity. The field maps were transferred to CNDE from PNL and the data compared quite well with model predictions. This represented the first successful electronic mailing of data between ISU and PNL.

An amplitude discrepancy between model prediction and experiment was apparent from the data at the amplitude null occurring between the maximum peaks of the 1/2V and 3/2V paths. The response of the experiment was greater than that predicted by theory. One explanation was that transient responses had corrupted the quasi-monofrequency response. By increasing the tone burst from 4 cycles to 8-20 cycles and using a signal-following gate the distortion was significantly reduced. Work then continued with data acquired for 0° and 45° L-wave field maps over the full sample thickness with no step as well as for three step discontinuities that had been agreed would be a good test to validate the model predictions. Preliminary experimental data appeared to match quite well with Gauss-Hermite predictions.

A joint paper between CNDE and PNL was submitted to the annual "Review of Progress in Quantitative Nondestructive Evaluation" conference. Mr. Ali Minachi of ISU presented a comparison between model predictions and PNL experimental data at the conference. Experimental data compared well with model predictions for a 0°, L-wave field incident on a step discontinuities which simulated an abrupt surface change from an excessive weld crown. A discrepancy, however, was reported for a 45°, L-wave and spurious signals were observed in the experimental data that interfered with the 45°, SV-waves. (Appendix B is a copy of the paper submitted for publication in the conference proceedings.)

The model code was transferred from CNDE to a PNL directory at ISU which permitted PNL to electronically acquire its first version of the model in September 1991. This was a significant accomplishment and CNDE and PNL found that electronic mailing was a quick and effective method to transfer computer codes between the two institutes. Most significantly, a code version now existed at PNL.

6.2.3 Future Work

The schedule of model refinement, model validation by PNL, and use of the model to determine code recommendations for surface condition will be addressed at the beginning of the fiscal year. Due to the limited resources, PNL will probably restrict validation tests on the CNDE model this next fiscal year for one of the four geometric conditions defined in past work; i.e., excessive weld crown (step discontinuity), diametrical shrinkage (convex shape), over ground (concave shape), and weld splatter (point pivot). Excessive weld crown will probably be selected due to its importance to field applications.

6.3 Field Pipe Characterization

6.3.1 Introduction

The objective of this subtask is to provide pipe weld specimens for studies to evaluate the effectiveness and reliability of ultrasonic inservice inspection (UT/ISI) performed on BWR piping. Documentation of the five safe-ends removed from the Monticello nuclear power station was reviewed and an extensive data package was assembled to accompany shipment of these specimens to Joint Research Centre, Ispra, Italy. However, PISC-III representatives decided that evaluation of removed-from-service components could no longer be supported by the PISC-III program. Processes to dispose of these samples are under review.

6.3.2 Status of Work Performed

The focus of the work on this activity was in trying to decide what are the requirements that will need to be met in order for these safe-ends to be properly buried. The requirements of state and federal agencies are rapidly changing and the process for burial is not straight forward.

6.3.4 Future Work

It is hoped that during the next reporting period that the requirements that will need to be met will be clarified and that the burial of the safe-ends can proceed.

6.4 PISC-III Activities

6.4.1 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.

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.

6.4.2 Status of Work Performed

The primary activities that occurred this reporting period included the participation in a PISC III Management Board meeting, work with some U.S. teams that participated in the PISC III Action 3 to resolve areas of concern on the proper interpretation of their data, and work to try to get teams form the U. S. to participate in the Action 4 studies.

The schedule for the PISC III program is getting tight in terms of completion of the work by the end of the calendar year 1993. This is the deadline for the completion of all work. However, the funding for this program has not been front end loaded as was needed and as a result things have been extended and now almost all of the activities are on the same schedule for completion. This is a problem from the standpoint of trying to get all of the destructive assay work performed and the data analysis performed. All of the actions will be competing for the same resources at the Joint Research Centre in Ispra, Italy. This will make getting all of the work accomplished a significant challenge.

Clearly progress is being made and several phases are completed and data analysis is occurring. The phase 1 of the FSV and the NDW are nearly completed. The capability study on wrought stainless steel of the AST is well under way and the specimen for the cast-to-cast and the cast-to-wrought are being fabricated. The participation of teams from the U.S. in the AST is a disappointment at this time since there are none confirmed. Several teams have provided verbal interest but this is not confirmed.

A PNL staff member participated as an invigilator for the studies of Action 7 by going to England and supporting for 2 weeks inspection exercises that were being conducted. These were thought to be well designed tests and that they will be generating useful information on this aspect of the reliability issues.

6.4.3 Future Work

Future will include attending future PISC III Management Board meetings, further coordination of teams from the U.S. for the AST and involvement in the data analysis of the data from the AST.

7.0 References

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Appendix A: Modeling of a Step Discontinuity

MODELING OF A STEP DISCONTINUITY

Margaret S. Greenwood

1.0 INTRODUCTION

Work is ongoing to recommend ASME Code changes regarding surface specifications for ultrasonic inservice inspection (ISI) of light-water-reactor piping and reactor vessels. The selected approach is to employ a validated computer model. To gain insight into the use of models, another simpler model is developed to perform a cursory analysis of how a weld crown can affect ISI of pipe welds. With this experience, better usage of the sophisticated model is expected and situations where it might be difficult to detect flaws can be evaluated. The technical goal of this modeling activity is to consider how the central beam from a probe travels through a stainless steel pipe as the probe passes over a weld crown (step discontinuity). (The sophisticated 3-dimensional model is being developed jointly by the NRC and EPRI. EPRI is sponsoring work at the Center for NDE at Iowa State University to refine an already existing model. NRC is sponsoring PNL to provide experimental data for model refinement and to assist in model validation.)

When the probe is tilted as it travels over a step discontinuity, the gel occupies the space between the probe and the horizontal metal surface. As a result, refraction occurs at a plastic-gel interface and another at a gel-metal interface. The amount of refraction obviously depends critically upon the speed of sound in the gel. One might expect it to be somewhat larger than that in water since the gel is stiffer. In any case, it seemed very important to measure this quantity. The experiments obtained the time to travel a known distance through the gel and yielded a value given by

$$\textit{Speed of Sound in Gel} = 0.081\ \textit{inches/microsec}.$$

This is about 39% larger than the speed of sound in water, which is 0.058 inches/microsec.

To begin the modeling, we shall consider a particular transducer and a particular step discontinuity. It's helpful to carry out some calculations first and then the general equations can be developed for any case.

2.0 PROPERTIES OF THE TRANSDUCER

The transducer had a frequency of 2.25 MHz and its size was 0.5 by 0.5 inches square. It was designed to produce a 45° shear wave in steel. Since the speed of the shear wave in steel is 0.125 inches/microsec and the speed of the longitudinal wave in plastic is 0.107 inches/microsec, Snell's law yields a value of θ_p equal to 37.25°, where θ_p is the angle that the wave in the plastic wedge makes with the normal to the bottom surface of the wedge. The wedge has a length of 1.16 inches.

3.0 MOTION OF THE TRANSDUCER ACROSS THE STEP DISCONTINUITY

We shall use the coordinate system shown in Figure 1 to describe the location of the transducer and that of point P, which is where the central beam reaches the bottom surface. We are particularly interested in the coordinates of point P and how they change as the transducer goes over the step. We shall also be interested in the time to reach point P and return to the transducer.

Figures 2 and 3 show an enlarged view of the ray traveling across the two interfaces. The calculations following Figure 2 show how to determine the location of point P as well as the round-trip time. We assume that the transducer goes over the step as shown in Figure 2 until the distance EF equal 0.4 inches and then it rotates to the level position shown in Figure 4, where the x coordinate of point H is 0.5 inches. This is called the final position.

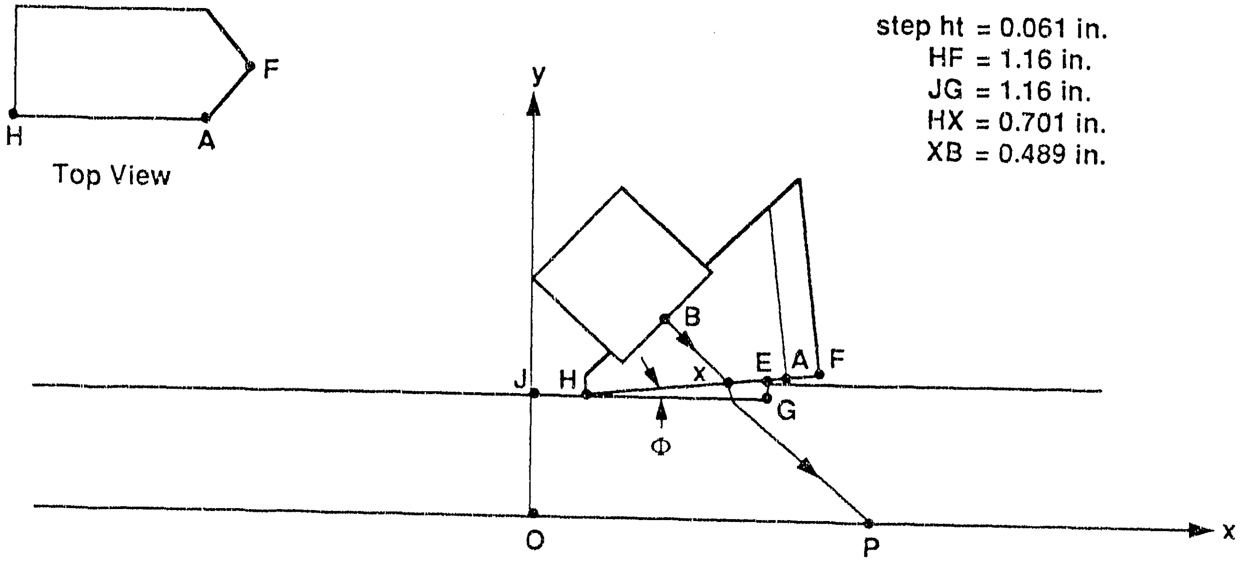

FIGURE 1. Transducer Crossing Step Discontinuity

Angle Φ is found from

$$\sin \Phi = \frac{0.061}{1.16 - EF} \tag{1}$$

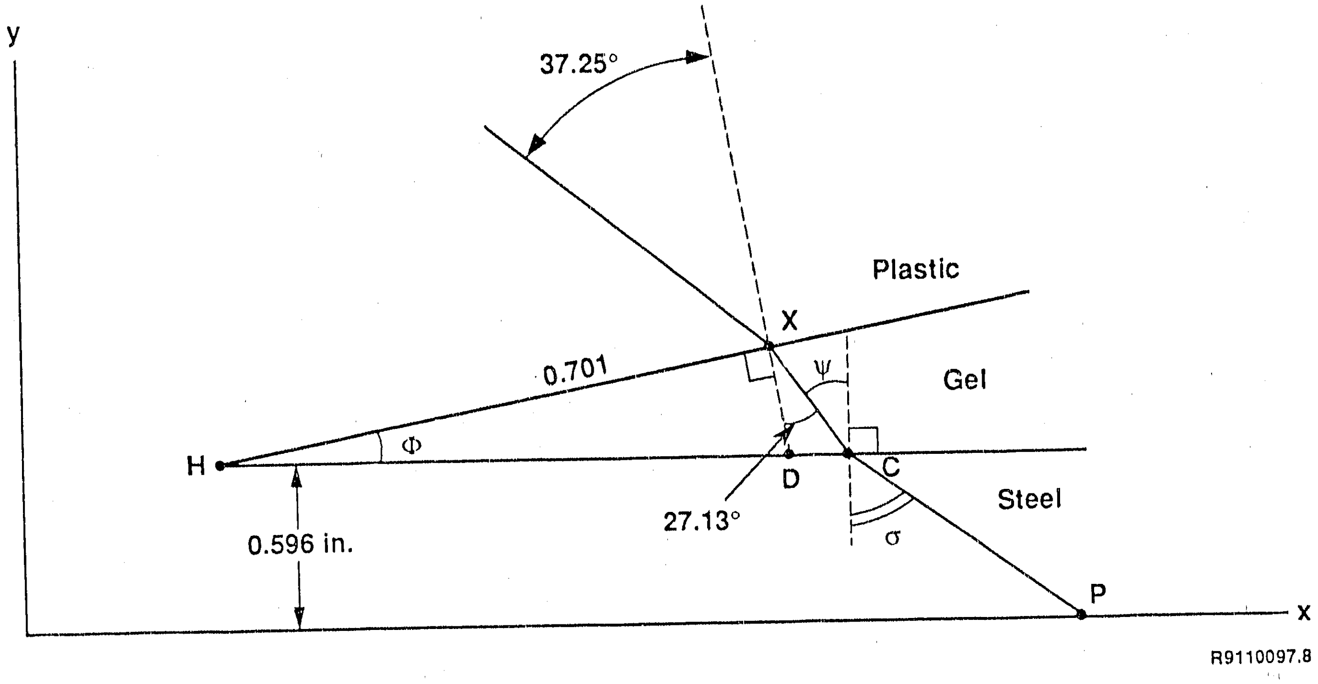

FIGURE 2. Enlarged View of Ray Traveling from Plastic through the Coupling Gel and into the Steel Plate

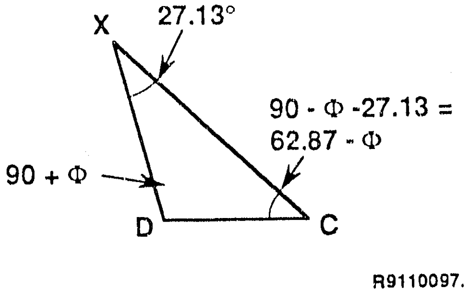

FIGURE 3. Enlarged View of Triangle XDC

The distance HG is given by

$$HG = \left[(1.16 - EF)^2 - (0.061)^2\right]^{0.5}$$

The location of point H is given by

$$H_x = 1.16 - HG \tag{2}$$

Using Snell's law at the plastic-gel interface we find:

$$\frac{\sin 37.25}{.107} = \frac{\sin \Theta_G}{.081}$$

$$\Theta_G = 27.13^\circ$$

From triangle HXC, we find that

$$\Psi = \Phi + 27.13$$

Using Snell's law at the gel-steel interface, we find

$$\frac{\sin \Psi}{.081} = \frac{\sin \sigma}{.125} \tag{3}$$

$$\sin \sigma = \left(\frac{.125}{.081}\right) \sin (\Phi + 27.13)$$

The next step is to determine the distance HC. First, we must determine the distances HD and DC. From triangle HXD, we obtain

$$XD = 0.701 \tan \Phi \quad and$$

$$HD = \frac{0.701}{\cos \Phi}$$

From the oblique triangle XDC, we can find XC and CD using the law of sines:

$$\frac{\sin (62.87 - \Phi)}{XD} = \frac{\sin 27.13}{DC} = \frac{\sin (90 + \Phi)}{XC}$$

Substituting for XD the value obtained above, we find:

$$DC = \frac{0.701 \tan \Phi \sin 27.13}{\sin (62.87 - \Phi)} \quad and$$

$$XC = \frac{0.701 \tan \Phi \sin (90 + \Phi)}{\sin (62.87 - \Phi)}$$

The distance HC is given by

$$HC = HD + DC$$

The location of point C relative to the coordinate system is given by

$$C_x = H_x + HC = H_x + HD + DC$$

The location of point P relative to the coordinate system is given by

$$P_x = C_x + 0.596 \tan \sigma = H_x + HD + DC + 0.596 \tan \sigma$$

$$P_x = 1.16 - [(1.16 - EF)^2 - (0.061)^2]^{0.5} + \frac{0.701}{\cos \Phi} + \frac{0.701 \tan \Phi \sin 27.13}{\sin (62.87 - \Phi)} + 0.596 \tan \sigma \qquad \textbf{(4)}$$

We can calculate the round-trip times using the distances obtained above. Basically, the time is given by:

$$time = \frac{2(distance\ in\ plastic)}{.107} + \frac{2(distance\ in\ gel)}{.081} + \frac{2(distance\ in\ steel)}{.125}$$

In the initial position the transducer has just reached the step so that the front end is at the step. It has not yet been rotated so that the transducer is flat against the metal surface. Here, the wave travels a negligible distance in the gel. The distance that the beam travels in plastic (XB) is 0.489 inches and that in steel is the thickness of the steel (0.596 in.) divided by cos 45. Inserting these values in the above equation yields a value of 22.63 microsec.

When the transducer is in the tilted position, the distance that the beam travels in the gel is given by the distance XC and that in steel, by CP, as shown in Figure 2. XC and CP have been determined above in terms of the angle Φ. The round-trip time is therefore given by

$$time = \frac{2(.489)}{.107} + \frac{2(.701)\tan\Phi\sin(90+\Phi)}{.081\sin(62.87-\Phi)} + \frac{2(0.596)}{.125\cos\sigma} \qquad (5)$$

In the final position the transducer has rotated across the step and is flat against the top surface, as shown in Figure 4. The calculation for the round-trip time is very similar to that for the initial position except that the thickness of the steel is now increased by the step height and is equal to 0.657 inches. The round-trip time is equal to 24.01 microsec.

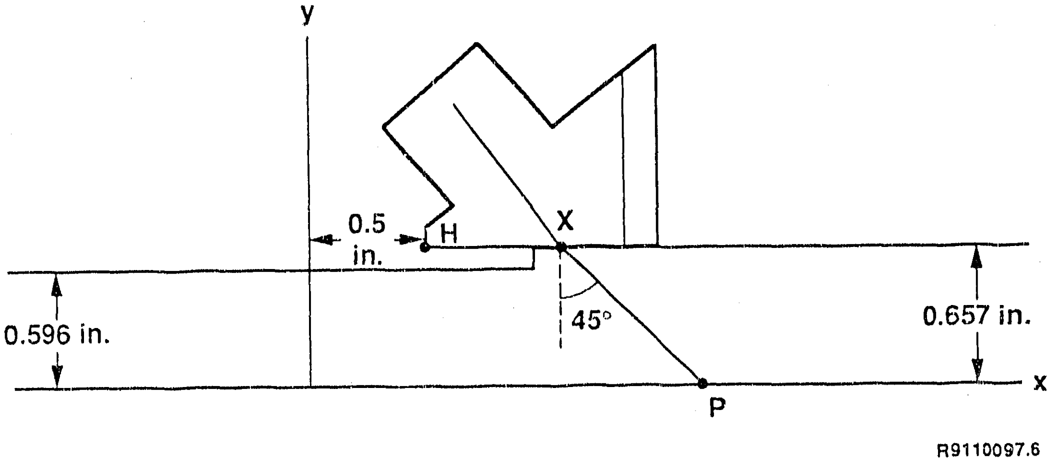

FIGURE 4. Final Position of Transducer After Traveling over Step Discontinuity

The positions of point P and the round-trip times were calculated using a spreadsheet. Angles Φ and σ were calculated from Eq. 1 and 3, respectively; H_x, from Eq. 2; P_x, from Eq. 4; and the total time, from Eq. 5. Results are shown in Table 1.

We note that point P shifts from a location of 1.30 inches in the initial position to 1.46 inches just in rotating the transducer to go over the step and that the round-trip time increases by about 3 microsec. In the coordinate system used here, the step is located at 1.16 inches. Therefore, it might be difficult to see a defect located from 0.14 inches to 0.30 inches behind the step. Similarly, when the transducer rotates from the tilted position back to its final position on top of the step, the central beam sweeps from a position of 1.98 inches to 1.86 inches. Therefore, the effect of the rotation results in some areas not being scanned very well by the central ray of the transducer.

TABLE 1. Run 1: Motion of Transducer Across Step Discontinuity

(change in location of point P and round-trip time)

dr	0.017453	changes angle from degrees to radians
rd	57.29578	changes angle from radians to degrees
thgel	27.13	angle theta in gel
hx	0.701	distance HX on transducer
stpht	0.061	step height
hf	1.16	total length HF of transducer
thkst	0.596	thickness of steel plate

EF	Φ	σ	H_X	P_X	Total Time
Initial Position	0		0	1.297	22.62607
0	3.016961	51.15749	0.001606	1.463204	25.40332
0.1	3.302151	51.77051	0.101758	1.581886	25.71229
0.184	3.587001	52.38940	0.185910	1.685212	26.03127
0.2	3.646927	52.52047	0.201942	1.705367	26.09977
0.3	4.072161	53.45967	0.302168	1.835803	26.60064
0.4	4.609770	54.67142	0.402455	1.976899	27.27460
Final Position			0.5	1.858	24.00633

The increased times in the tilted position leads to another possibility. The response from a crack might suggest that it is positioned deeper and farther away than it actually is. Hence, it will be harder to distinguish the response from a crack and the root signal.

The tip signal is used to determine the siz eof a crack. The following relationship gives the size a:

$$a = \frac{vt}{2\cos\theta}$$

where t is the time between the tip signal and that from the bottom of the crack. Normally, θ is 45 degrees, but when the transducer is tilted the beam hitting the crack makes a larger angle with the vertical. For example, the value of angle sigma in Table 1 ranges from 51° to 55°. If one used 45° rather than the correct value of, say, 51° in the formula, the value of the length a would be smaller than it actually is. Therefore, for a crack tip the tendency would be to position the crack farther away and to systematically undersize the crack.

4.0 COMPARISON WITH EXPERIMENT

We made measurements using the stainless steel plate shown in Figure 5, which contains a step discontinuity having a height of 0.061 inches and notches at various intervals behind the step. Results using this plate have previously been reported by M. S. Good.[1]

4.1 SPEED OF SOUND IN GEL COUPLANT

To measure the speed of sound in gel, we filled notch F with gel, placed the transducer on top, and obtained the oscilloscope trace shown in Figure 6. In this case, a transducer (rated at 2.25 MHz with dimensions 1.07 in. by 0.72 in.) was used that produced a wave perpendicular to the surface. The cursors are set on successive reverberations in the gel and the time is 8.13 microsec. Note that the first pulse from the reflection of the gel-steel interface also occurs at 8.13 microsec. The notch depth is 50% of the plate thickness or 0.3285 inches. The speed in the gel is given by 2(.3285)/8.13, or 0.081 inches/microsec.

4.2 CORNER PULSE

To find the location of the transducer which would produce a pulse of maximum voltage, we experimented with the corner pulse. That is, the beam is reflected from the vertical surface at the edge of the plate. Point A on the transducer is a convenient reference point since there is an edge there. We found that point A should be located 0.375 inches from the vertical surface of the plate. Since the distance AX is 0.274 inches, this means that point X is a distance of 0.375 + 0.274 = 0.649 inches from the vertical surface. Since the plate thickness is 0.657 inches, this is in very good agreement with the expected result for a pulse in steel at 45°.

4.3 EXPERIMENTS WITH NOTCH B

We carried out experiments using notch B, which is 0.457 inches behind the step. Optimally point X on the transducer should be a distance of 0.649 inches from the notch for a maximum signal. This means that notch B must be observed as the transducer goes over the step, as shown in Figure 7a.

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[1] M. S. Good. Significance of Surface Condition Upon Ultrasonic Inspection. Internal report, June 1987.

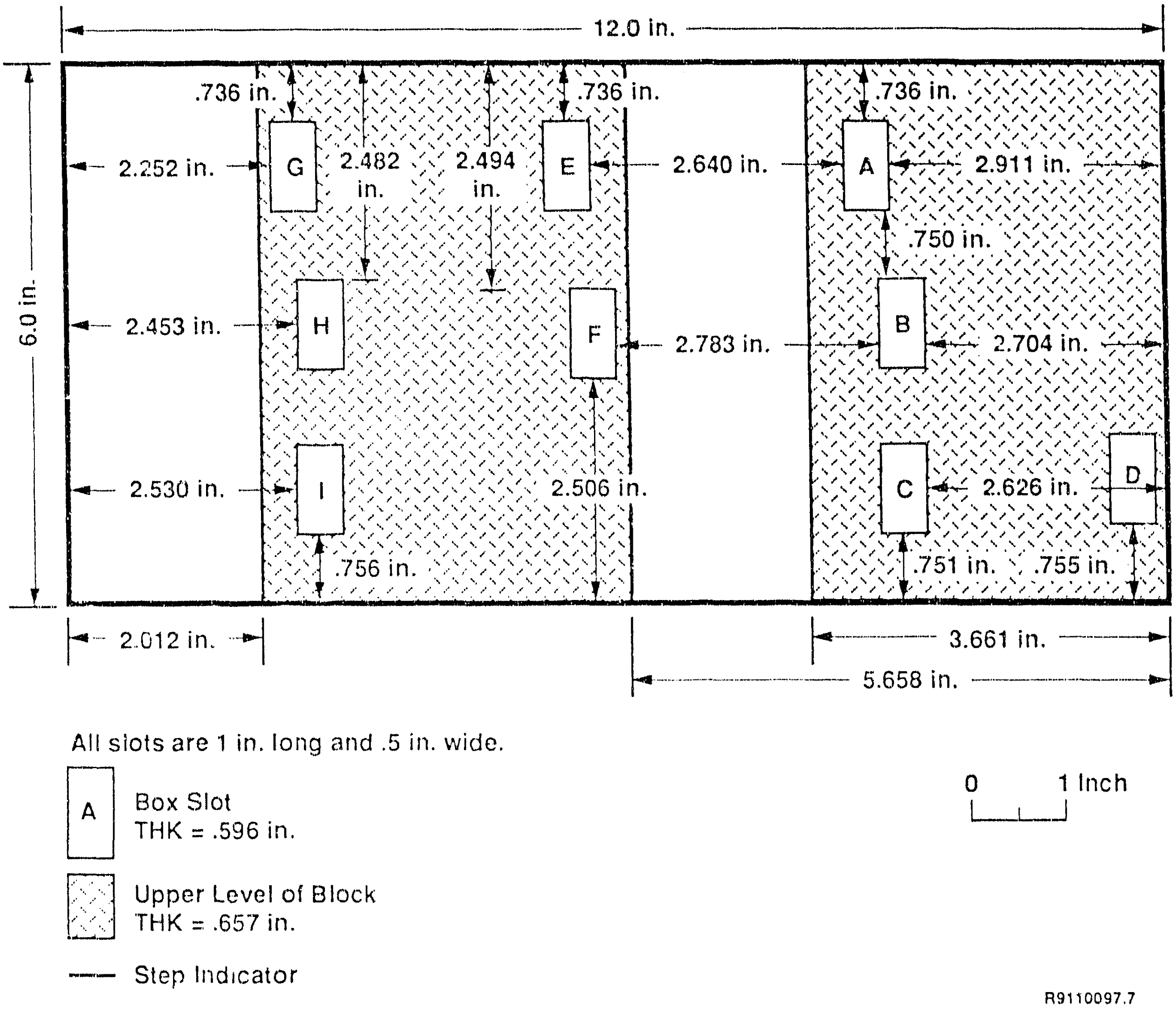


FIGURE 5. Top View of Steel Block Showing Steps and Notches Located on the Bottom Surface. Notches A, B, C, and D are 10% of thickness of the steel (0.657 inches); notches E and F are 50% of thickness; and notches G, H, and I are 20% of thickness.

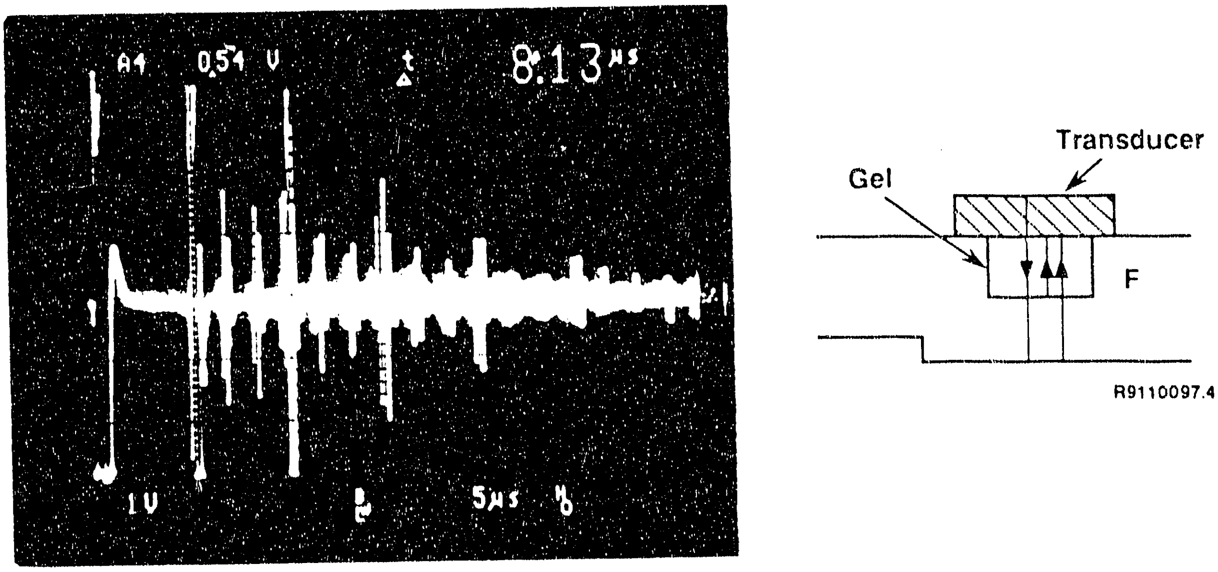


FIGURE 6. Oscilloscope Trace for Waves Directed Perpendicular to Notch in Steel Plate Filled with Gel. The larger peaks show successive reverberations of waves from the gel-steel interface and the transducer surface. The smaller peaks show successive reverberations of waves from the gel-steel interface and the bottom of the steel plate.

When the transducer was held at an angle over the step so that point A was at the corner of the step, as shown in Figure 7b, we observed a signal with a voltage of 1.31 volts and a time of 24.78 microsec. The distance AF is equal to 0.184 inches. Table 1 shows that when EF equals 0.184 inches, the location of point P is 1.69 inches and the round-trip time is 26 microsec. Thus, the pulse is found very nearly where it was expected: 1.62 inches experimentally versus 1.69 inches theoretically. Very good agreement! The time is also quite close: 24.78 microsec experimentally versus 26.0 microsec theoretically. (However, when the experiment was repeated a time of 22.28 microsec and a voltage of 1.07 volts were obtained.)

It is also possible to obtain a signal when the transducer is held level over the step as shown in Figure 7c. Point A was measured to be 0.225 inches to the right of the step and we saw a signal having a voltage of 1.38 V and a time of 22.96 microsec. Point X on the transducer is 0.049 inches in from of the step. This means that point X is 0.506 inches from notch B. This distance is close to the thickness of metal (0.596 in), which would yield very nearly a 45° shear wave in the steel.

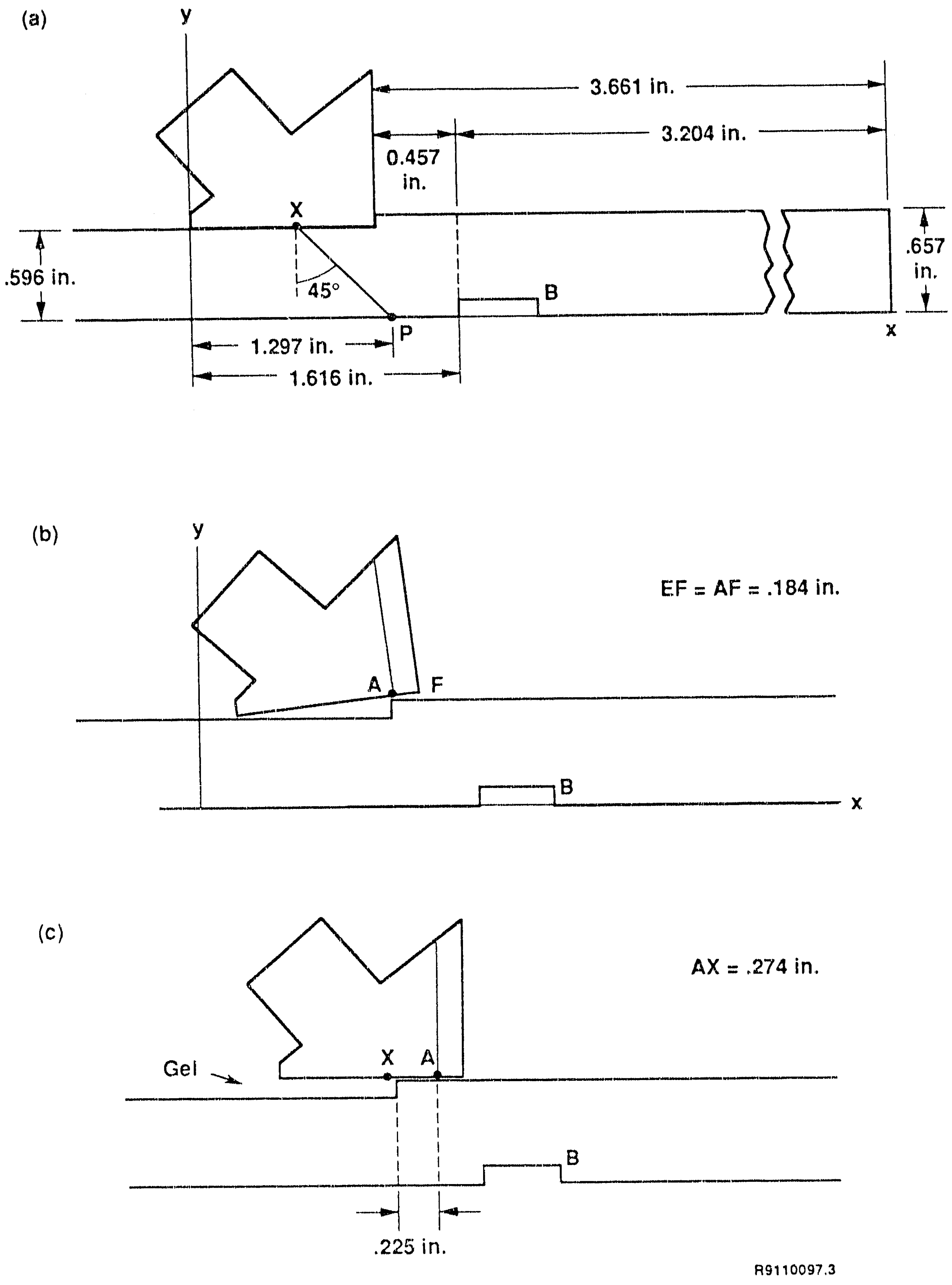


FIGURE 7. Orientation of Transducer Relative to Step Discontinuity for Measurements Obtained for Notch B: a) initial position, b) point A located at edge of step in tilted position, c) transducer held level over the edge of step

4.4 EXPERIMENTS WITH NOTCH F

When the transducer is tilted as it travels over a step discontinuity, refraction occurs at the plastic-gel interface and another at the gel-metal interface. Since the speed of sound in the gel is less than in steel or plastic, significant refraction occurs at each interface and the angle of the shear wave in the steel will be greater than 45°. The beam, therefore, will travel a greater distance in the steel and hence, the time will be larger. Because of this larger angle, the transducer must be positioned at a larger distance (compared to the normal horizontal transducer position) from a perpendicular notch. The effects just described are what the calculations show based upon refraction at the two interfaces. The object of these experiments is to observe these effect. To do that, we chose notch F in Figure 5, but we did not want to involve the step discontinuity. Therefore, we chose to observe the left side of notch F so that the transducer moved toward the right in the region between notch H and notch F. The object of these experiments is to mimic a step discontinuity by tilting the transducer and at the same time to have a smooth surface over which to move the transducer, as shown in Figures 8 and 9. Results from these experiments could then be compared with the predictions from this simple model.

The first step was to observe notch F in the normal fashion as indicated in Figure 8a. Since experiments with the corner pulse showed that point A should be 0.375 inches from the vertical surface, the dimensions show that to observe notch F, point A should be 1.204 inches from the step. Here, the step serves only as a convenient reference point. When the experiment was carried out, we found that the maximum signal occurred when point A was 1.192 inches from the step. The voltage was 3.34 V and the time was 20.37 microsec. The oscilloscope trace is shown in Figure 8b.

The second step was then to tilt the transducer by placing a wooden piece 0.072 inches thick under the tip of the transducer as shown in Figure 9a so that the distance EF (defined in Figure 1) is equal to 0.1 inch. The calculations are shown in Table 2. We see that, when the transducer goes from its normal horizontal position to a tilted one, the transducer should be moved back by an amount equal to 1.70 minus 1.36, or 0.34 inches. The time should also increase from 24.0 to 26.4 microsec. Therefore, point A should be located at 0.34 + 1.20 = 1.54 in. The transducer was placed in this position and we found a pulse at 24.03 microsec with a voltage of 1.12 volts. The oscilloscope trace is shown in Figure 9b. The model is successful in predicting where a pulse should be located. Experimentally, we find that the time increases from 20.37 to 24.03 microsec. The theory, however, predicts an increase from 24.0 to 26.0 microsec. The time is not predicted as well as the location.

Next we shifted the position of the transducer slightly to see how the signal changed. We found that the maximum signal occurred when point A was 1.33 inches from the step. In this case, the time was 23.04 microsec and the voltage was 2.78 volts. The oscilloscope trace is shown in Figure 9c. These results show that the transducer should be moved back from the normal position (point A at 1.20 inches). However, the maximum signal occurred when point A was 1.33 inches from the step, rather than 1.54 as predicted by the theory. A

TABLE 2. Run 2: Motion of Transducer Across Step Discontinuity

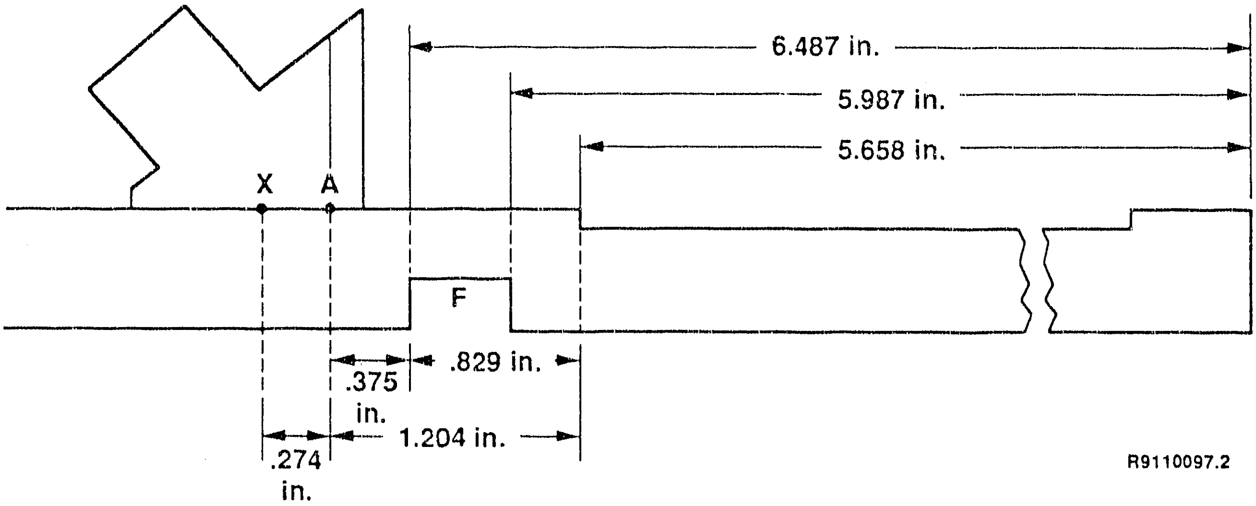
(change in location of point P and round-trip time)

dr	0.017453	changes angle from degrees to radians
rd	57.29578	changes angle from radians to degrees
thgel	27.13	angle theta in gel
hx	0.701	distance HX on transducer
stpht	0.072	step height
hf	1.16	total length HF of transducer
thkst	0.657	thickness of steel plate

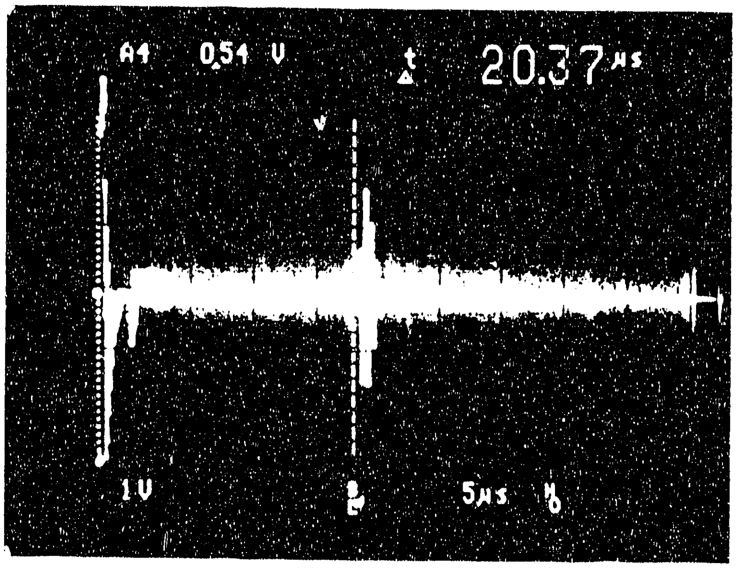
EF	Φ	σ	H_x	P_x	Total Time
Initial Position	0		0	1.358	24.00633
0	3.561651	52.33405	0.002238	1.578827	26.00245
0.1	3.898471	53.07408	0.102450	1.704706	26.39284
0.184	4.234927	53.82356	0.186662	1.815744	26.79964
0.2	4.305715	53.98261	0.202706	1.837594	26.88751
0.3	4.808081	55.12579	0.303022	1.980864	27.53596
0.4	5.443357	56.61074	0.403422	2.140533	28.42609
Final Position			0.5	1.93	25.63550

slightly smaller pulse was observed when point A was at the predicted 1.54 inches from the step.

In the final step of this experiment we placed a thickness of wood equal to 0.144 inches under the tip of the transducer so that EF = 0.1 inches and moved it until we found a maximum signal. This occurred when point A was 2.101 inches from the step. The time was 32.06 microsec and the voltage was 0.710 volts. This oscilloscope trace is shown in Figure 9d. We had not carried out the calculation of the expected position for this situation before the experiment was carried out. Table 3 now shows the theoretical calculation. When the transducer is tilted, it should be moved back by a distance of 2.14 minus 1.36, or 0.78 inches, from the normal position. Since the normal position of point A (see above) is 1.204 inches from the step, then, in this tilted position, the transducer should be located at 1.204 + 0.78 = 1.98 in from the step. The calculations also show that the time should be 32.8 microsec. There is excellent agreement between theory and experiment for the position and time: 2.101 inches experimentally versus the predicted 1.98 in and 32.06 microsec experimentally versus the expected 32.8 microsec.

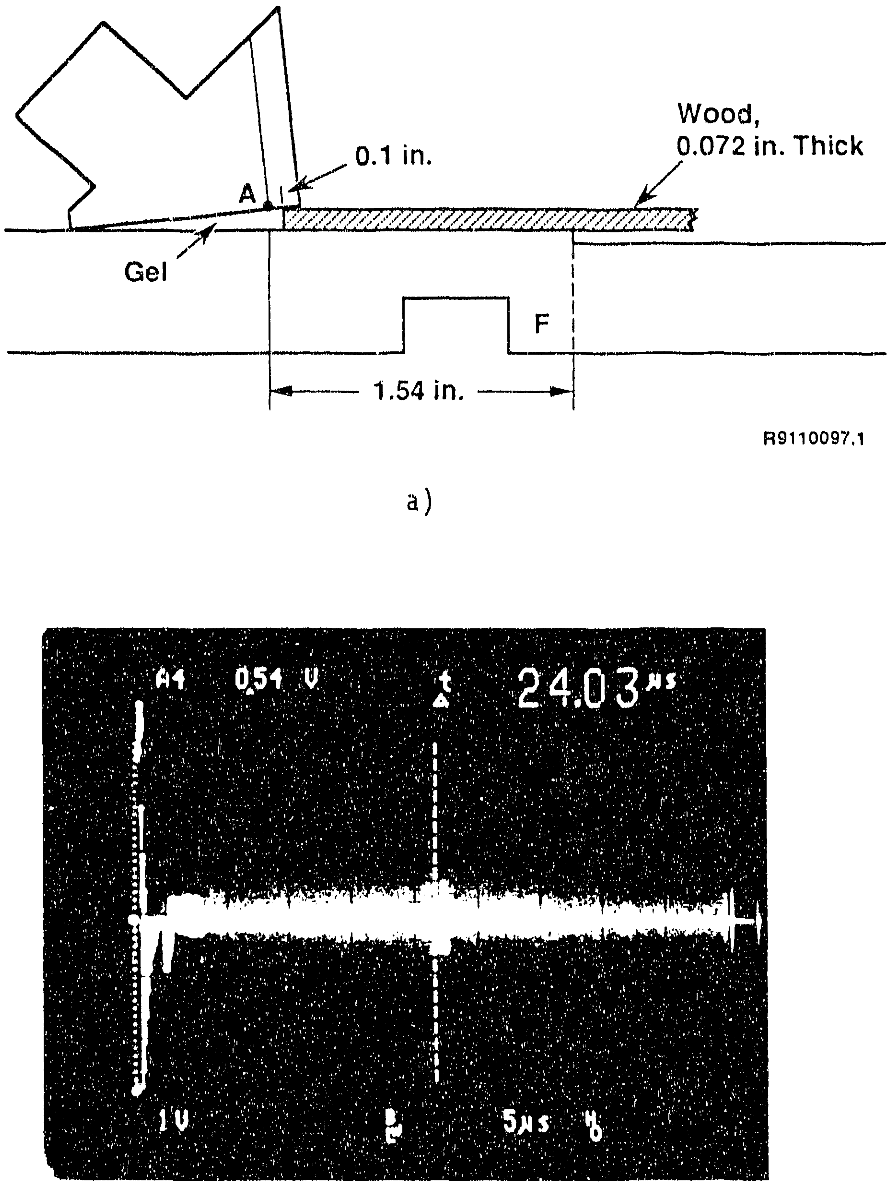


a)



b)

FIGURE 8. a) Normal Orientation of Transducer Positioned to Obtain Signal of Maximum Strength from Left Side of Notch F; b) Resulting Oscilloscope Trace Yielding a Time of 20.37 Microsec and a Voltage of 3.34 Volts.



a)

b)

FIGURE 9. a) Tilted Orientation of Transducer Positioned at 1.54 Inches from Step, which was the Location Calculated to Give Signal of Maximum Strength; b) Resulting Oscilloscope Trace Yielding a Time of 24.03 Microsec and a Voltage of 1.12 Volts.

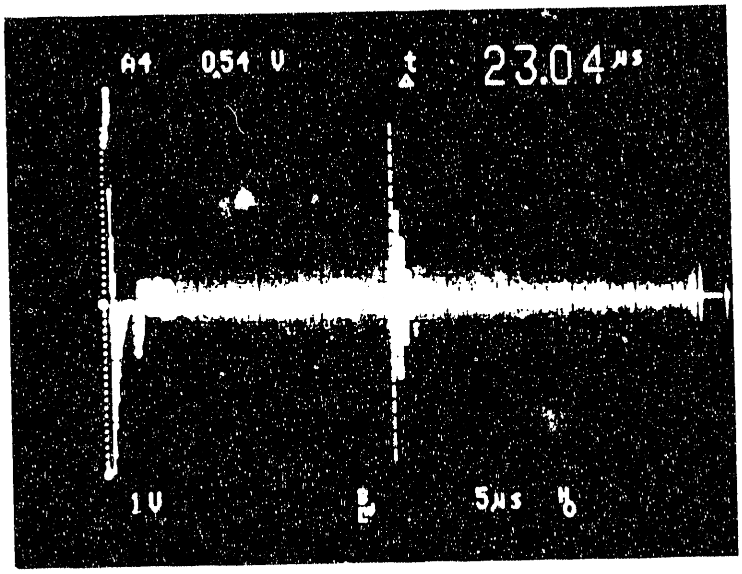


FIGURE 9c. Oscilloscope Trace Obtained when Transducer was Positioned at 1.33 Inches from Step. This location was obtained by shifting the transducer until a signal of maximum strength was obtained. This yielded time of 23.04 microsec and a voltage of 2.78 volts.

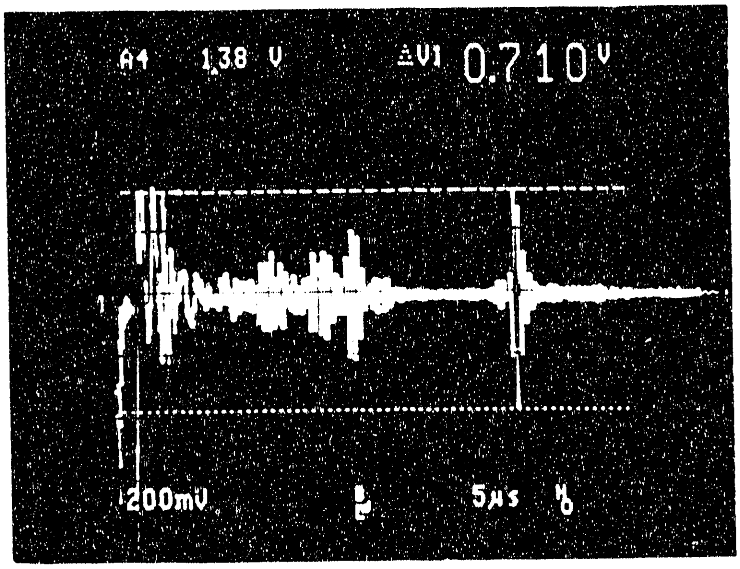


FIGURE 9d. Oscilloscope Trace Obtained when Transducer was Tilted by Placing a Piece of Wood 0.144 Inches under the Tip. The transducer was shifted to find the signal of maximum strength and yielded time of 32.06 microsec and a voltage of 0.710 volts.

TABLE 3. Run 3: Motion of Transducer Across Step Discontinuity

(change in location of point P and round-trip time)

dr	0.017453	changes angle from degrees to radians
rd	57.29578	changes angle from radians to degrees
thgel	27.13	angle theta in gel
hx	0.701	distance HX on transducer
stpht	0.144	step height
hf	1.16	total length HF of transducer
thkst	0.657	thickness of steel plate

EF	Φ	σ	H_X	P_X	Total Time
Initial Position	0		0	1.358	24.00633
0	7.137159	60.83456	0.008980	1.941102	31.32283
0.1	7.815138	62.66297	0.109836	2.141813	32.79095
0.184	8.493216	64.59567	0.194692	2.345537	34.52864
0.2	8.635991	65.01813	0.210872	2.389829	34.93828
0.3	9.650445	68.21433	0.312155	2.734863	38.47459
0.4	10.93663	72.98465	0.413785	3.353051	45.91869
Final Position			0.5	2.002	27.26467

5.0 CONCLUSIONS

We find that the experimental results using notch B and F are generally in very good agreement with the theoretical predictions. These results show that the refraction that occurs at the plastic-gel and the gel-steel interfaces play an important role in determining where the pulses will be observed and influence the round-trip time as well.

Some of the earlier conclusions can be summarized as follows:

1. The effect of the rotation results in some areas not being scanned very well by the central ray of the transducer.

2. The response from a crack might suggest that the crack is positioned farther away than it actually is. Hence, it will be harder to distinguish it from the root signal.

3. For a crack tip the tendency would be to position the crack farther away and systematically undersize the crack.

Appendix B: Ultrasonic Wave Propagation Through an Interface with a Step Discontinuity

PART IV.

ULTRASONIC WAVE PROPAGATION THROUGH AN INTERFACE WITH A STEP DISCONTINUITY

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INTRODUCTION

The condition of an interface through which an ultrasonic wave passes as it enters a material is an important factor in ultrasonic nondestructive evaluation. Most modeling studies of ultrasonic inspection assume that this interface is smooth. However, in real life this may not be the case. In the case of nuclear reactor components, factors such as weld overlay, claddings, grinding and diametrical shrink can give part surfaces a wavy, corrugated or abruptly stepped topography. M. S. Good [1] has provided some estimate of what surface conditions exist in nuclear reactor components, with some examples being illustrated in Fig. 1. These irregular surfaces can severely distort or redirect the ultrasonic beam, leading to false indications of size and location of defects.

The object of this study is to develop a model to predict the distortion of ultrasonic beams passing through rough, irregular interfaces. Such a model could be used to investigate the inspectability of particular components, e.g. to decide if a rough surface needs more smoothing for an accurate ultrasonic inspection. In this paper, the physical assumptions underlying the model are reviewed. The results of preliminary validation tests are reported. In those tests, the model is evaluated for a few test cases involving a step discontinuity and those results are compared to experiment.

THEORETICAL BACKGROUND

General Overview of the Model

The model utilizes a hybrid technique, which separately considers three stages for propagation of the ultrasonic beam through a rough interface. From the transducer to the rough surface, the propagation phenomena, within the Fresnel approximation, is fully included based on the Gauss-Hermite beam model. From the interface to an imaginary transmitted plane in the immediate vicinity of the interface, a ray tracing technique is used to account for the aberrations induced on the beam in that vicinity. In this region, the effects of beam spread due to diffraction are neglected. From the transmitted plane and beyond, the Gauss-Hermite model is again applied.

Gauss-Hermite Beam Model

The Gauss-Hermite beam model which has been developed over the past several years can be used to describe ultrasonic beam propagation in fluids [2] and isotropic [3] and anisotropic solid media [4-6]. In this Gauss-Hermite model, the beam is represented as a superposition of bound basis functions, each of which spreads during propagation in accordance with the principles of diffraction. The behavior of each of these basis functions is derived by representing it as an angular spectrum of plane waves and then employing the Fresnel approximation to allow the integrals over spatial frequency to be evaluated analytically. For the case of anisotropic media, certain parameters in the theory, which determine beam skew and divergence, can be directly related to the slowness surfaces. The net results is that the radiation of an ultrasonic source propagating in the z-direction can be represented as

$$u(x,y,z) = \sum_{m=0}^{\infty} \sum_{n=0}^{\infty} C_{mn}\, u_{mn}(x,y,z) \qquad (1)$$

Here the u_{mn} are the Gauss-Hermite eigenfunctions, whose transverse variations have the form of a complex Gaussian exponential multiplied by a Hermite polynomial, with amplitude, phase and width parameters depending only on the axial coordinate. The Gauss-Hermite complex constant coefficients, C_{mn} are computed by using the orthogonality property of the Gauss-Hermite functions and knowledge of radiation pattern at the source ($z = 0$). Once these coefficients are known, the displacement amplitude can be computed for any point (x, y, z). Also, by using equation (1), the normal vector to the phase front is computed by finding the gradient vector of the phase.

Ray Tracing

A ray tracing model is used to calculate the effect of the interface on the beam, following an approach described previously [7]. As noted above, by using the Gauss-Hermite model, the vector normal to the phase front and the displacement for the incident wave are computed at each point on the interface. These normal vectors to the phase fronts define rays which pass through the interface (see Figure 2) and intersect a transmitted plane. This transmitted plane is perpendicular to the central ray and selected to lie close to the interface. To calculate the field amplitude on the transmitted plane, the rays are considered to define flux tubes, and conservation of energy is applied from the interface to the transmission plane. Thus we require

$$Z_1 \iint_{\text{interface plane}} u_i^2\,(x,\, y)\, dA = Z_2 \iint_{\text{transmission plane}} T^2\, u_t^2\,(x,\, y)\, dA \qquad (2)$$

where u_I is the displacement of the incident wave, and u_t is the displacement amplitude on the transmission plane, T is the interface transmission coefficient, and Z is the acoustic impedance. T is assumed to have a spatial variation consistent with the angles of incidence and refraction of the involved rays.

After reconstruction of the beam pattern on the transmitted plane, new Gauss-Hermite coefficients C_{mn} are computed. Then the displacement at any point beyond the transmitted plane can be computed using equation (1).

Limitations of the Model

The model involves two approximations. The Fresnel approximation is inherent everywhere because of the use of the Gauss-Hermite model. This is generally not a severe limitation and has been discussed elsewhere [2-6]. In addition, use of ray tracing ignores any diffraction related beam spread between the interface and the transmitted plane. We have not fully determined the errors involved in the use of this approximation. However, we speculate that the rate of spatial variation of the surface profile should be relatively low. For very high rates of spatial variation, the ray tracing will predict excessive refraction, and the constructed beam on the transmitted plane becomes meaningless. Figure 3 shows an illustration of the rough surfaces with low and high rates of spatial variation. The ratio of ultrasonic wavelength to the spatial periods of interface roughness clearly should be small to use this approximation.

Another consequence of the neglect of beam spread is an error in width. Thus, the transmitted plane must be chosen as close as possible to the interface, so that ray tracing is done over as small of a distance as possible.

EXPERIMENTAL PROCEDURE

The sample used in these experiments was a stainless steel block with three different step sizes on it. The heights of the steps were 0.01, 0.03, and 0.06 inches (.025, .076 and .152 cm). The ultrasonic source used in all the experiments was a 0.5 inch, (1.27 cm) diameter planar transducer with a 2 MHz center frequency. For normal incidence, the transducer, which was excited by a tone burst, was placed 6 cm directly above each step. For oblique incidence, it was inclined at an angle such that a 45° L-wave was generated in the solid and the central ray passed through the top of the step. At the bottom of the sample, a microprobe was used to receive the distorted signal passed through the step. The microprobe was scanned on a square area of 2 inch (5 cm) sides. Figure 4 shows the configuration.

The transmitted plane for normal incidence was assumed to be 0.2 cm below the top of each step. In the case of oblique incidence, the transmitted plane was inclined at an angle of 45 degrees, passing through the top of each step. In this way minimum path lengths are used. Figure 5 shows the position as the transmitted planes for each case.

RESULTS AND DISCUSSION

Normal Incidence

In this case the transducer was placed directly above the step. Although the microprobe performed a c-scan, to compare the experimental and theoretical results, a 2-D graph of experiment and theory is presented. Figure 6 shows the comparison between theory and experiment for all the three steps.

As can be seen from the graphs, there is good agreement between experiment and theory near the center of the beam, with the theory predicting the general shape of the beam profile quite well. However, the difference between the two increases as one moves further away from the central ray. This is what was expected for two reasons. First, the model accuracy decreases away from the central ray due to the paraxial approximation used in the theory. Second, the ray tracing which does not consider the diffraction of the beam can cause errors in the beam's width. It must be noted that most of the energy is concentrated near the center of the beam and that this is the energy usually involved in flaw detection experiments. Errors in predictions of the side lobe structure may not affect the signal reflected from cracks or other flaws during ultrasonic inspections. The deviations between theory and experiment may also be due to errors in experimental measurements or to the fact that the transducer used in the measurements did not radiate exactly as a piston source, as assumed in the theory. These possible sources of error are being studied.

It is quite encouraging that the theory does a good job of predicting the constructive and destructive interference of the beam due to presence of steps. The size of the step and the frequency control this interference. A 0.03 inch (0.07 cm) step size, there is almost a constructive interference, but as the step size increases, it changes to a destructive interference. The possibility of such interferences must be considered in the ultrasonic examinations. As it is shown in the graphs, the inspectibility is severely reduced under certain combinations of frequency and step size.

Oblique Incidence

The oblique incident was selected such that the transmitted wave isrefracted at 45 degrees from the normal. The comparison of theory and experiment for oblique incidence are shown in Figure 7. The origin of the abscissa is relative in these plots, with the central ray passing through the lobe that is furthest to the right in each case. The results for a 0.01 inch (0.025 cm) step shows a good agreement within the main lobe, but for the other two step sizes there are considerable differences between experimental measurements and theoretical predictions. The sources of these deviations are still under study. The disagreements could have been originated from deficiencies in either the experiments or the theoretical model. Experimentally, it is much harder to adjust the transducer to a predetermined position in the oblique incidence configuration than for normal incidence, and there is the possibiity of mode converted transverse waves being detected as well as the longitudinal waves. On the theoretical side, ray tracing is more involved in the oblique incidence case than in the normal incidence case. Further studies are in process to determine whether these disagreements represent fundamental limitations on the theory or initial errors in our analytical or experimental work.

CONCLUSION

This study clearly shows the importance of the surface condition in ultrasonic inspection of materials. It also shows that, in the normal incidence case, the theoretical predictions of the beam profile closely matched the experimental data. The disagreements were mostly in side lobes and away from the central ray. This could have been caused by a) the Fresnel approximation implicit in the Gauss-Hermite model or b) the ray tracing that does not consider the beam spread near the interface.

In the oblique incidence experiments, the predictions were not in as good agreement with experiment. The results were qualitatively similar but exhibited some quantitative differences. The problems could have been due to a) the ray tracing, b) the operation of the microprobe which is primarily sensitive to the normal component of the displacement, or c) errors in experimental setup and procedures. The oblique incidence case will be studied in more detail.

ACKNOWLEDGMENTS

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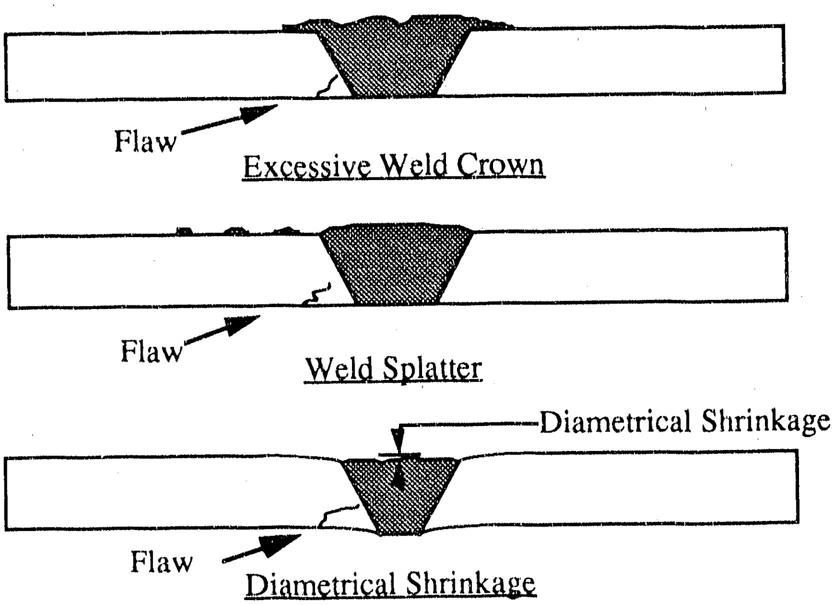


Figure 1. Irregular surface conditions in nuclear reactor components.

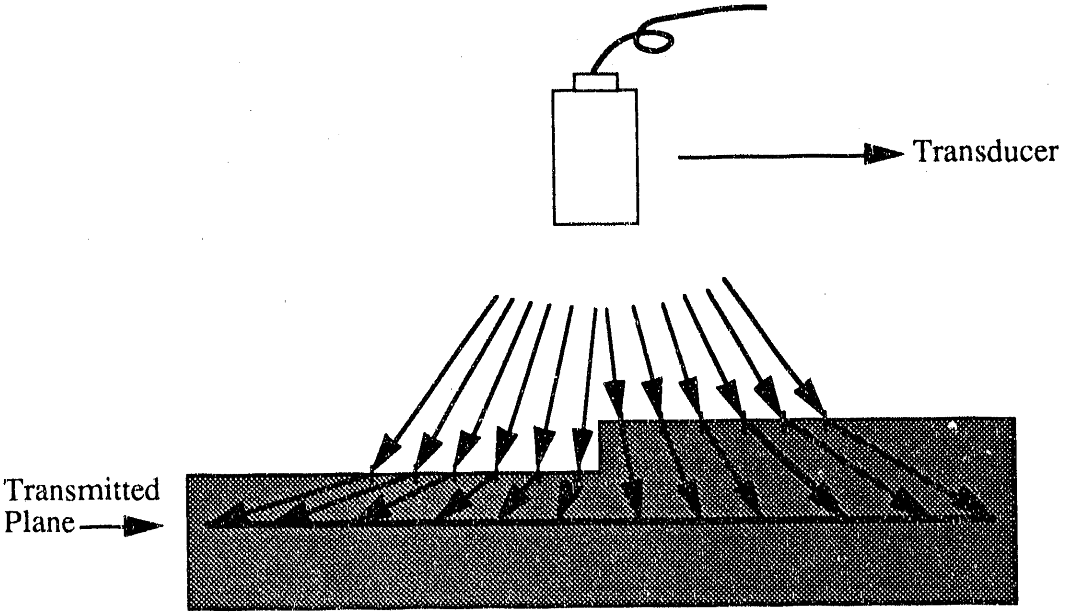


Figure 2. Schematic drawing of rays transmitting inside the sample and propagating toward transmitted plane.

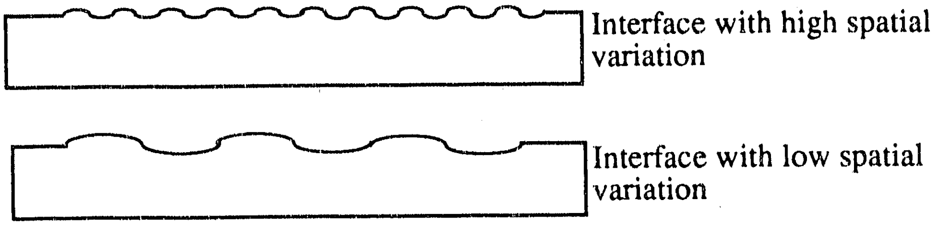


Figure 3. Surface conditions that affect the validity of the model.

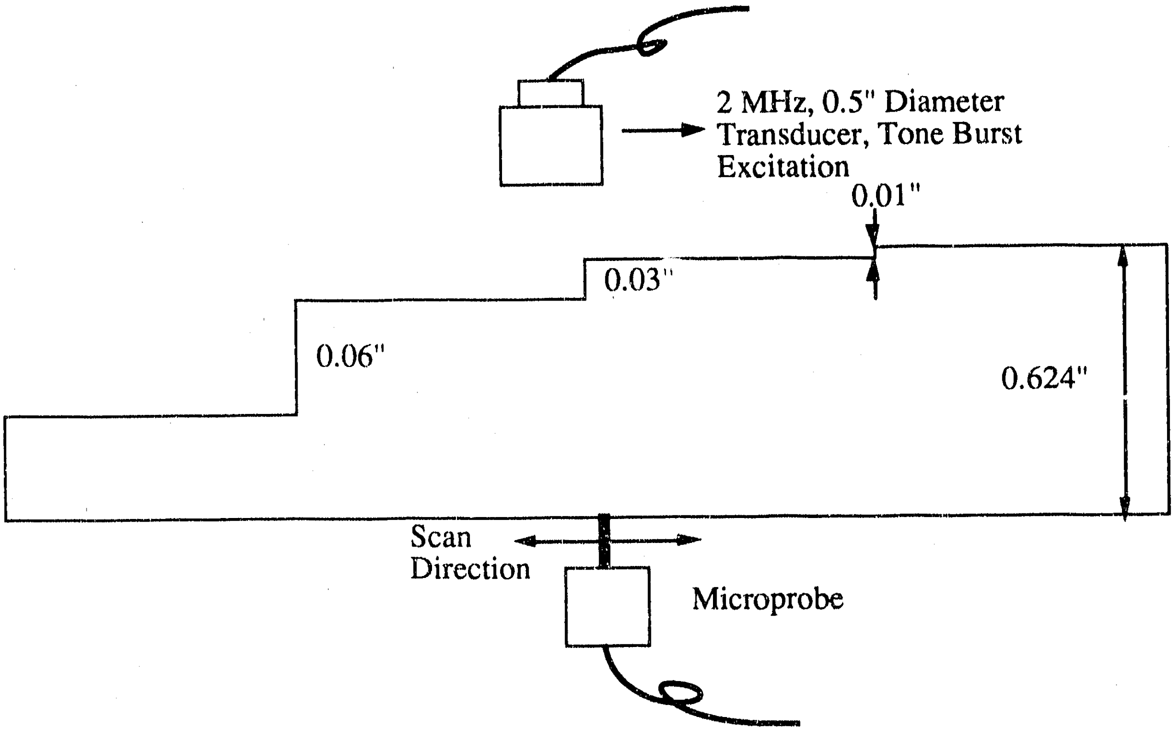
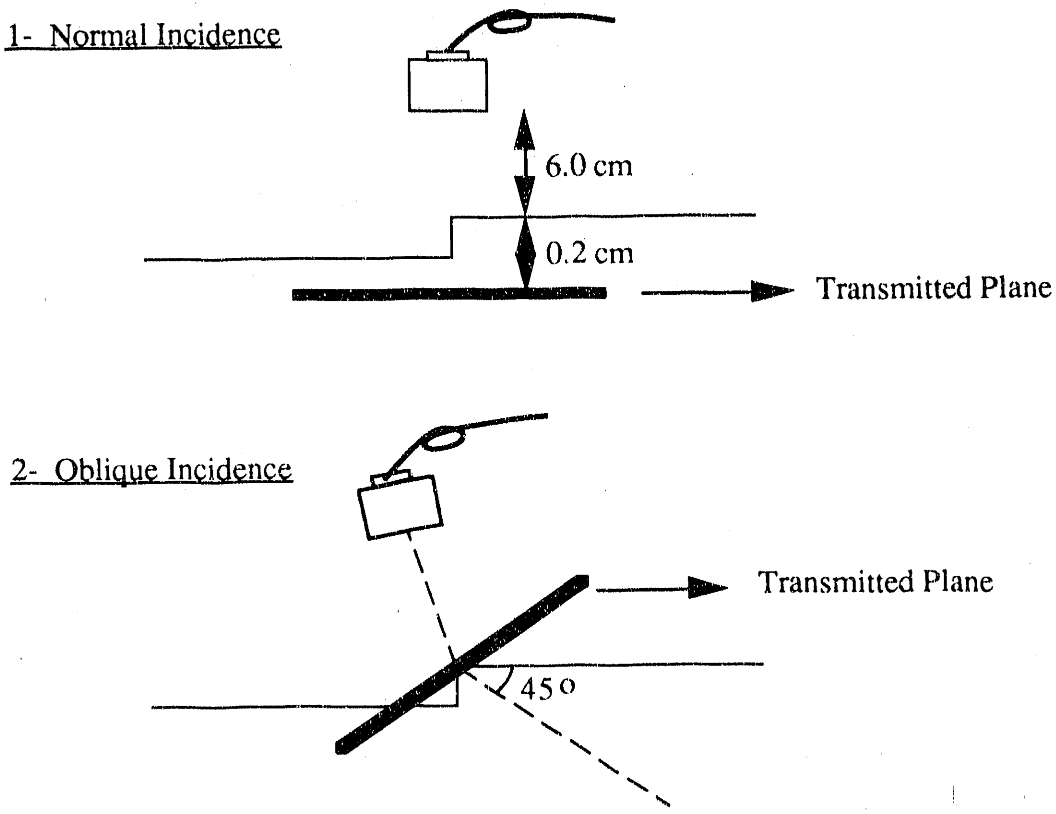


Figure 4. Experimental setup.

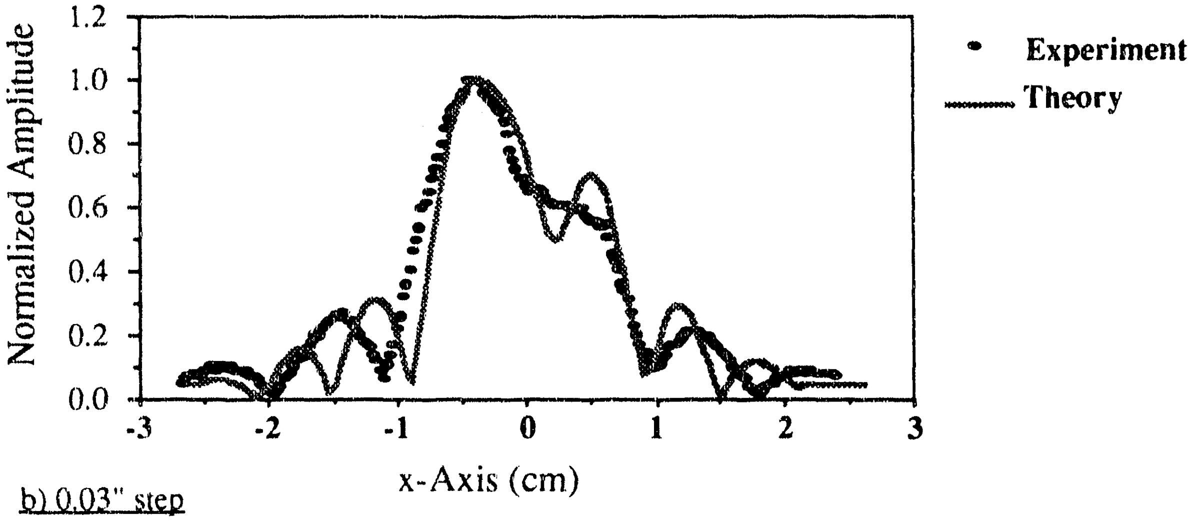


1- Normal Incidence

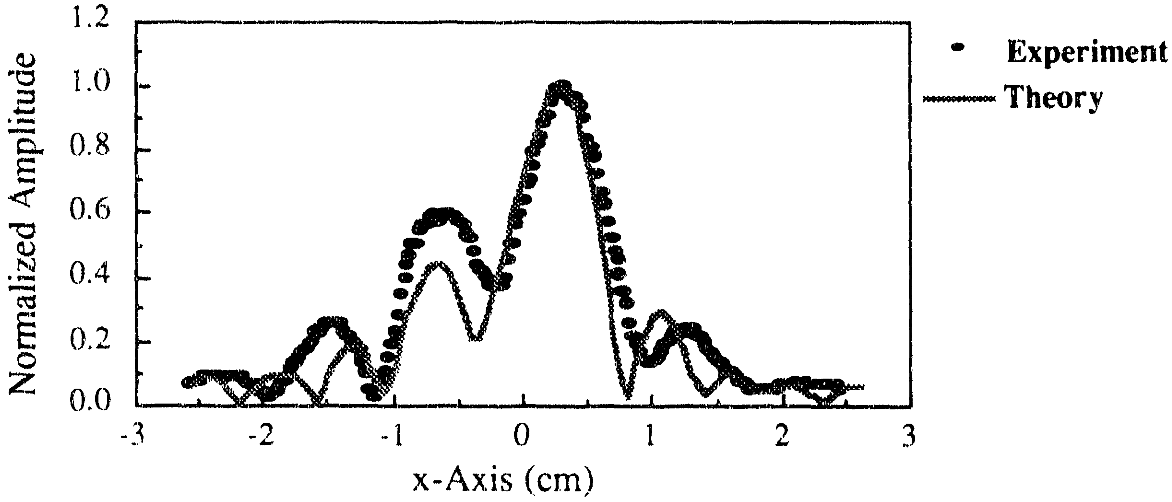
2- Oblique Incidence

Figure 5. The position of transmitted plane in normal and oblique incidence case.

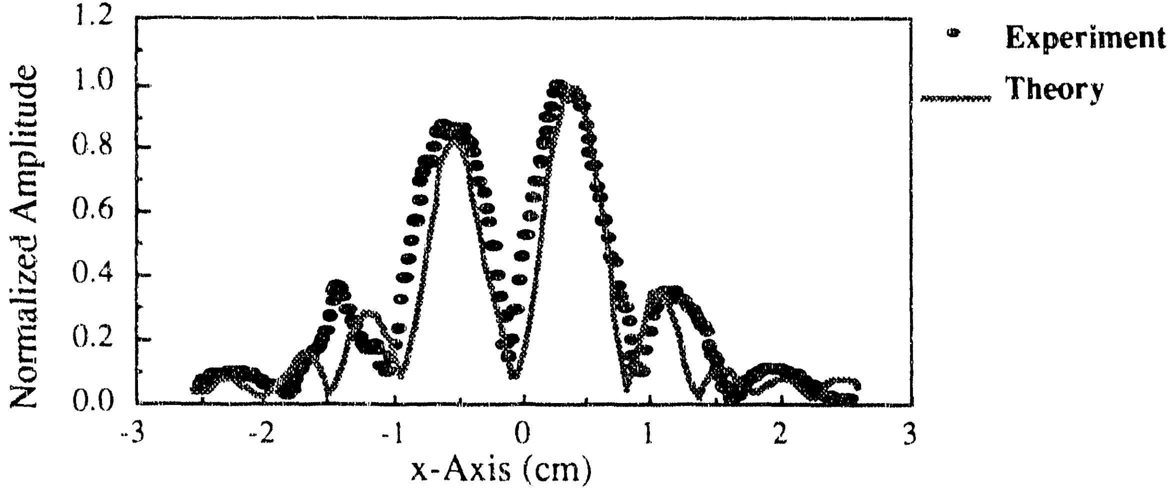
a) 0.01" step



b) 0.03" step



c) 0.06" step



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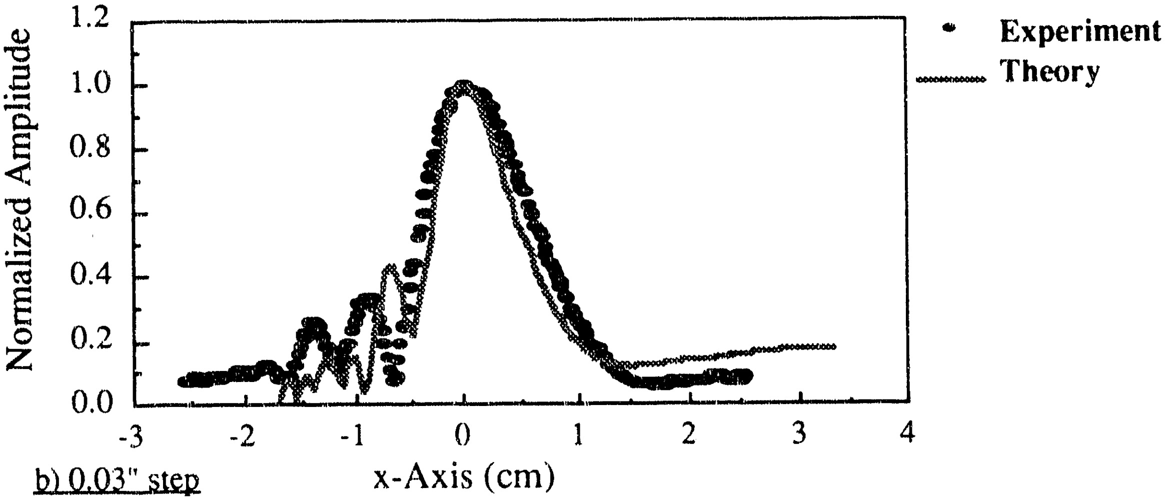

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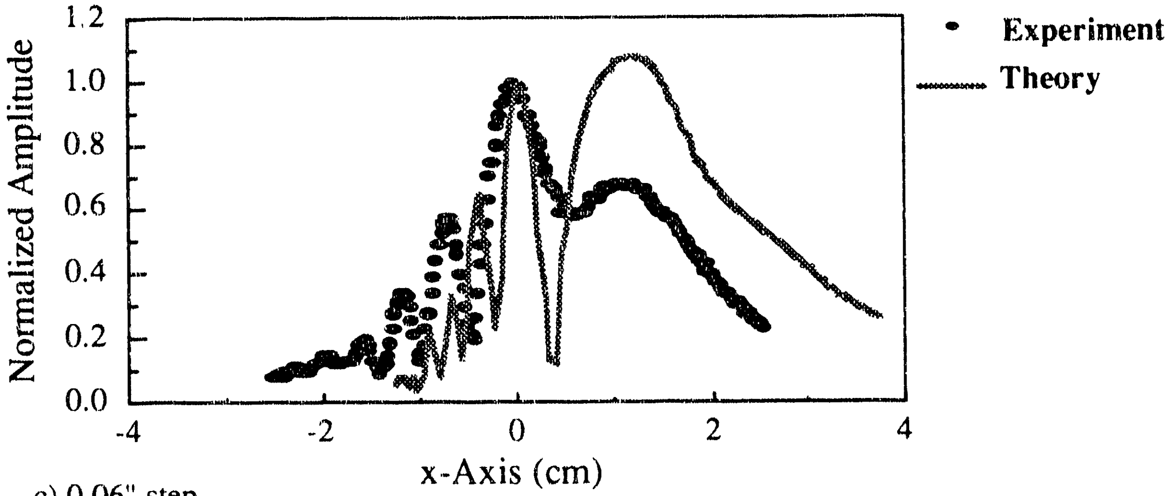

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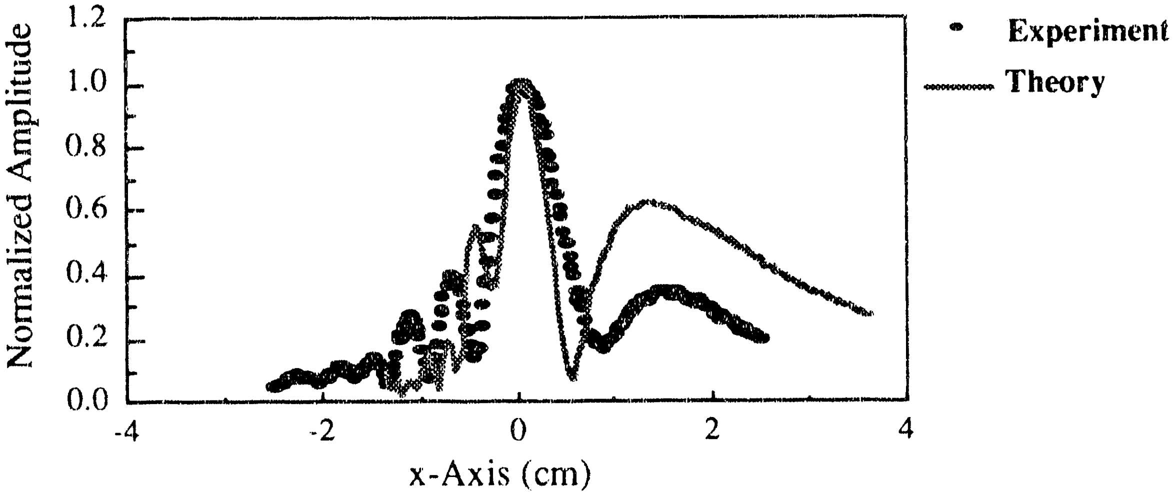

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11. ABSTRACT *(200 words or less)*

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 April 1991 through September 1991.

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