REGIONAL DEPOSITORY DR-487 JAN 7 1992
U.S. GOVERNMENT INFORMATION DIVISION 92-012
OKLAHOMA DEPARTMENT OF LIBRARIES
NUREG/CR-5645
PNL-7479

Acoustic Emission/Flaw Relationships for Inservice Monitoring of LWRs

WITHDRAWN FROM
OKLAHOMA DEPARTMENT OF LI

WITHDRAWN FROM
OKLAHOMA DEPARTMENT OF LIBRARIES

Prepared by
P. H. Hutton, R. J. Kurtz, M. A. Friesel, J. R. Skorpik, J. F. Dawson

Pacific Northwest Laboratory
Operated by
Battelle Memorial Institute

WITHDRAWN FROM
PARTMENT OF LIBRARIES

Prepared for
U.S. Nuclear Regulatory Commission

metadc1202751

AVAILABILITY NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 2120 L Street, NW., Lower Level, Washington, DC 20555
2. The Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013-7082
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC bulletins, circulars, information notices, inspection and investigation notices; licensee event reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, international agreement reports, grant publications, and NRC booklets and brochures. Also available are regulatory guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG-series reports and technical reports prepared by other Federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions. *Federal Register* notices, Federal and State legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Office of Administration, Distribution and Mail Services Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, for use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

DISCLAIMER NOTICE

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, or any of their employees, makes any warranty, expressed or implied, or assumes any legal liability of responsibility for any third party's use, or the results of such use, of any information, apparatus, product or process disclosed in this report, or represents that its use by such third party would not infringe privately owned rights.

NUREG/CR-5645
PNL-7479
GS, R5

Acoustic Emission/Flaw Relationships for Inservice Monitoring of LWRs

Manuscript Completed: September 1991
Date Published: October 1991

Prepared by
P. H. Hutton, R. J. Kurtz, M. A. Friesel, J. R. Skorpik, J. F. Dawson

Pacific Northwest Laboratory
Richland, WA 99352

Prepared for
Division of Engineering
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555
NRC FINs B2088, B2913

ABSTRACT

The program concerning Acoustic Emission/Flaw Relationships for Inservice Monitoring of LWRs was initiated in FY76 with the objective of validating the application of acoustic emission (AE) to monitor nuclear reactor pressure-containing components during operation to detect cracking. The program has been supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research.¹ Research and development has been performed by Pacific Northwest Laboratory, operated for the Department of Energy by Battelle Memorial Institute. The program has shown the feasibility of continuous, on-line AE monitoring to detect crack growth and produced validated methods for applying the technology. Included are relationships for estimating flaw severity from AE data and field applications at Watts Bar Unit 1 Reactor, Limerick Unit 1 Reactor, and the High Flux Isotope Reactor. This report discusses the program scope and organization, the three program phases and the results obtained, standard and code activities, and instrumentation and software developed under this program.

¹Work at Watts Bar Unit 1 was supported in part by the Tennessee Valley Authority.

EXECUTIVE SUMMARY

INTRODUCTION

Acoustic emission (AE) operates on the basis of detecting energy released as a crack grows in a solid material by detecting the resulting stress waves propagated through the material. The concept of applying AE technology as a means of continuously monitoring nuclear reactor systems to detect flaw growth has received a wide variety of support and condemnation over the past 20 years. Initially, it suffered from over enthusiasm on the part of proponents, which resulted in some disappointing application efforts. It also resulted in an unsuccessful and premature effort to incorporate AE methods into the U.S. ASME Code. A very pessimistic assessment of AE techniques for continuous monitoring arose from these activities. For the most part, the points raised were well thought out, and they helped clarify what had to be accomplished to make AE technology useful for continuous monitoring. A carefully planned research program to accomplish the needed technology development for application to reactor systems was conceived in 1976. The program has been supported by the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research with the research performed at Pacific Northwest Laboratory (PNL), operated by Battelle Memorial Institute.

The objective of the NRC/PNL AE program was to validate AE technology for continuous monitoring of reactor pressure boundaries to detect crack initiation and growth. The benefits expected from the completed work are:

- Provide continuous monitoring for early detection of cracking in the primary pressure boundaries for early detection and location of active flaw sites.
- Provide an alternate method for surveillance of identified flaws to detect growth during operation and thereby enhance plant safety.
- Provide a leak detection method that is more sensitive and timely than current leak detection methods.
- Could reduce the need for augmented manual inspection of service-sensitive piping which reduces personnel exposure consistent with ALARA concepts.

AE TECHNOLOGY DEVELOPED

The NRC/PNL AE program contained three primary phases designed to develop the needed technology elements in an organized sequence. These phases were:

1. Develop initial AE/flaw relationships by laboratory specimen testing to identify crack growth AE signals and to use crack growth AE data to estimate flaw severity.

2. Evaluate and refine the technology developed in the laboratory by monitoring pipe bend tests, heavy section steel technology vessel tests, and fatigue testing of a heavy section vessel under simulated reactor conditions.
3. Validate continuous AE monitoring on a nuclear power reactor system.

Phase 1 produced an AE rate/crack growth rate relationship for fatigue crack growth in A533B steel in support of flaw evaluation. Crack growth rates in the laboratory work ranged from about 3.0×10^{-7} to 4×10^{-4} in./second, and test temperatures used were room temperature and 550°F. A statistical pattern recognition method for identification of crack growth AE signals and an instrumentation concept were also developed.

NRC/PNL collaborated with the German Materialprüfungsanstalt (MPA) Laboratory in a one-year long test of an intermediate scale pressure vessel called the ZB-1 vessel to accomplish Phase 2. The test was performed under non-laboratory "field conditions." The vessel was 5.5-ft. outside diameter by 12-ft. long with a 4.75-in. wall thickness and fabricated from A508 steel. The vessel included a 30 x 60 x 4.75 in. insert of A533B steel. Fatigue loading was supplied by cyclic internal pressurization with water. Test temperatures of 150 and 550°F were used, and crack growth rates ranged from 1×10^{-7} to 5×10^{-6} inches per second. Test results demonstrated the ability to detect and interpret AE from growth of the known flaws and also from an unanticipated crack that developed adjacent to a weld.

Phase 3 was started at the Tennessee Valley Authority (TVA) Watts Bar Unit 1 nuclear reactor with TVA sharing the cost. Selected locations have been monitored during preservice tests at Watts Bar -- cold hydro and hot functional. AE equipment is in place to monitor the same locations at Watts Bar during reactor startup and operation. A current activity in Phase 3 is to monitor a flaw indication in a recirculation nozzle-to-safe end weld at Philadelphia Electric Company's Limerick Unit 1 reactor during reactor operation. This work is supported by Philadelphia Electric Company and NRC Research.

With this background, the program accomplishments will be discussed under the following topics:

- Instrumentation
- Flaw detection and location
- Noise interference
- AE signal identification
- Flaw severity estimation
- Standards and Code development.

Instrumentation

A fundamental requirement is an AE instrument system capable of operating continuously over a long period (one-year minimum), capable of withstanding the reactor environment, and capable of mass data storage.

The AE monitor system used in the ZB-1 vessel test performed with very limited problems during the one-year duration of the test and has continued in use on laboratory tests since that time. An engineering prototype system based on the ZB-1 experience is in use at Limerick Unit 1 reactor.

The instrument is a prototypic data acquisition/recording subsystem, which receives data from the sensors, is composed of eight channels divided into two arrays. Data collected from each array consists of delta time of signal arrival, RMS signal level, signal peak time, pulse height, sensor hits, and a flag indicating whether the signal is crack-growth AE or noise. Delta-times are used for source location and are measured with one microsecond resolution. RMS is measured on each channel for leak detection analysis. Peak time and pulse height, measured on one channel of each array, provide an indication of the acoustic signal characteristics. Sensor hit totals give a measure of the intensity of AE activity and can also be used to judge sensor functional condition. The signal identification flags are generated by a pattern recognition module, and they are used to identify the data originating from crack growth. All data are recorded on a 67-Mbyte cartridge tape for long-term storage. This data storage media has proven to be very effective in field testing. The data analysis is designed to be performed on a standard computer such as a DEC 11/23 or 11/73.

A key element of the AE monitor system is a high-temperature sensor. Two problems have been experienced in attempts to develop AE sensors to operate directly in an environment of 600°F or higher: 1) the high-temperature piezoelectric materials tend to have poor sensitivity, and/or 2) the sensors are very costly. Long-term acoustic coupling of these sensors to the structure surface that is acceptable for reactor use can also be a problem. These problems have been overcome by using metal waveguides with a sensing crystal (PZT5A) and a preamplifier mounted on the outer end. It is pressure coupled to the surface with a force of 30 pounds, which produces an interface pressure of 15,000 psi at the pointed waveguide tip. The waveguides are nominally 0.130 inches in diameter. These sensors have been used effectively on both the ZB-1 vessel test and on hot functional testing at Watts Bar. During tests under a separate program, this type of sensor assembly has been exposed to gamma radiation in the 1 to 1.3 MeV range at 5×10^4 R/hr for 120 days for a cumulative dose of 14×10^7 R with no evident degradation. The waveguide AE sensors being used at Limerick reactor are nine feet long to place the crystal/preamp head outside the high-temperature region, and they still show good sensitivity.

Summarizing, an instrument system capable of long-term monitoring under reactor conditions with a mass data storage feature has been demonstrated.

Flaw Detection and Location

Given the needed instrument system, the next consideration is the effectiveness of flaw detection and location. In the ZB-1 vessel test, fatigue crack growth from laboratory-produced flaws was detected consistently during both low-temperature (150°F) testing and during high-temperature (550°F) testing. The most significant example of fatigue crack detection during the ZB-1 vessel test, however, involved an unexpected crack along an

insert weld. The presence of a growing fatigue crack in the weld region was identified early in high-temperature testing by the AE data and called out as such. This was well ahead of confirmation of the flaw by ultrasonic inspection. Another significant aspect is the fact that the signals had to be received by all AE sensors in the array, some of which were 15 feet from the source, in order for the data to be accepted and located by the monitor. This provides a calibration of the range of AE detection for 500 kHz tuned waveguide sensors pressure coupled to the vessel surface.

It is important to note at this point that overpressure tests performed as part of the ZB-1 test indicated that AE would not be effective for detecting service-induced flaws in a reactor vessel by monitoring only during the normal overpressure testing performed on an inservice vessel (leak tests, inservice hydrostatic tests, and pressure tests) where the maximum overpressure is normally 1.1 x operating pressure. The overpressure tests performed on the ZB-1 vessel ranged from 1.0 x operating pressure in the first test to 1.36 x operating pressure in the sixth and last test. Even though one of the flaws was about 80% through-wall by the last test, the flaws produced very little if any AE under these conditions. Results indicated that a stress intensity factor of 70-80 ksi/ $\sqrt{\text{in.}}$ was needed to produce detectable AE from an existing crack. It must be made clear that these comments do not apply to preservice acceptance hydro tests or inservice overpressure tests required following major repair or replacement for detection of fabrication defects. The key difference in the latter case is that the flaws to be detected have not been previously stressed.

Stress corrosion cracking (SCC) is another cracking process of major interest. All of our information to date on detection of SCC by AE methods has come from laboratory specimens ranging from 4 to 24-inch diameter austenitic stainless steel pipe. SCC has been successfully detected by AE in all of these tests. Current work at Limerick Unit 1 addresses detection of SCC under reactor operating conditions.

In summary, effective detection and location of fatigue crack growth under simulated reactor conditions has been demonstrated. Fatigue cracking has been detected by AE prior to detection/confirmation by ultrasonic inspection. Stress corrosion cracking has been repeatedly detected in laboratory tests, and testing is in progress to validate these results on an operating reactor.

Noise Interference

One of the major concerns with continuous AE monitoring on an operating reactor has been the background noise from reactor coolant flow. Coolant flow noise for both PWRs and BWRs was characterized earlier. Required sensor characteristics to avoid interference from flow noise were developed using that information. Results from hot functional testing at Watts Bar Unit 1 showed that the tuned waveguide sensors do overcome the problem of noise interference. Three types of sensors were used on Watts Bar -- waveguide sensors tuned to 375 and 500 kHz plus a high-temperature surface-mounted commercial sensor. All were pressure coupled to the structure. During full cold flow at 150°F-400 psig, even the tuned sensors are ineffective; however,

increasing coolant temperature brings about a drastic reduction in noise in the high frequency range (>300 kHz) starting at about 250°F.

In summary, the coolant flow noise problem can be overcome using high-frequency tuned waveguide sensors. These same sensors have shown capability to detect AE at a distance of at least 10 feet.

AE Signal Identification

A key element of this program was to develop a method for separating AE signals produced by crack growth from acoustic signals produced by innocuous sources. Signal pattern recognition is the technique chosen for this purpose. The original approach using a statistical method involving signal spectral and power features to identify signals had limited success. The method that is proving to be much more effective is also much more straightforward. Visual examination of recorded waveforms from the ZB-1 vessel test showed evidence of a consistent three-pulse signal pattern present in practically all waveforms originating near known and suspected regions of crack growth. It was determined experimentally that the three pulses are different wave modes traveling at different group velocities. The separation of these wave modes takes place in the waveguide as a response to a delta function input pulse -- i.e., an AE signal. Incorporation of this feature into a pattern recognition approach has produced correct classification rates of about 95% when applied to recorded waveforms from the ZB-1 vessel test.

Summarizing, a pattern recognition approach utilizing characteristics of the metal waveguide sensors produces a correct classification rate of about 95% on ZB-1 vessel test data. The algorithm has been implemented for on-line monitoring.

Flaw Severity Estimates

Another important need is to develop a relationship to estimate flaw severity from AE data. The ZB-1 vessel test is the principal vehicle for evaluating a flaw severity relationship developed in the laboratory. To test the flaw severity relationship, AE data for each of three installed flaws (Flaws A, B, and C) was filtered with respect to source location, load position, and signal amplitude to minimize spurious data. Flaws A and B were ID flaws and Flaw C was an OD flaw. With the filtered information for each flaw, AE rate values were established for each of the cyclic loading steps. The AE rate in events/second was then used in conjunction with the lower bound flaw severity relations developed from laboratory data to estimate crack growth rates in microinches/second. These results were compared with measured crack growth rate taken from COD gages and post-test fracture surface examination. The results show some scatter, but are consistent with using the conservative lower bound relationships and are within the scatter of the laboratory data. When the relationship was applied to AE from the weld crack, it overpredicted the crack growth rate; however, the reason is understandable. The geometry of this crack was very complex; and as a result, it probably produced a large number of noise signals from crack face rubbing, which biased the data. Subsequent investigation indicates that this problem can be overcome using the current AE signal identification method.

In summary, the flaw growth rate relationship was effective in predicting crack growth rates in the ZB-1 test. Effective identification of crack-growth AE signals will significantly improve the consistency of crack-growth rate prediction.

Standards and Code Development

An American Society for Testing and Materials (ASTM) Standard E1139 titled "Continuous Monitoring of Acoustic Emission from Metal Pressure Boundaries" was approved in 1987. This standard provides guidelines for the application of continuous AE monitoring to detect cracking in metal pressure boundaries.

Development of a Code Case to be followed by a Nonmandatory Appendix to the American Society of Mechanical Engineers (ASME) Section XI Code to address continuous, inservice AE monitoring of reactor components was started in January 1987. A Code Case "Acoustic Emission for Successive Inspections, Section XI, Div. 1" has been fully approved.

CONCLUSIONS

The feasibility of on-line AE monitoring to detect, locate, and estimate the severity of crack growth in reactor components has been demonstrated in the course of the research program. The technology and hardware required for beneficial application has also been developed. The technology and equipment design has been made publicly available through this report and publication of the Code Case. In support of wide use of the AE technology developed, the following would be useful:

- Make the technology and hardware available commercially.
- Further demonstration of the effectiveness of inservice AE monitoring to detect crack growth on an operating reactor.

ACKNOWLEDGMENTS

This program was unique in the scope of activities, duration, and in the fact that it took a technology (acoustic emission) that was in its infancy and developed the application to on-line reactor monitoring including on-reactor demonstration, and standards and code development. Many experts around the world had concluded that the problems were too many and too difficult to achieve on-line AE monitoring. I am proud to have been a part of this program and wish to express my profound appreciation of the efforts of the many people without whom, the program could not have succeeded.

As in most programs of this scope, there are certain key people who deserve special notice.

Dr. Joe Muscara has been the Nuclear Regulatory Commission program manager for this program since its inception. His vision in helping formulate the program structure, technical support, confidence in ultimately achieving the end objectives, patience when results were slow in coming, and persuasive capability in obtaining continuing financial support, and defending the program went well beyond the average program manager role.

Jerry (G. J.) Posakony was in a PNL management role overseeing the program for its entire duration. Jerry's expert business and management guidance as well as his technical interest and support were invaluable.

Earl Schwenk (currently with Westinghouse Hanford Company) and Rick Kurtz, PNL played a sequential role in providing the key function of materials and fracture mechanics expertise. Earl was instrumental in the original formulation of the program and activities through Phase I. Rick assumed the lead role in that area at that time and continued through to the end of the program. In addition to developing AE-flaw growth relationships, he was very prominent in managing the Phase II vessel test in Germany.

Jack Dawson, PNL, has been my right hand in the AE area since the beginning of time. His dedication and innovative contributions to AE technology during the program are unsurpassed.

Jim Skorpik, PNL, has provided the expertise to develop all of the special AE instrumentation to meet program needs. These needs constantly increased in complexity and they were vital to obtaining good program results.

Prof. Karl Kussmaul, Staatliche Materialpruefungsanstalt (MPA), Germany, deserves special thanks for his vigorous and effective support of the ZB-1 vessel test conducted at Mannheim, West Germany. This test was the most significant single activity in the course of the program. Without Karl's support, it might never have been completed.

Another MPA staff member, Rainier Gillot, played a special role in the ZB-1 test. He was responsible for the mechanical test system preparation and conduct of that part of the test for the entire one-year test duration. Without his ingenuity in overcoming problems, again, the test might never have been completed.

Practicality dictates that from this point on, I express my appreciation by just listing names. The contributions of the following people were significant to the success of the program in many ways.

U. S. Nuclear Regulatory Commission

C. Z. Serpan

PNL - Battelle

W. A. Anderson
D. A. Buelt
D. A. Chaudier
L. S. Dake
P. G. Doctor
S. M. Faber
M. A. Friesel
J. C. Harris
K. E. Hass
R. T. Landsiedel
R. B. Melton
R. A. Pappas
C. Pavloff
C. B. Perry
J. C. Spanner
T. T. Taylor

Oak Ridge National Laboratory

P. Holtz
G. D. Whitman

MPA

H. Beyer

Grosskraftwerk Mannheim

F. Bauemuller
Schweitzer
Spoehrer

IzfP

Dr. G. Deuster
Prof. P. Hoeller

P. H. Hutton
Project Manager
PNL

CONTENTS

EXECUTIVE SUMMARY	v
1.0 INTRODUCTION	1-1
2.0 PROGRAM SCOPE AND ORGANIZATION	2-1
3.0 PHASE I - LABORATORY RESEARCH	3-1
3.1 APPROACH AND APPARATUS	3-1
3.1.1 Material and Test Specimens	3-1
3.1.2 Acoustic Emission Measurement	3-4
3.2 RESULTS	3-8
3.2.1 Tensile Tests	3-8
3.2.2 Fracture Tests	3-9
3.2.3 HSST Irradiated Fracture Toughness Tests	3-11
3.2.4 HSST Vessel Tests	3-14
3.2.5 Fatigue Crack Growth Tests	3-15
3.2.6 Stress Corrosion Cracking Tests	3-25
3.3 RELATIONSHIPS DEVELOPED	3-28
3.3.1 Monotonic Loading Conditions	3-30
3.3.2 Cyclic Loading Conditions	3-31
3.4 SUMMARY	3-34
4.0 PHASE II - EVALUATION BY VESSEL TESTING	4-1
4.1 TEST CRITERIA	4-1
4.2 TEST FACILITY SELECTION	4-2
4.3 TEST DESCRIPTION	4-4
4.4 MAJOR TEST RESULTS	4-6
4.4.1 Hydrostatic Test Results	4-6
4.4.2 Special Test Features	4-8

4.4.3	AE Monitoring at 150°F	4-10
4.4.4	AE Monitoring at 550°F	4-10
4.4.5	AE/Flaw Growth Interpretation	4-11
4.4.6	Crack Growth AE Signal Identification	4-12
4.5	CONCLUSIONS	4-13
5.0	PHASE III - VALIDATION ON REACTOR SYSTEMS	5-1
5.1	APPROACH	5-1
5.2	WATTS BAR UNIT 1 REACTOR	5-1
5.2.1	Objective	5-2
5.2.2	Cold Hydrostatic Test	5-2
5.2.3	Hot Functional	5-3
5.2.4	AE System Calibration On-Reactor with Simulated AE	5-4
5.2.5	Operational Monitoring	5-5
5.3	LIMERICK UNIT 1 REACTOR	5-6
5.3.1	Objective	5-6
5.3.2	Arrangements with PECO	5-6
5.3.3	Installation	5-6
5.3.4	Operational Monitoring	5-15
6.0	TECHNOLOGY TRANSFER	6-1
6.1	ASTM STANDARD	6-1
6.2	ASME SECTION XI APPENDIX AND CODE CASE	6-1
6.3	INFORMATION SEMINARS	6-2
6.4	PROTOTYPIC INSTRUMENT SYSTEM	6-2
6.4.1	Data Acquisition Subsystem	6-2
6.4.1.1	Signal Processors	6-5

6.4.1.2	Array Processors	6-5
6.4.1.3	Peak Time/Pulse Height	6-6
6.4.1.4	RMS	6-6
6.4.1.5	Signal Recognition Module	6-6
6.4.1.6	Parametrics	6-6
6.4.1.7	External Digital	6-6
6.4.1.8	Data Storage	6-7
6.4.1.9	Analysis Subsystem Interface	6-7
6.4.1.10	Front End Controller	6-7
6.4.1.11	Data Display	6-7
6.4.2	Data Analysis Subsystem	6-8
6.4.3	Signal Recognition Design	6-11
6.4.4	Cabling and Preamplifiers	6-13
6.4.4.1	Microamp	6-13
6.4.4.2	Midamp	6-14
6.4.4.3	Cabling	6-14
6.4.5	Waveguide AE Sensor	6-14
7.0	SUMMARY OF PROGRAM RESULTS AND BENEFITS TO NUCLEAR REGULATION	7-1
8.0	REFERENCES	8-1
APPENDIX A:	AE/FLAW CHARACTERIZATION ELECTRONICS DRAWINGS No. 0790-00 THROUGH 0790-06	A-1
APPENDIX B:	SOFTWARE	B-1

FIGURES

3.1	HSST V7-B Vessel AE Monitored During Testing - July 1977	3-5
3.2	Functional Description of the Phase 1 Laboratory Test AE Monitor System	3-5
3.3	EPROM Digital Storage Devices	3-6
3.4	Phase 1 Laboratory Test AE Monitor System with Interface Module and Digital Cassette Recorder	3-6
3.5	Shop-Made Surface Mount AE Sensors Used in Phase 1 Laboratory Specimen Tests	3-7
3.6	Acoustic Emission Event Count versus Stress Concentration Factor for Uniaxial Tensile Tests	3-9
3.7	Load and AE Event Count versus Time for Fracture Test of Specimen B2-1B	3-10
3.8	Load and AE Event Count versus COD for Fracture Test of Specimen 1-2A-4A	3-10
3.9	2T Compact-Tension AE Event Count versus Stress Intensity Factor	3-12
3.10	Single-Edge Notch AE Event Count versus Stress Intensity Factor	3-12
3.11	Surface Notch AE Event Count versus Stress Intensity Factor	3-13
3.12	4T Compact-Tension AE Event Count versus Stress Intensity Factor	3-14
3.13	HSST Vessel AE Event Count versus Stress Intensity Factor	3-16
3.14	Room Temperature Fatigue Crack Growth Rate versus Stress Intensity Factor Range	3-18
3.15	Effect of Cyclic Frequency on AE Event Rate for A533B SEN Specimens Tested at Room Temperature and R=0.1	3-19
3.16	Effect of Specimen Geometry on AE Event Rate for A533B Specimens Tested at Room Temperature, R=0.1, Frequency=1-3 Hz	3-21

3.17	Effect of Material on AE Event Rate for SEN Specimens Tested at Room Temperature, R=0.1, Frequency=1-4 Hz	3-21
3.18	Effect of Material on AE Event Rate for 2T-CT Specimens Tested at Room Temperature, R=0.1, Frequency=3 Hz	3-22
3.19	Effect of R-Ratio on AE Event Rate for Base Metal SEN Specimens Tested at Room Temperature and Frequency=1-3 Hz	3-22
3.20	Effect of R-Ratio on AE Event Rate for Base Metal SEN Specimens Tested at 550°F and Frequency=2-3 Hz	3-23
3.21	Effect of Temperature on AE Event Rate for Base Metal SEN Specimens Tested at R=0.1 and Frequency=2-3 Hz	3-23
3.22	Effect of Temperature on AE Event Rate for Weld Metal SEN Specimens Tested at R=0.1 and Frequency=1 Hz	3-24
3.23	Effect of Temperature on AE Event Rate for Base Metal SN Specimens Tested at R=0.1 and Frequency=1-2 Hz	3-24
3.24	Heavy Section Cylindrical Bend Specimen and Electro-Discharge Machined Notch Dimensions	3-26
3.25	Test Specimen and Bending Equipment	3-26
3.26	Summation Acoustic Emission Event Count Versus Load Cycles for Fatigue Cycling of the Heavy Section Cylindrical Bend Specimen	3-27
3.27	Acoustic Emission/Stress Corrosion Crack Growth Test Arrangement	3-27
3.28	Acoustic Emission/Stress Corrosion Cracking Test Results	3-29
3.29	Hypothetical Acoustic Emission Versus Test Pressure Curve for a Nuclear Pressure Vessel Hydro Test	3-30
3.30	Composite Plot of AE/FCG Data and Lower Bound Relationships for A533B Steel at Room Temperature and 550°F	3-32

3.31	Schematic Model of Corrosion Fatigue Crack Growth, Time Domain Form	3-33
3.32	Schematic Model of Corrosion Fatigue Crack Growth, Fracture Mechanics Form	3-33
4.1	Vessel Test Facility at Grosskraftwerk Mannheim (GKM), West Germany	4-3
4.2	ZB-1 Vessel in Test Bunker at GKM	4-3
4.3	ZB-1 Vessel Test Instrumentation in Trailer at GKM Test Site	4-5
4.4	ZB-1 Test Vessel with AE Sensors in Place	4-5
4.5	Instrumentation Mounted on Inside Wall of ZB-1 Test Vessel	4-6
4.6	Unplanned Crack in KS07R Insert Weld in ZB-1 Test Vessel	4-9
4.7	AE Source Location Data Indicating Growth of Unplanned Crack in Fabrication Weld - ZB-1 Test Vessel	4-9
4.8	Accuracy of Crack Growth Rate Estimate Based on AE Data - ZB-1 Test Vessel	4-11
4.9	AE Signal Identification by Three-Pulse Waveguide Sensor Response	4-13
5.1	Areas Monitored for AE at Watts Bar Unit 1 Reactor	5-2
5.2	Sensitivity Test of Waveguide AE Sensors on Calibration Block	5-7
5.3	Location of AE Sensors on Pipe Weld VRR-1RD-1A N2H - Limerick Unit 1	5-9
5.4	Waveguide AE Sensors Mounted Near Weld VRR-1-1A N2H - Limerick Unit 1	5-10
5.5	Waveguide AE Sensors Being Pressure Coupled to the Pipe Surface - Limerick Unit 1	5-10
5.6	Electronic Pulser Mounted Near AE Sensor No. 3 - Limerick Unit 1	5-11
5.7	Modified Waveguide AE Sensors Emerging from Weld Region - Limerick Unit 1	5-12

5.8	AE System Preamplifiers Mounted in Junction Box in Reactor Enclosure, Elevation 283' - Limerick Unit 1	5-13
5.9	AE Monitor Instrument Mounted in a Cabinet in Reactor Enclosure, Elevation 283' - Limerick Unit 1	5-13
5.10	Test of Signal Source Location by AE System Using Signal Input from 0.5-mm Pencil Lead Breaks - Limerick Unit 1	5-14
5.11	Sensitivity Test of Waveguide AE Sensors Installed on Reactor Pipe - Limerick Unit 1	5-15
6.1	Flowchart of Prototypic AE Monitor System	6-3
6.2	Eight-Channel AE Monitor Instrument Developed by Pacific Northwest Laboratory	6-4
6.3	Twenty-Four Channel AE Monitor/Analysis Prototype Developed by Pacific Northwest Laboratory	6-4
6.4	Pattern Recognition Trigger-Window Sequence	6-12
6.5	Design of Waveguide AE Sensor Developed at Pacific Northwest Laboratory	6-15
6.6	Signal Preamplifier Mounted Integral with Waveguide AE Sensor	6-16

TABLES

3.1	Test Conditions for HSST Vessel and Irradiated Fracture Tests	3-3
3.2	Test Conditions for Uniaxial Tensile Tests	3-3
3.3	Test Conditions for Fracture Tests	3-4
3.4	Test Conditions for FCG Tests	3-17
3.5	Least-Squares Regression Coefficients for FCG Rate - ΔK Equation	3-18
5.1	Comparison of Watts Bar and Limerick AE Installations	5-17
6.1	Module Functions	6-10

PUBLISHED REPORTS UNDER THIS PROGRAM

By intention, this list does not include routine monthly and quarterly reports.

Hutton, P. H. and E. B. Schwenk, Pacific Northwest Laboratory, "Program to Develop Acoustic Emission-Flaw Relationship for Inservice Monitoring of Nuclear Pressure Vessels, Progress Report No. 1, July 1, 1976-February 1, 1977," USNRC Report NUREG-0250-1 (1977).

Hutton, P. H. and E. B. Schwenk, Pacific Northwest Laboratory, "Program to Develop Acoustic Emission-Flaw Relationship for Inservice Monitoring of Nuclear Pressure Vessels, Progress Report No. 1, July 1, 1976-February 1, 1977," USNRC Report NUREG-0250-2 (1977).

Hutton, P. H., R. J. Kurtz, E. B. Schwenk, and C. Pavloff, Pacific Northwest Laboratory, "Program to Develop Acoustic Emission-Flaw Relationship for Inservice Monitoring of Nuclear Pressure Vessels, Annual Report, July 1, 1976-October 1, 1977," USNRC Report NUREG/CR-0123 (1978).

Hutton, P. H., R. J. Kurtz, E. B. Schwenk, and C. Pavloff, Pacific Northwest Laboratory, "Program to Develop Acoustic Emission-Flaw Relationship for Inservice Monitoring of Nuclear Pressure Vessels, Progress Report, October 1, 1977-January 1, 1978," USNRC Report NUREG/CR-0124 (1978).

Hutton, P. H., E. B. Schwenk, and R. J. Kurtz, Pacific Northwest Laboratory, "Accomplishments: AE Characterization Program for Remote Flaw Evaluation," PNL Report PNL-SA-6977, presented at ASNT Fall Conference, Denver, Colorado, October 2, 1978.

Doctor, P. G., T. P. Harrington, and P. H. Hutton, Pacific Northwest Laboratory, "Pattern Recognition Methods for Acoustic Emission Analysis," USNRC Report NUREG/CR-0910 (1979).

Hutton, P. H., Pacific Northwest Laboratory, "Progress Report - AE Flaw Characterization Program," presented at the Materials Division Seminar, Aeronautical Research Laboratory, May 1979, Melbourne, Australia.

Hutton, P. H., E. B. Schwenk, and R. J. Kurtz, Pacific Northwest Laboratory, "Estimation of Flaw Significance from Acoustic Emission Data," in Proceedings of Conference on Periodic Inspection for Pressurized Components, p. 267-274 (Mechanical Engineering Pub. Ltd. for the Institution of Mechanical Engineers, London, England, 1979).

Hutton, P. H., E. B. Schwenk, and R. J. Kurtz, Pacific Northwest Laboratory, "Estimate of Feasibility to Develop Acoustic Emission Flaw Relationships for Inservice Monitoring of Nuclear Pressure Vessels," USNRC Report NUREG/CR-0800 (1979).

Hutton, P. H., Pacific Northwest Laboratory, "Acoustic Emission as a Measure of Reactor Pressure Vessel Integrity - A Survey Paper," SMIRT Post Conference Seminar on Assuring Structural Integrity of Steel Reactor Pressure Vessels, August 1979, West Berlin, Germany.

Hutton, P. H., E. B. Schwenk, and R. J. Kurtz, Pacific Northwest Laboratory, "Progress Toward Acoustic Emission Characterization for Continuous Monitoring Reactor Pressure Vessels," Symposium - New Methods of Nondestructive Testing of Materials and Their Application in Nuclear Engineering, September 1979, Saarbrücken, West Germany.

Hutton, P. H. and R. J. Kurtz, Pacific Northwest Laboratory, "Development of Techniques for Continuous Acoustic Emission Monitoring of Light Water Reactors," ASNT Spring Conference, Philadelphia, Pennsylvania (1980).

Hutton, P. H. and R. J. Kurtz, Pacific Northwest Laboratory, "Acoustic Emission: Flaw Relationship for In-Service Monitoring of Nuclear Pressure Vessels," PNL Report PNL-SA-8691, presented at ACRS Metal Component Subcommittee for Reactor Safety Meeting, Washington, D.C., June 17, 1980.

Hutton, P. H., R. J. Kurtz, and J. R. Skorpik, Pacific Northwest Laboratory, "Acoustic Emission for Continuous Surveillance of Various Structures," PNL Report PNL-SA-10191, Eurotest International Conference on New Trends in NDT, Brussels, Belgium (1982).

Hutton, P. H. and R. J. Kurtz, Pacific Northwest Laboratory, "Acoustic Emission and Estimation of Flaw Significance in Reactor Pressure Boundaries," p. 37-44 in Proceedings of Fifth International Conference on Inspection of Pressurized Components (British Institute of Mechanical Engineers) (Mechanical Engineering Publications, London, England, 1982).

Hutton, P. H., T. T. Taylor, J. F. Dawson, R. A. Pappas, and R. J. Kurtz, Pacific Northwest Laboratory, "Acoustic Emission Monitoring of ASME Section III Hydrostatic Test," USNRC Report NUREG/CR-2880 (1982).

Hutton, P. H., R. J. Kurtz, and R. A. Pappas, Pacific Northwest Laboratory, "AE/Flaw Characterization for Nuclear Pressure Vessels," PNL Report PNL-SA-11396, presented at Review of Progress in Quantitative NDE, Santa Cruz, California, August 7, 1983.

Hutton, P. H., R. J. Kurtz, and R. A. Pappas, Pacific Northwest Laboratory, "Acoustic Emission Monitoring of ZB-1 Intermediate Scale Vessel Test," PNL Report PNL-SA-11636, Materialprüfungsanstalt (MPA) Seminar on Safety of Light Water Reactor Pressure Vessels, Stuttgart, West Germany, October 13, 1983.

Hutton, P. H. and R. J. Kurtz, Pacific Northwest Laboratory, "Acoustic Emission for On-line Monitoring: Results of Intermediate Vessel Test Monitoring and Reactor Hot Functional Testing," Vol. 4, p. 1-19, in Proceedings of the Eleventh Water Reactor Safety Research Information Meeting, USNRC Conference Proceeding NUREG/CP-0048, Vol. 4 (1984).

Hutton, P. H. and R. J. Kurtz, Pacific Northwest Laboratory, "Acoustic Emission for On-Line Reactor Monitoring: Results of Intermediate Scale Vessel Test Monitoring and Reactor Hot Functional," PNL Report PNL-SA-12237, presented at Review of Progress in Quantitative NDE, LaJolla, California, July 8, 1984.

Hutton, P. H., J. F. Dawson, M. A. Frisel, J. C. Harris, and R. A. Pappas, Pacific Northwest Laboratory, "Acoustic Emission Monitoring of Hot Functional Testing," USNRC Report NUREG/CR-3693 (1984).

Doctor, S. R., T. T. Taylor, and P. H. Hutton, Pacific Northwest Laboratory, "Overview of NRC Funded NDE Research at U.S. Pacific Northwest Laboratory," presented MPA Seminar on Safety of Light Water Reactor Pressure Vessels, Stuttgart, F.R. Germany, October 1984.

Hutton, P. H., R. J. Kurtz, and R. A. Pappas, Pacific Northwest Laboratory, "AE/Flaw Characterization for Nuclear Pressure Vessels," p. 735-743 in Review of Progress in Quantitative Nondestructive Evaluation, Vol. 3B, Ed. D. O. Thompson and D. E. Chimenti (Plenum Press, New York, New York) (1984).

Hutton, P. H. and R. J. Kurtz, Pacific Northwest Laboratory, "Acoustic Emission/Flaw Relationship for In-service Monitoring of Nuclear Pressure Vessels, Quarterly Report, October 1983-March 1984," USNRC Report NUREG/CR-3825, Vols. 1 and 2 (1984).

Hutton, P. H., R. J. Kurtz, and R. A. Pappas, "Acoustic Emission for On-Line Reactor Pressure Boundary Monitoring," presented at the 7th International Conference on NDE in the Nuclear Industry, February 1985, Grenoble, France.

Hutton, P. H., R. J. Kurtz, M. A. Friesel, R. A. Pappas, J. R. Skorpik, J. F. Dawson, Pacific Northwest Laboratory, "Summary of Detection, Location, and Characterization Capabilities of AE for Continuous Monitoring of Cracks in Reactors," Vol. 4, p. 362-380 in Proceedings of the Twelfth Water Reactor Safety Research Information Meeting, USNRC Conference Proceeding NUREG/CP-0058, Vol. 4 (1985).

Hutton, P. H., R. J. Kurtz, and R. A. Pappas, Pacific Northwest Laboratory, "Acoustic Emission for On-Line Reactor Pressure Boundary Monitoring," p. 363-366 in Proceedings of the 7th International Conference on NDE in the Nuclear Industry (American Society of Mechanical Engineers, 1985).

Hutton, P. H. and R. J. Kurtz, Pacific Northwest Laboratory, "Acoustic Emission/Flaw Relationship for In-service Monitoring of Nuclear Pressure Vessels, Quarterly Report, April-September 1984," USNRC Report NUREG/CR-3825, Vols. 3 and 4 (1985).

Hutton, P. H. and R. J. Kurtz, Pacific Northwest Laboratory, "Acoustic Emission for On-line Monitoring: Results of Intermediate Vessel Test Monitoring and Reactor Hot Functional Testing," p. 3-11, Nuclear Engineering and Design, April 1985.

Hutton, P. H. and R. J. Kurtz, Pacific Northwest Laboratory, "Current Capabilities for Continuous AE Monitoring to Detect Flaws in Reactor Pressure Boundaries," PNL Report PNL-SA-13035, presented at 1985 ASME Piping and Pressure Vessel Conference, New Orleans, Louisiana, June 24, 1985.

Hutton, P. H., Pacific Northwest Laboratory, "Continuous AE Monitoring of Nuclear Plants to Detect Flaws - Status and Future." Presented at OECD/CSNI Specialists Meeting on Continuous Monitoring Techniques for Assuring Coolant Circuit Integrity, August 1985, London, England.

Hutton, P. H., R. A. Pappas, and M. A. Friesel, Pacific Northwest Laboratory, "Acoustic Emission Monitoring of Preservice Testing at Watts Bar Unit 1 Nuclear Reactor," PNL Report PNL-SA-12567, presented at Structural Mechanics in Reactor Technology Conference No. 8, August 1985, Brussels, Belgium.

Doctor, S. R., T. T. Taylor, and P. H. Hutton, Pacific Northwest Laboratory, "Overview of NRC Funded NDE Research at U.S. Pacific Northwest Laboratory," Nuclear Engineering and Design (1985).

Hutton, P. H., R. J. Kurtz, R. A. Pappas, J. F. Dawson, L. S. Dake, and J. R. Skorpik, Pacific Northwest Laboratory, "Acoustic Emission Results Obtained From Testing the ZB-1 Intermediate Scale Pressure Vessel, USNRC Report NUREG/CR-3915 (1985).

Hutton, P. H., Pacific Northwest Laboratory, "On-Line Acoustic Emission Monitoring of Nuclear Reactor Systems - Status and Future." Presented at 2nd International Conference on Acoustic Emission, October 1985, Lake Tahoe, Nevada.

Hutton, P. H., R. J. Kurtz, and R. A. Pappas, Pacific Northwest Laboratory, "Acoustic Emission Monitoring of ZB-1 Intermediate Scale Vessel Test," Nuclear Engineering and Design, 84:2, p. 171-178 (1985).

Hutton, P. H. and R. J. Kurtz, Pacific Northwest Laboratory, "Acoustic Emission for On-Line Reactor Monitoring: Results of Intermediate Vessel Test Monitoring and Reactor Hot Functional Testing," Nuclear Engineering and Design, 86:1, p. 3-11 (1985).

Hutton, P. H. and R. J. Kurtz, Pacific Northwest Laboratory, "Acoustic Emission for On-line Reactor Monitoring of Intermediate Vessel Test Monitoring and Reactor Hot Functional," p. 701-708 in Review of Progress in Quantitative Nondestructive Evaluation, Vol. 4B, Ed. D. O. Thompson and D. E. Chimenti (Plenum Press, New York, New York) (1985).

Hutton, P. H. and R. J. Kurtz, Pacific Northwest Laboratory, "Acoustic Emission/Flaw Relationship for In-service Monitoring of Nuclear Pressure Vessels, Progress Report, October 1984-March 1985, USNRC Report NUREG/CR-4300, Vol. 1 (1985).

Hutton, P. H., R. J. Kurtz, and M. A. Friesel, Pacific Northwest Laboratory, "Progress for On-Line Acoustic Emission Monitoring of Cracks in Reactor Systems," Vol. 2, p. 553-564 in Proceedings of the Thirteenth Water Reactor Safety Research Information Meeting, USNRC Conference Proceeding NUREG/CP-0072, Vol. 2 (1986).

Hutton, P. H. and R. J. Kurtz, Pacific Northwest Laboratory, "Acoustic Emission/Flaw Relationship for In-service Monitoring of Nuclear Pressure Vessels, Progress Report, April-September 1985," USNRC Report NUREG/CR-4300, Vol. 2 (1986).

Hutton, P. H. and R. J. Kurtz, Pacific Northwest Laboratory, "Acoustic Emission/Flaw Relationship for In-service Monitoring of Nuclear Pressure Vessels, Progress Report, October 1985-March 1986, USNRC Report NUREG/CR-4300, Vol. 3, No. 1 (1986).

Hutton, P. H., Pacific Northwest Laboratory, "Acoustic Emission/Flaw Relationship for In-service Monitoring of Nuclear Pressure Vessels, Progress Report, April 1986-September 1986," USNRC Report NUREG/CR-4300, Vol. 3, No. 2 (1987).

Hutton, P. H., R. J. Kurtz, and M. A. Friesel, Pacific Northwest Laboratory, "Progress for On-Line Acoustic Emission Monitoring of Cracks in Reactor Systems," p. 135-140, Nuclear Engineering and Design, NEDEA, 1987.

Hutton, P. H., M. A. Friesel, and R. J. Kurtz, Pacific Northwest Laboratory, "Progress for On-Line Acoustic Emission Monitoring of Cracks in Reactor Systems," Vol. 2, p. 43-56, in Proceedings of the Fourteenth Water Reactor Safety Research Information Meeting, USNRC Conference Proceeding NUREG/CP-0082, Vol. 2 (1987).

Hutton, P. H., M. A. Friesel, J. F. Dawson, and J. C. Harris, Pacific Northwest Laboratory, "Acoustic Emission System Calibration at Watts Bar Unit 1 Nuclear Reactor," USNRC Report NUREG/CR-5144 (1988).

Hutton, P. H., M. A. Friesel, and R. J. Kurtz, Pacific Northwest Laboratory, "On-line Acoustic Emission Monitoring for Crack Growth in LWRs," Vol. 2, p. 333-342, in Proceedings of the Fifteenth Water Reactor Safety Research Information Meeting, USNRC Conference Proceeding NUREG/CP-0091, Vol. 2 (1988).

Hutton, P. H., Pacific Northwest Laboratory, "Acoustic Emission for Continuous Monitoring of Light Water Reactor Systems - Current Status," PNL Report PNL-SA-17058, presented at the 1989 Post-SMIRT Conference No. 14, Monterey, California, August 13-14, 1989.

ACOUSTIC EMISSION/FLAW RELATIONSHIPS
FOR INSERVICE MONITORING OF LWRs
FIN B2088 and B2905

FINAL REPORT

Volume 1 - Program Structure, Results, Conclusions, and Benefits

1.0 INTRODUCTION

Acoustic emission (AE) denotes a nondestructive method for volumetric surveillance of metal or composite structures to detect cracking. AE operates on the basis of detecting energy released as a crack grows in a solid material by detecting the resulting stress waves propagated through the material. Fixed sensors acoustically coupled to the structure detect stress waves originating some distance from the sensor (the detection distance is a variable depending on the type of material in the structure and the monitoring frequency). Information from more than one sensor used in combination also allows identification of the source of the stress wave. Since it is not necessary to move the sensors over the surface of the structure, the AE method can provide constant remote surveillance for crack growth over any selected time period. Since AE systems are inherently very sensitive and operate in real-time on signals from a source, they are also effective in detecting leakage of fluid from a crack or other opening. This is particularly true of high pressure-high temperature fluids leaking to atmosphere through a crack in the wall of a container. Work under a separate program has indicated that the acoustic data can be interpreted to estimate leak rate as well as leak location. In this perspective, the AE method is complementary to other forms of nondestructive inspection (ultrasonic, eddy current, radiography, etc.). It brings to the nondestructive inspection field the unique features of remote volumetric surveillance, flaw location, and capability for continuous surveillance over time. Crack growth rate can be estimated from AE information; however, other methods such as ultrasound or radiography must be used to determine crack size and shape.

Concerted development of AE as a nondestructive testing tool to evaluate the integrity of structures began in the United States in the early 1960s. One of the earliest applications was made by Aerojet General in about 1962 to monitor proof testing of filament-wound composite rocket engine casings. Study and development of an understanding of AE principles was also in progress at locations such as Lawrence Livermore Laboratory, University of Michigan, Lessels and Associates, and Boeing Aircraft-Seattle. Development for nuclear applications began in earnest in 1965 with two parallel programs funded by the U.S. Atomic Energy Commission. One was at Pacific Northwest Laboratory (PNL) in Richland, Washington and the other at the National Reactor Test Station, Idaho Nuclear Corporation in Idaho Falls, Idaho. The field of participants in development of AE expanded rapidly both nationally and

internationally from these beginnings. There have been several applications of AE monitoring to nuclear systems over the intervening years such as:

- A group from the Idaho Nuclear Corporation, Idaho Falls, Idaho monitored a military power reactor vessel during an annealing procedure to reduce embrittlement of the vessel caused by radiation damage (Parry and Robinson 1970). The reactor system was designated SM-1A and was located at Fort Greeley, Alaska. The work which took place in 1967 consisted of measuring the AE produced by the vessel during a hydrostatic test before and after the annealing. Reported reduction in AE associated with annealing agreed with the measured reduction in embrittlement within 10%.
- The reactor vessel and primary coolant system at Calvert Cliffs Unit 1 nuclear power plant were AE monitored during cold hydrostatic preservice testing in 1973 by the Exxon Nuclear Company's Acoustics Group (Exxon Nuclear Company 1973). Several AE indications were detected from the vessel and two of the loops, but all were reported as being insignificant.
- Ultrasonic inservice inspection showed flaw indications associated with two vessel nozzles on the Pilgrim Unit 1 Nuclear Power Station in 1976. These two nozzles were subsequently monitored during a hydrostatic test by the Exxon Nuclear Company's Acoustics Group (Exxon Nuclear Company 1976). Although some AE was detected, the conclusions agreed with other analysis approaches that the flaw indications were insignificant.
- The Philadelphia Electric Company, together with the Technology for Energy Corporation, performed continuous on-line AE monitoring of feedwater nozzles, core spray lines, and recirculation by-pass lines at the Peach Bottom Unit 3 Nuclear Generating Station over the period 1976 to 1980 (McElroy 1982). There was very limited AE detected. The core spray lines produced the most AE, but it was still quite low and considered to be insignificant. Inservice inspection (ISI) results agreed with the AE information; i.e., in regions where there was little or no AE, no flaw indications were found and in regions where there was limited AE, flaw indications were found but they were insignificant. The work also demonstrated that AE monitoring could be performed during reactor operation without being overwhelmed by background noise. The obvious missing element due to the circumstances is that the work did not demonstrate detection of a significant flaw correlated by ultrasonic inspection.
- A hydrostatic test was performed on the High Flux Isotope Reactor vessel at Oak Ridge National Laboratory in 1987 to evaluate safety implications of neutron embrittlement around the beam tube nozzles. These areas were AE monitored by PNL during the test (hydrostatic overpressure) to determine if crack pop-in occurred during the pressure test. AE results agreed with the other indicators that crack growth or pop-in did not take place (U.S. Nuclear Regulatory Commission 1988).

- AE technology is currently being used on a number of commercial power reactors for surveillance of critical valve performance and loose parts detection (Hartman 1986).

Development of AE technology suffered initially from over enthusiasm on the part of the AE community in the U.S. This led to some disappointing results in application efforts and to a premature and unsuccessful attempt to incorporate AE in the American Society for Mechanical Engineers (ASME) Code in 1974. Although this retarded the development support for a period, it had a beneficial side in the sense that it tended to highlight the parts of the technology that needed developmental emphasis to overcome problems. Fortunately, it did not lead to discarding AE as being without substance.

There are two primary periods in the life of a nuclear pressure containing system when AE monitoring is applicable. The first is during the preservice period to monitor vessel acceptance testing and preservice testing of the system (cold hydrostatic testing and hot functional testing). The second is during its service life. The AE monitoring requirements and techniques for these two periods are very different. This report focuses on the inservice period only and conclusions presented do not necessarily apply to the preservice period. For inservice AE monitoring, two potential basic approaches have been evident from the beginning for surveillance of reactor pressure containing components to detect cracking and/or leaks. One approach is to monitor the components only during hydrostatic testing of the pressure system which normally takes place at the end of a reactor outage. The system pressure is increased a modest amount above operating pressure; in the U.S., the overpressure ranges from 102 to 110% of operating pressure depending on system temperature and plant age. The other approach is to monitor the components continuously during reactor operation when the components are exposed to the environment which could generate flaws or grow existing flaws. The problems associated with continuous monitoring (high noise background, hostile temperature and radiation environment, lack of access to the pressure system during reactor operation, etc.) caused most researchers both domestic and foreign to concentrate on hydrostatic test monitoring. There are, however, considerations in favor of continuous monitoring that justified an effort to overcome the problems. These include:

- enhanced safety by constant surveillance of a component(s) to detect and locate cracking as it occurs and estimate crack severity during reactor operation.
- increased assurance of detecting a crack by monitoring during the initiation and growth of the crack compared to monitoring only during the brief period of a hydrostatic test. The limited increase in stress on a component during hydrostatic testing compared to the operating stress level that would have caused a subcritical defect or crack to form and grow during operation is insufficient to produce significant AE even from a major crack in one pressure cycle.

- potential for significant savings in plant operating costs by forestalling augmented ISI required in certain cases to evaluate flaw indications detected during normal ISI inspections.

With these benefits in mind, a carefully planned research program with the objective of validating AE technology for continuous monitoring of reactor pressure boundaries to detect crack initiation and growth was conceived in 1976. The primary program support has come from the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research with limited supplemental funding provided by the Tennessee Valley Authority. The work has been performed by PNL.

2.0 PROGRAM SCOPE AND ORGANIZATION

The scope of the NRC-sponsored AE/Flaw Characterization Program was challenging. It included the following topics:

- Characterize AE from a limited range of defects and material property conditions recognized as being of primary concern to reactor pressure system integrity.
- Characterize AE from innocuous sources which can be reasonably expected to exist in a pressure vessel.
- Develop a criterion for distinguishing significant flaws from innocuous sources.
- Develop an AE/flaw damage model to serve as a basis for relating inservice AE to pressure vessel integrity.
- Test and refine the AE/flaw relationships by monitoring an off-reactor vessel test(s) which simulates the conditions on an operating reactor. Also, utilize HSST vessel tests to develop and refine relationships.
- Design and fabricate a prototypic AE monitor and analysis system suitable for continuous monitoring of an operating reactor.
- Validate the AE technology by installing an AE monitoring system on a reactor and demonstrate detection of crack growth during reactor operation.
- Prepare and obtain approval of ASTM Standards and ASME Code documents needed for effective utilization of the technology by industry.
- Prepare documentation of program software and hardware designs to facilitate technology transfer to industry.

The program was organized into three phases aimed at fulfilling the requirements of the scope. These phases consisted of:

- Phase I. Utilize laboratory specimen testing to develop initial AE/flaw relationships for distinguishing crack growth AE signals from noise and for estimating flaw severity. This also included investigation of AE characteristics from different cracking modes (fatigue, fracture, IGSCC, etc.), temperature effects, material effects, size effects, and prestrain effects.
- Phase II. Evaluate and refine the technology developed in the laboratory by monitoring fatigue testing of a heavy section vessel under simulated reactor conditions.

Phase III. Validate technology in the field and transfer technology to industry. Included here is:

- validation of continuous AE monitoring on an operating nuclear power reactor system.
- obtain ASTM and ASME approval of the appropriate documents covering continuous on-line AE monitoring of reactor system components.
- document software and hardware design information to facilitate commercialization of the technology.

3.0 PHASE I - LABORATORY RESEARCH

Two primary criteria were agreed to at the outset of the AE/Flaw Characterization Program to emphasize the applied research nature of the program:

- A restricted number of variables in terms of specimen geometries, flaw type, flaw growth mechanisms, innocuous noise sources, material properties, and environmental conditions were to be considered for practical reasons of focusing the scope and limiting the duration of the program.
- The program sought to develop macro-scale AE/flaw growth relationships. Although current knowledge of basic AE source mechanisms was by necessity used in analysis, it was not the function of this program to further investigate basic AE source mechanisms.

These are practical engineering decisions applied to a research program. They recognize that not all possible conditions and combinations need be directly investigated to reach the program objectives. The program focused on those variables considered to be most important to developing effective AE monitoring of nuclear reactor components to detect inservice crack growth.

3.1 APPROACH AND APPARATUS

An "Analysis Before Test" document¹ was prepared at the outset of the program wherein the test approach, test equipment, and data analysis approach were defined. These items were addressed under the topics of "Material and Test Specimens" and "Acoustic Emission Measurement."

3.1.1 Material and Test Specimens

The primary material used in Phase I testing was SA-533 Type B, Class 1 pressure vessel steel. The A533B material came from two principal sources: Lukens Steel Company of Coatsville, Pennsylvania and the Oak Ridge National Laboratory Heavy Section Steel Technology (HSST) program. Most test specimens were fabricated from an 8-1/4-in. x 104-in. x 175-in. plate purchased from Lukens Steel. Chemical analysis, physical properties, Charpy impact, drop weight transition temperature data, heat treatment and nondestructive test results on this plate have been reported by Hutton and Schwenk (1977a). Two additional A533B plates were obtained from the HSST program. The first plate

¹"Analysis Before Test - Program to Develop Acoustic Emission Characterization of Flaw Growth in A533B Pressure Vessel Steel," submitted to Metallurgy and Materials Branch, Reactor Safety Research Division, U.S. Nuclear Regulatory Commission, by P. H. Hutton and E. B. Schwenk, Pacific Northwest Laboratory, December 3, 1976.

(57G) was 6 in. x 20 in. x 23 in. with a through-thickness weld running parallel to the 23-in. dimension at the midpoint of the 20-in. dimension. Details on the material and welding conditions of this plate were reported by Hutton and Schwenk (1977b). The second plate (01NB1) was 12 in. x 21 in. x 28 in. with a through-thickness weld running parallel to the 21-in. dimension and centered in the 28-in. dimension. The material and welding conditions for this plate were presented in reports by Childress (1969 and 1971). In addition, a small quantity of 1-1/2 in. thick A533B Class 2 steel was available at PNL. This material was originally purchased from Lukens Steel and was used to fabricate five of the tensile specimens tested.

SA106 Grade B piping steel used to perform two fatigue crack growth tests was obtained from Kilsby Tube Supply of Kent, Washington. A 4 ft. length of 24-in. Schedule 160 pipe was procured to fabricate appropriate test specimens. Chemical and mechanical properties for this material were given by Edler (1982).

Stress corrosion cracking tests were performed using 4 inch, Schedule 80, Type 304 stainless steel pipe. Material with a 0.06 carbon content was used to enhance sensitization.

The basic types of mechanical tests conducted during Phase I included uniaxial tensile, fracture, and fatigue crack growth (FCG). These types of tests were performed to assess AE associated with a growing elastic/plastic strain field, a growing crack-tip plastic zone, and a growing crack. Test specimens were machined from either skin material (approximately the outer two inches of a plate) or core material (approximately the center four inches of a plate). The test matrix emphasized skin material specimens, since it was assumed that flaws in an operating reactor would initiate and grow from free surfaces. Most specimens were removed from base metal, but some specimens were machined so that flaw growth occurred in weld metal only. Specimen types included round uniaxial tensile specimens (1/4-in. diameter gauge section), plate-tension specimens (nominally 1-in. thick with either a 3-in. or 6-in. wide gauge section), 2T compact-tension (CT) specimens, and a heavy-section cylindrical bend (HSCB) specimen. Plate-tension specimens were either single-edge notched (SEN) or surface notched (SN) to simulate both through-wall and part-through notch conditions. An overview of specimen fabrication methods was described by Hutton, Schwenk, Kurtz, and Pavloff (1978). All base metal specimens fabricated from the Lukens A533B plate were stress-relieved by heating to $1150^{\circ}\text{F} \pm 25^{\circ}\text{F}$ for 40 hours. After heat treatment, these specimens were lightly grit blasted to remove heat treatment induced oxidation, which may have caused spurious acoustic emissions (AEs).

Tables 3.1-3.4 list the specific specimen and material combinations used for each uniaxial tensile, fracture, and FCG test, respectively. Also listed is the test temperature and loading conditions employed. The Phase I mechanical test procedures have been documented previously (Hutton, Schwenk, Kurtz, and Pavloff 1978).

TABLE 3.1. Test Conditions for HSST Vessel and Irradiated Fracture Toughness Tests

Specimen Number	Specimen Type	Thickness, in.	Width in.	Material	Test Temperature °F	Comments
ITV-7B	ITV	6.00	-	WM	200	Flaw in HAZ of repair weld
ITV-8	ITV	6.00	-	WM	-9	
67W-23	CT	4.00	8.00	WM	392	Irradiated (5.3×10^{18} n/cm ²)
65W-25	CT	4.00	8.00	WM	392	Irradiated (3.8×10^{18} n/cm ²)
65W-22	CT	4.00	8.00	WM	392	Unirradiated

ITV = Intermediate Test Vessel
CT = Compact Tension
HAZ = Heat Affected Zone
WM = Weld Metal

TABLE 3.2. Test Conditions for Uniaxial Tensile Tests

Specimen Number	Material	Test Temperature, °F	Notch Condition	Stress Conc. Factor	Initial Crosshead Velocity, in./min.	Final Crosshead Velocity, in./min.	Comments
A-1-5	SW(1)	RT	UN	1	0.002	0.02	
AA-1-1	CW(1)	RT	UN	1	0.002	0.02	
AA-2-1	CB(1)	RT	UN	1	0.002	0.02	
AET-21	SB(3)	RT	UN	1	0.002	0.02	
AET-25	SB(3)	RT	N	≤3.8	0.002	0.002	
AET-22	SB(3)	RT	UN	1	0.002	0.002	Heat Treated to RC 45
A-1-4	SW(1)	RT	UN	1	0.002	0.02	
A-1-1	SW(1)	RT	UN	1	0.002	0.02	
AET-23	SB(3)	RT	UN	1	0.002	0.02	
AET-10	SB(3)	RT	UN	1	0.002	0.02	
A-2-2	SB(1)	RT	UN	1	0.002	0.02	
A-2-3	SB(1)	RT	UN	1	0.002	0.02	
A-2-1	SB(1)	550	UN	1	0.02	0.02 AND 0.0005	
5A-5-1	SB(2)	RT	N	5.8	0.002	0.002	
5A-5-2	SB(2)	RT	N	≤4.9	0.002	0.002	
5A-5-4	SB(2)	RT	N	2.1	0.002	0.002	

SW = Skin Weld
SB = Skin Base
CW = Core Weld
CB = Core Base

RT = Room Temperature
UN = Unnotched
N = Notched

(1) = HSST Plate 57G
(2) = 8-1/4" Primary Material (Lukens)
(3) = PNL A533B, Class 2 Material

TABLE 3.3. Test Conditions for Fracture Tests

Specimen Number	Specimen Type	Thickness, in.	Width, in.	Material	Test Temperature °F	Precrack Kmax, ksi./in.
B2-3	CT	1.75	4.00	SB(1)	RT	31.5
B2-1B	CT	1.75	4.00	SB(1)	RT	74.2
B2-3A	CT	1.75	4.00	SW(1)	RT	31.5
1-2A-2B	SEN	1.00	6.00	SB(2)	RT	20.8
1-2A-6B	SEN	1.00	6.00	SB(2)	550	119.0
2-1A-2B	SN	1.00	3.00	SB(2)	RT	41.7
1-2A-4A	SN	1.00	3.00	SB(2)	550	37.0
2W-1A	SN	1.00	3.00	SW(4)	550	41.7

CT = Compact Tension
SEN = Single Edge Notch
SN = Surface Notch

SB = Skin Base
SW = Skin Weld
RT = Room Temperature

(1) = HSST Plate 57G
(2) = 8-1/4" Primary Material (Lukens)
(4) = HSST Plate 01NB1

To broaden the Phase I data base, PNL also participated in two HSST intermediate-scale pressure vessel tests conducted at ORNL (Figure 3.1 shows an HSST vessel monitored) and three 4T compact-tension fracture toughness tests performed at the Naval Research Laboratory (NRL). Two of these 4T compact-tension specimens had been irradiated, thus offering a unique opportunity to determine the AE response of irradiated A533B weldment. Monitoring of these tests also yielded an opportunity to acquire data from specimens that more closely approximate the thickness of an actual reactor pressure vessel. Table 3.1 lists the test conditions for the HSST vessel and irradiated fracture toughness tests. More detailed descriptions of the test conditions may be found in reports by Bryan et al. (1978); Bryan et al. (1979); and Hiser, Loss, and Menke (1984).

3.1.2 Acoustic Emission Measurement

The AE instrument system used for monitoring laboratory tests was a PNL-built, two-channel laboratory instrument which recorded data in digital form on an EPROM (erasable, programmable, read only memory) chip. The instrument was subsequently modified to store the data on digital cassette tape to facilitate reading the information into a computer for analysis. The monitor instrument is shown in Figure 3.2 together with a description of the functions performed by the various modules. Data recorded on the EPROMs during a test were retrieved by a special memory reader instrument for analysis. EPROMs are shown in Figure 3.3. The EPROM approach for data storage suffered from a major disadvantage when many parameters were being measured simultaneously as in the Phase I laboratory tests. It was very cumbersome to enter the data from the EPROMs into a computer for efficient analysis. This was remedied by replacing the EPROMs with an interface unit and a commercial cassette recorder as shown in Figure 3.4, which facilitated reading the data directly into a computer for analysis.

**HSST V7-B
TEST VESSEL**

**~8 FT. HIGH
39 INCHES O.D.
6 INCH WALL
NUCLEAR
PRESSURE
VESSEL STEEL**

FIGURE 3.1. HSST V7-B Vessel AE Monitored During Testing - July 1977

FIGURE 3.2. Functional Description of the Phase 1 Laboratory Test AE Monitor System

FIGURE 3.3. EPROM Digital Storage Devices

FIGURE 3.4. Phase 1 Laboratory Test AE Monitor System with Interface Module and Digital Cassette Recorder

In addition to the AE instrument described above, an NRC-owned Dunegan/Endevco 1032 computerized commercial AE monitor instrument was used in HSST vessel tests monitored to give real-time source location information. The PNL-built instrument was also used in parallel to obtain data directly relatable to laboratory test data.

AE sensors used in the laboratory work, which are shown in Figure 3.5, were shop made. These sensors were built into a BNC connector component which made them very convenient to use and inexpensive. They consist of a 1-MHz cut PZT5 piezoelectric crystal with a ceramic wear face to protect the crystal. They are not suited to a high-temperature environment. Metal standoffs were used for all of the high-temperature tests to keep the sensor at a low enough temperature to avoid damage. A more refined high-temperature waveguide AE sensor is discussed later in the report under Phase II and Phase III sections.

FIGURE 3.5. Shop-Made Surface Mount AE Sensors Used in Phase 1 Laboratory Specimen Tests

3.2 RESULTS

3.2.1 Tensile Tests

Sixteen tensile specimens were tested as shown in Table 3.2. All were 0.25-in. diameter (nominal) except for the heat-treated specimen which required further reduction in gauge section area to ensure that specimen maximum loads did not exceed the capability of the tensile test machine.

Fourteen of the specimens were removed from the skin surface material of which three included the HSST weldment 57G in the gauge section. The remaining two specimens were removed from the core or mid-thickness region of the 6-in. thick HSST material. Four specimens were notched to obtain different levels of stress concentration factor. One specimen was heat treated to Rockwell "C" 45 to investigate the effect of hardness (strength) on AE. One specimen was tested at 550°F (nominal). All of the unnotched specimens were characterized by a relatively small amount of AE (1 to 25 event count with the average of about 8 event count per specimen). Some of the few AE event counts occurred around small, relatively uniform load drops following yielding, some around load increases associated with increasing crosshead speed. Crosshead travel rates ranging from 0.0005 in./min. to 0.02 in./min. had no apparent effect on the few event counts produced. It should be noted that no discontinuous or serrated yielding occurred in any of the specimens. The gauge section surfaces were also machined to an 8 μ in. finish to assist in detecting Luders bands, none of which were seen.

Differences in metallurgical condition (namely, Class 1 versus Class 2 material, skin versus core, base metal versus weld metal, and hardness) had no apparent effect in the small amount of AE detected. Testing at 550°F also had no significant affect on AE. It should be noted, however, that only one specimen was tested at an elevated temperature.

The AE data detected from the notched tensile specimens is plotted in Figure 3.6 as a function of stress concentration factor. The average AE count produced from 11 unnotched specimens is also shown in Figure 3.6. The results indicate that one specimen (AET-25) gave considerably higher AE than all the other specimens. Because of this inconsistency, optical metallography and scanning electron microscopy (SEM) were performed on specimens AET-25 and 5A-5-3 (6 event count) to determine if an abnormal AE source existed at or near the notch in AET-25. The metallurgical analysis results suggested that the increased AE detected in AET-25 may have been related to inclusion morphology. SEM examination showed that both fracture surfaces exhibited the expected ductile rupturing; but on AET-25, several large, cracked inclusions were noted, whereas these same features were generally absent from the 5A-5-2 fracture surface. As-polished longitudinal metallographic sections indicated that the number of inclusions near the notch was nearly equal. The only difference between the two microstructures was the orientation and morphology of inclusions. Inclusions in AET-25 were elongated along the longitudinal axis of the specimen, whereas inclusions in 5A-5-2 were smaller and more equiaxed.

FIGURE 3.6. Acoustic Emission Event Count Versus Stress Concentration Factor for Uniaxial Tensile Tests

In summary, the results of the tensile tests showed that within the range of conditions studied, AE was relatively unaffected by microstructure, temperature, and strength level. No difference was observed between skin base and core base, skin weld and core weld, Class 1 and Class 2, heat-treated Class 2 material, crosshead travel rate, temperature, and stress concentration factor. Little AE was detected during elastic-plastic deformation except when inclusions were involved. One of the notched specimens produced significantly higher AE than the unnotched specimens. Metallurgical analysis suggested that this was related inclusion morphology.

3.2.2 Fracture Tests

Eight laboratory fracture tests were performed using three specimen geometries as summarized in Table 3.3. The CT and SEN specimens simulated a through-wall crack orientation, and the SN specimens simulated part-through-wall cracks. Typical test results are plotted in Figures 3.7 and 3.8. The plots show the load and AE event count versus time (Figure 3.7) or crack opening displacement (Figure 3.8) for a 2T-CT and an SN specimen, respectively. These data were typical for all of the laboratory fracture tests in that most of the AE was detected after maximum load was exceeded and, therefore, occurred from ductile crack extension during mechanical instability.

FIGURE 3.7. Load and AE Event Count Versus Time for Fracture Test of Specimen B2-1B

FIGURE 3.8. Load and AE Event Count Versus COD for Fracture Tests of Specimen 1-2A-4A

To compare results from different specimens, plots of the AE event count versus stress intensity factor were developed. Stress intensity factors were calculated up to the maximum load attained during a particular test. It was assumed that no crack extension occurred prior to maximum load. It was recognized that for most specimens net-section yielding probably began prior to reaching maximum load. This condition would make the calculation of stress intensity factors based on linear elastic fracture mechanics concepts invalid; however, no attempt was made to modify or correct the calculated stress intensity factors for net-section yielding conditions. Figures 3.9-3.11 show the AE event count versus stress intensity factor plots by specimen geometry. The results were plotted to the same scale to facilitate comparisons between different specimen geometries. All of the specimens showed similar behavior in that little AE was detected during the early portion of the fracture test. Typically, fewer than ten AE events were recorded prior to onset of net-section yielding, and no more than 30 to 40 total AE events were detected up to maximum load except for one specimen. The lack of AE at low stress intensity levels was not related to fatigue pre-cracking conditions. As shown in Table 3.3, the final fatigue pre-cracking stress intensity factors varied from 20.8 ksi/ $\sqrt{\text{in.}}$ up to 119 ksi/ $\sqrt{\text{in.}}$. The specimen with the highest fatigue pre-cracking stress intensity (1-2A-6B) also produced the highest AE response at low stress intensity factors. This specimen also gave the highest overall AE response up to maximum load. No explanation has been developed that accounts for the difference in AE response from specimen 1-2A-6B compared to the other laboratory fracture test specimens.

In addition to the lack of a specimen geometry effect, the AE response during fracture testing was also unaffected by metallurgical condition. Specimens made from weld metal gave essentially the same AE response as base metal specimens (see Figures 3.9 and 3.11). The effect of temperature, on the other hand, was not clear from the fracture test results. Comparing results for SEN specimens tested at RT and 550°F (Figure 3.10) suggests that increasing temperature causes increased AE response. In contrast, the results for SN specimens (Figure 3.11) showed no significant effect of temperature. Since the specimen displaying a temperature effect was markedly different than all the other fracture specimens, it seems reasonable to conclude that test temperatures from RT to 550°F probably do not affect AE under monotonic loading conditions. This conclusion is also supported by the tensile test results presented above.

3.2.3 HSST Irradiated Fracture Toughness Tests

Two irradiated and one unirradiated fracture tests of A533B weldment were monitored for AE at the NRL (see Table 3.1 for test conditions). All three specimens were 4.0-in. thick compact tension specimens with 20% side grooves. The tests were run at 392°F, and J-R curves were generated by the unload compliance technique. The J-integral data, obtained from NRL, was converted to a pseudo-equivalent stress intensity factor (K_I) to facilitate comparisons with the laboratory fracture data and HSST vessel test results.

FIGURE 3.9. 2T Compact-Tension AE Event Count Versus Stress Intensity Factor

FIGURE 3.10. Single-Edge Notch AE Event Count Versus Stress Intensity Factor

FIGURE 3.11. Surface Notch AE Event Count Versus Stress Intensity Factor

The K_j values were calculated from the following equation:

$$K_j = [EJ_I / (1 - \mu^2)]^{1/2} \quad (1)$$

where E and μ are Young's modulus and Poisson's ratio, respectively.

The AE event count versus stress intensity factor results for these tests are given in Figure 3.12. Only the AE data obtained during the ramp loading and first hold period of the test sequence are plotted in Figure 3.12. All AE detected during the ramp unload and second hold period were excluded from the analysis since no crack extension was expected during this portion of the test.

The data indicate significantly more AE from the thicker 4T-CT specimens than the 1.0 to 1.75-in. thick laboratory fracture specimens. Also at low to moderate stress intensity levels (0 to 100 ksi/ $\sqrt{\text{in.}}$), the slope of the AE-K curve was greater for the 4T test specimens. At a stress intensity of 100 ksi/ $\sqrt{\text{in.}}$, the average number of AE events was about 50 for the thicker specimens compared to about 10 to 20 for the thin specimens. For the 4T-CT specimens, continuation of AE at higher stress intensity levels was observed and was probably due to the larger specimens being linearly elastic over a greater range of stress intensity. Comparing irradiated to unirradiated

FIGURE 3.12. 4T Compact-Tension AE Event Count Versus Stress Intensity Factor

specimens shows that irradiation did not significantly influence the AE behavior at the test temperature employed.

3.2.4 HSST Vessel Tests

Two HSST intermediate-scale test vessels (ITV) were monitored during the course of the program (see Table 3.1 for test conditions). A typical vessel is shown in Figure 3.1. The vessels were 39-in. OD with 6-in. wall thickness. The cylindrical test sections were 54-in. long and closed on one end by a welded hemisphere and a bolted flange on the other end. Vessel material was A533, Grade B, Class 1 steel. The objective of test ITV-7B was to evaluate the affect of unstress-relieved repair welds in the vicinity of large flaws. The flaw for test ITV-7B was located in the heat-affected zone of a repair weld made in accordance with ASME Boiler and Pressure Vessel Code, Section XI. The ITV-7B flaw was initially machined to within about 0.4 in. of the desired crack depth, then a narrow electron-beam weld was made along the periphery of the notch, followed by hydrogen charging until a sharp crack popped in. Testing of ITV-7B consisted of pressurizing to failure at 200°F while monitoring for AE.

Test ITV-8 also involved a machined flaw, but the flaw was located in weld material adjacent to a repair weld. The main objectives of the ITV-8 test were to study residual stress effects on flaws in repair weld regions and

to evaluate linear elastic fracture mechanics as a means of predicting the combined effects of residual stress and pressure stress on fracture. The second objective necessitated conducting ITV-8 at low temperature to ensure that the flaw was located in a region of low fracture toughness. Thus, ITV-8 was pressurized to failure at -9°F. Unlike the flaw in ITV-7B, the flaw in ITV-8 was fatigue pre-sharpened by cyclically pressurizing a cavity machined in the vessel wall.

The AE data from both vessel tests is plotted in Figure 3.13. During both of the vessel tests, the AE accumulated in a manner that indicated the feasibility of monitoring of large structures. The AE consistently increased with increasing flaw severity, as measured by stress intensity factor. Moreover, pop-in precursor activity was also detected (as experienced in the ITV-8 test), indicating a sensitivity to relatively small amounts of plasticity and/or crack extensions. During the most dynamic part of the pop-in (including leakage) where signal density was very high (as observed on an oscilloscope), the AE system locked out. This was due to a design feature to avoid multiple event counts from one signal.

Both sets of data display a linear dependence of accumulated AE on stress intensity factor up to the point of large-scale yielding (at the end of ITV-7B). The form of the curves obtained from the vessel tests was similar to both the irradiated and laboratory fracture specimens except that far more AE was detected at low stress intensity levels. This suggests that prototypic thickness specimens are important for obtaining relevant AE-stress intensity data.

3.2.5 Fatigue Crack Growth Tests

Results of two surveys (Gibbons and Hackney 1964 and Phillips and Worwick 1968) of flaws in pressure vessels indicated that flaws occurring during manufacturing and fabrication, coupled with operation-induced FCG from those flaws, was the dominant cause of vessel failure. AE is generated during the growth of fatigue cracks. The use of a continuous monitoring technique such as AE presents a possible method for detecting, locating, and evaluating FCG during reactor operation. Thus, a considerable portion of Phase I research work involved developing an empirical relationship between AE and the rate of FCG.

A total of 14 FCG tests were performed during Phase I. Table 3.4 lists the specific test conditions employed. The test matrix was designed to determine the effects of specimen geometry, metallurgical condition, test temperature, R-ratio, cycle rate, and environment on the rate of AE production as correlated to the FCG rate.

Figure 3.14 gives the relationship between the FCG rate and the stress intensity factor range for all specimens tested in room temperature air. There are 234 data points plotted in Figure 3.14, and it is interesting to note that the FCG rate ranges over almost three orders of magnitude from about 3×10^{-7} in./cycle to about 2×10^{-4} in./cycle. Despite differences in

FIGURE 3.13. HSST Vessel AE Event Count Versus Stress Intensity Factor

TABLE 3.4. Test Conditions for FCG Tests

Specimen Number	Specimen Type	Thickness, in.	Width, in.	Material	Test Temperature, °F	R-Ratio	Cycle Rate, Hz	Comments
B2-1B	CT	1.75	4.00	SB(1)	RT	0.1	3	
B2-1A	CT	1.75	4.00	SW(1)	RT	0.1	3	
1-1A-2A	SEN	0.50	5.00	SB(2)	RT	0.1	1-4	3% Prestrain
1-2A-1B	SEN	1.00	6.00	SB(2)	RT	0.1	2-3	
1-2A-2B	SEN	1.00	6.00	SB(2)	RT	0.1	10	
1-2A-6B	SEN	1.00	6.00	SB(2)	550	0.1, 0.5	2-3	
2-1A-2B	SN	1.00	3.00	SB(2)	RT	0.1	1-2	
1-2A-6A	SN	1.00	3.00	SB(2)	RT, 550	0.1	1	
1-1A-3B	SEN	1.00	6.00	SB(2)	RT	0.1, 0.6	0.1, 2	
1-2A-3B	SEN	1.00	6.00	SB(2)	RT	0.6	1	K-Decreasing
2W-1B	SEN	1.00	6.00	SW(4)	RT, 550	0.1	1	
A106B-1	SEN	1.00	6.00	CB(5)	RT	0.1, 0.6	1	
A106B-2	SEN	1.00	6.00	CB(5)	RT	0.1, 0.6	1	Water Environment

CT = Compact Tension
 SEN = Single Edge Notch
 SN = Surface Notch
 HSCB = Heavy-Section
 Cylindrical Bend

SB = Skin Base
 SW = Skin Weld
 CB = Core Base
 RT = Room Temperature

(1) = HSST Plate 57G
 (2) = 8-1/4" Primary Material (Lukens)
 (3) = HSST Plate 01NB1
 (4) = 24 in. Schedule 160, A106B Pipe

FIGURE 3.14. Room Temperature Fatigue Crack Growth Rate Versus Stress Intensity Factor Range

material, R-ratio, and cyclic frequency, there is very little data scatter. The room temperature data were least squares regression fit to the well known Paris-Erdogan FCG equation:

$$da/dn = C(\Delta K)^m \quad (2)$$

where C and m are constants that depend on the material and test environment, da/dn is the FCG rate in in./cycle, and ΔK is the stress intensity factor range in ksi/√in. Curve fits were also performed for the specimens tested at 550°F and for the A106B pipe material tests. Plots similar to Figure 3.14 were obtained for these specimens. Table 3.5 gives the results of these calculations.

TABLE 3.5. Least-Squares Regression Coefficients for FCG Rate - ΔK Equation

Material	Temperature, °F	n	C	m
A533B	RT	234	1.08×10^{-9}	2.55
A533B	550	60	4.61×10^{-9}	2.27
A106B	RT	80	1.24×10^{-9}	2.62

The curve fit results show that the FCG rate for the 550°F and A106B specimens was greater than the room temperature A533B specimens at all stress intensity factor ranges.

An AE/Flaw Growth Model was developed by correlating the rate of change of AE and FCG. It was assumed that since the FCG rate was power-law dependent on the stress intensity factor range that the AE rate would also exhibit power-law dependency on the FCG rate. An equation of the form

$$dN/dn = C_1 (da/dn)^{m_1} \quad (3)$$

where dN/dn is the AE rate in event count per load cycle was assumed as a relationship between AE and crack growth during fatigue. It was recognized early in the program that a model based on load cycles would not be practical for field monitoring since this parameter could not be determined. Thus, one of the important laboratory experiments was to determine the effect of load cycle rate on the AE rate. Figure 3.15 shows a plot of the AE rate in events/load cycle versus the FCG rate in in./cycle for single-edge notch specimens tested in room temperature air over a cyclic frequency ranging from 0.1 Hz to 10 Hz. Based on these results, it was assumed that cyclic frequency did not have a significant effect on the AE rate and that an equation of the form

$$dN/dt = C_2 (da/dt)^{m_2} \quad (4)$$

FIGURE 3.15. Effect of Cyclic Frequency on AE Event Rate for A533B Single-Edge Notch Specimens Tested at Room Temperature and R=0.1

where dN/dt is the AE rate in event count per second and da/dt is the crack growth rate in microinches/sec. Thus, the remainder of the results showing the effects of specimen geometry, material, R-ratio, and test temperature will be presented in the form of log-log plots of AE rate in event count/sec. versus FCG rate in in./sec.

Figure 3.16 shows the effect of specimen geometry on AE behavior. Room temperature AE/FCG results for base metal specimens representing 2T-CT, SN, and SEN specimen geometries are plotted in Figure 3.16. It is evident from these data that specimen geometry did not affect the AE rate.

Figures 3.17 and 3.18 plot AE/FCG results for SEN and 2T-CT specimens segregated with respect to metallurgical condition. In Figure 3.17 results for four different material conditions are shown for SEN specimens tested at room temperature. Results for A533B base metal and weld metal are given along with A533B base metal that was plastically prestrained 3% prior to FCG testing. In addition, results for A106B piping steel are also plotted. It is evident from the data that very little difference was observed between A533B base metal and weld metal conditions. The 3% prestrained material on the other hand appeared to produce slightly less AE for the same FCG rate. The results for the A106B steel show significant scatter and, over the limited range of crack growth rates investigated, a clear correlation with FCG rate was not observed. It was noted, however, that nearly all of the A106B steel data was bounded by the 550°F trend line developed for A533B specimens (see Section 3.3.2). Thus, it is recommended that the relationship developed for A533B materials be used for evaluating flaws in A106B components and structures. The results for A533B base metal and weld metal 2T-CT specimens (Figure 3.18) tested at room temperature support the conclusion from the SEN specimen results that the AE results for unprestrained material are essentially the same for base and weld metal.

Figures 3.19 and 3.20 summarize the effect of R-ratio on the AE/FCG results for A533B base metal SEN specimens tested at room temperature and 550°F, respectively. Figure 3.19 also includes results for an SEN specimen tested under displacement control such that the stress intensity factor decreased with increasing crack length (data group between about 3×10^{-7} in./sec. and 2×10^{-6} in./sec.). The room temperature results suggested that increasing R-ratio should decrease crack closure AE during the decreasing portion of the load cycle. In contrast to the above results, limited 550°F data suggests that R-ratio had no significant effect on the AE behavior. Thus, it is difficult to draw definitive conclusions regarding the effect of R-ratio on AE during FCG.

Figures 3.21-3.23 give experimental data showing the influence of temperature on the AE rate. Figures 3.21 and 3.22 show results for A533B base metal and weld metal SEN specimens, respectively. Figure 3.23 plots results for A533B base metal SN specimens. The results displayed in Figure 3.21 indicate a clear effect of temperature on the AE rate. The results given in Figures 3.22 and 3.23 also show a trend towards decreasing AE rate with increasing temperature, although experimental scatter blurred the distinction

FIGURE 3.16. Effect of Specimen Geometry on AE Event Rate for A533B Specimens Tested at Room Temperature, $R=0.1$, Frequency=1-3 Hz

FIGURE 3.17. Effect of Material on AE Event Rate for Single-Edge Notch Specimens Tested at Room Temperature, $R=0.1$, Frequency=1-4 Hz

FIGURE 3.18. Effect of Material on AE Event Rate for 2T Compact-Tension Specimens Tested at Room Temperature, $R=0.1$, Frequency=3 Hz

FIGURE 3.19. Effect of R-Ratio on AE Event Rate for Base Metal Single-Edge Notch Specimens Tested at Room Temperature and Frequency=1-3 Hz

FIGURE 3.20. Effect of R-Ratio on AE Event Rate for Base Metal Single-Edge Notch Specimens Tested at 550°F and Frequency=2-3 Hz

FIGURE 3.21. Effect of Temperature on AE Event Rate for Base Metal Single-Edge Notch Specimens Tested at R=0.1 and Frequency=2-3 Hz

FIGURE 3.22. Effect of Temperature on AE Event Rate for Weld Metal Single-Edge Notch Specimens Tested at $R=0.1$ and Frequency=1 Hz

FIGURE 3.23. Effect of Temperature on AE Event Rate for Base Metal Single-Edge Notch Specimens Tested at $R=0.1$ and Frequency=1-2 Hz

between low and high temperature. The effect of temperature on the AE rate during FCG was the most pronounced effect observed during the laboratory testing phase of the program.

To evaluate the affect of pressurized 550°F water on the production of AE, an FCG test using a specially designed and built heavy-section cylindrical bend (HSCB) specimen was performed. The HSCB specimen, shown in Figure 3.24, was an 8-in. OD by 2-in. ID cylindrical section of A533B steel with 1-in. deep circumferential ID notch. The specimen was loaded in tension-tension bending at a cycle rate of 0.5 Hz and an R-ratio approximately equal to 1. Figure 3.25 shows a drawing of the test specimen welded to the loading frame. During fatigue cycling, the specimen was internally exposed to either RT air or 550°F water at 1350 psig.

The AE system used on the HSCB test was the same monitor system used on previous laboratory tests. It used zone isolation techniques to focus on AE events originating from the notched area of the specimen. For this test, three waveguide sensors were used to establish an area approximately 2 in. by 6 in. surrounding the notch from which AE signals were accepted.

Results from the HSCB test are plotted in Figure 3.26. These data show that introduction of pressurized high temperature water did not significantly affect the production of AE. The data in Figure 3.26 include both unfiltered and peak-time filtered information. The two curves are very similar during fatigue testing in RT air; but with the introduction of 550°F water, apparent abnormal signals are included in the data. Eliminating the long peak-time signals produced a data curve at 550°F that was consistent with the RT data. The source of the abnormal peak-time signals was not identified. The most significant result from this experiment was that AE from a growing fatigue crack was readily detected under conditions relevant to an operating nuclear power plant. Because of problems encountered with the design of the HSCB loading frame and because a more extensive evaluation of the affect of 550°F water was being planned for Phase II of the program (see Section 4), the HSCB test was terminated.

3.2.6 Stress Corrosion Cracking Tests

Stress corrosion cracking (SCC) tests were performed using 4-inch, Schedule 80, Type 304 stainless steel pipe specimens. These specimens were arranged with a flange on one end and a plug in the other end which fit into a loading clevis as shown in Figure 3.27. When the specimen was installed into the test frame, a bayonet fixture carried high-temperature water to the bottom of the specimen and then it returned along the specimen walls. During testing, the specimen was monitored for AE using waveguide AE sensors pressure coupled to the outer surface of the specimen and the assembly was insulated to maintain the desired temperature. The test weld in the middle of the specimen was fabricated with high heat input in an effort to sensitize the weld and promote SCC in the heat affected zone.

FIGURE 3.24. Heavy Section Cylindrical Bend Specimen and Electro-Discharge Machined Notch Dimensions

FIGURE 3.25. Test Specimen and Bending Equipment

FIGURE 3.26. Summation Acoustic Emission Event Count Versus Load Cycles for Fatigue Cycling of the Heavy Section Cylindrical Bend Specimen

FIGURE 3.27. Acoustic Emission/Stress Corrosion Crack Growth Test Arrangement

Some difficulty was experienced in generating SCC in the specimens. Test results shown in Figure 3.28, however, illustrate the SCC detection resolution attainable. The test was terminated shortly after the AE began to build up as shown in the AE vs. time plot. When the test specimen was cut open, the limited cracking that produced the AE was revealed as shown in the fractography. The primary shortcoming in the AE/SCC test results was the fact that they did not produce a calibrated set of data relating AE to crack growth rate due largely to the lack of a method for reliably measuring crack growth. This was needed to determine if the AE rate/crack growth rate relationship generated from fatigue crack growth was applicable to SCC.

AE monitoring techniques developed in this program were applied in another program supported by the Electric Power Research Institute to study SCC in 24-inch, Schedule 80, Type 304 stainless steel pipe with good results (Kurtz, Hutton, and Hall 1987). There was general correlation between AE crack growth but, again, the data was not sufficiently precise to permit developing a relationship between AE rate and crack growth rate.

Summarizing the AE/SCC test results, SCC is unquestionably detectable by AE, but a correlation has not been developed to specifically estimate SCC significance from AE data with confidence. The only relationship now available is based on fatigue crack growth. Current field application work at Limerick Unit 1 reactor (Philadelphia Electric Company) where a nozzle-to-safe end weld is being monitored for SCC growth during operation using AE may produce information that will support an AE/SCC relationship.

3.3 RELATIONSHIPS DEVELOPED

Section XI of the ASME Boiler and Pressure Vessel Code contains rules and procedures for performing pre-service and periodic in-service inspections of nuclear power plant components and for evaluating flaw indications detected during the inspections. The flaw evaluation procedures of Section XI for ferritic components are based on the concepts and principles of linear elastic fracture mechanics (LEFM). The potential application of AE to flaw assessment, described here, is also based on LEFM.

To relate AE obtained during reactor operation or hydrotest to fracture mechanics variables requires that experimental data exist for the relevant material and environmental conditions anticipated during pressure vessel service. The objective of the laboratory testing phase of this program was to generate AE data from fracture mechanics specimens tested under conditions relevant to reactor operation. Testing concentrated on characterizing the effect of metallurgical condition, flaw geometry, temperature, and mechanical loading on the AE response. Models for quantitative flaw evaluation during both hydrotest and operational conditions were then derived from the test data.

Dye Penetrant Indications

Crack Depth at Center of Indication

IGSCC on Inside of Pipe at End of Test

Specimen:

4-in. Sch. 80, Type 304 SS Pipe
Circumferential Weld
No Post Weld Stress Relief
Yield Strength at 550°F = 26,190 psi

All testing with internal water at 550°F and 1400 psig pressure

Unload-Reload: External load dropped to 2340 psi nominal stress - Hold for 10 minutes - Return load to 20,900 psi nominal stress

Test Sequence and AE Count

R9009131.1

FIGURE 3.28. Acoustic Emission/Stress Corrosion Cracking Test Results

FIGURE 3.29. Hypothetical Acoustic Emission Versus Test Pressure Curve for a Nuclear Pressure Vessel Hydrotest

3.3.1 Monotonic Loading Conditions

To relate AE to the stress intensity factor, K_I , during hydrotest requires that experimental AE- K_I data exist for the relevant material conditions anticipated during service. This implies that the influence of environmental and loading conditions upon the material--and, hence, the AE response from that material--are known. From the work presented in Section 3.2.2, it was observed that very little AE was obtained from laboratory-scale fracture mechanics specimens loaded monotonically to failure. Most of the AE occurred after net-section yielding. In contrast, larger laboratory specimens and intermediate-scale test vessels produced significantly more AE during the linear-elastic portion of the loading. The HSST vessel test data, while showing the positive potential for flaw evaluation by AE, represent a limited range of test conditions. Nevertheless, a concept for a quantitative flaw evaluation procedure during hydrotest can be developed from the vessel test data. Figure 3.29 gives a hypothetical AE vs. test pressure curve for a nuclear pressure vessel hydrostatic test.

The stress intensity factor resulting from a flaw stressed during hydrotest may be expressed as:

$$K_I = \beta_1 P_h \quad (5)$$

where P_h is the maximum hydrotest pressure, taken as 110 percent of the nominal^h reactor operating pressure, and β_1 is a constant that includes flaw

geometry, vessel geometry, and structural loading (e.g., conversion of pressure to stress) considerations. Equation (5) may be rearranged to yield:

$$K_I = \beta_1 P_o + \beta_1 (P_h - P_o) \quad (6)$$

where P_o is the nominal reactor operating pressure. The change in stress intensity factor due to the overpressure of the hydrotest is the second term in Equation (6) or:

$$\partial K_I = \beta_1 (P_h - P_o) \quad (7)$$

Hence, for a particular quantity of AE from a given location, the relationship between AE and K_I may be expressed as:

$$\partial N = f(\partial K_I) \quad (8)$$

where $f(\partial K_I)$ represents the mathematical relationship for the flaw severity-AE data shown in Figure 3.13. Over the range of expected K_I , this relationship may be taken as a linear function:

$$\partial N = \beta_2 \partial K_I \quad (9)$$

Substituting the right hand side of Equation (9) into Equation (7) yields:

$$\partial N = \beta_2 \beta_1 (P_h - P_o) \quad (10)$$

Solving for β_1 and substituting back into Equation (5) gives an expression that may be used to estimate flaw severity from AE data during hydrotest conditions.

$$K_I = \partial N / \beta_2 * \{P_h / (P_h - P_o)\} \quad (11)$$

The stress intensity factor represents a quantitative measure of flaw severity which may be used to calculate the degree of structural degradation. It may be used in a fatigue crack growth analysis to predict end-of-life for a given component or structure.

3.3.2 Cyclic Loading Conditions

In Figure 3.30 a composite plot of the AE/FCG data measured from laboratory specimens is presented. The principal conclusion from the laboratory testing phase of the program was that the AE event count rate (counts/sec.) depends primarily on the crack growth rate (in./sec.) and test temperature. The two parallel lines are conservative trend lines, which provide a lower bound on the room temperature and 550°F test data. These lines are represented by the following equations:

$$dN/dt = 1.6 \times 10^{-4} (da/dt)^{1.9} \text{ (RT)} \quad (12)$$

$$dN/dt = 2.1 \times 10^{-5} (da/dt)^{1.9} \text{ (550°F)} \quad (13)$$

FIGURE 3.30. Composite Plot of AE/FCG Data and Lower Bound Relationships for A533B Steel at Room Temperature and 550°F

These relationships may be used to predict flaw severity in terms of crack growth rate and thus estimate end-of-life for a component or structure. Determination of stress intensity factor range (ΔK) and crack size/shape from AE data is a much more difficult task, but it may be possible given additional information about the pressure vessel material chemistry and loading conditions.

Recent research work by Eason, Nelson, and Gilman (1990) on methods for estimating FCG rates for A533B and A508 steels in water environments indicates that FCG rates are influenced by the cyclic frequency of the loading, the flow velocity and temperature of the water in the test autoclaves, the sulfur content of the material, the R-ratio (K_{min}/K_{max}), and ΔK .

A format developed for displaying environmentally enhanced FCG data is the time domain format given in Figure 3.31. The abscissa of Figure 3.31 is the FCP rate in air divided by the load rise time (time period during which the load is rising). The ordinate in Figure 3.31 is the FCP rate observed in water, similarly expressed in the time domain. The range of FCP behavior can be represented as either low sulfur behavior or environmentally accelerated cracking (EAC). For steels with low sulfur content (i.e., ≤ 0.005 weight percent), the FCG rate in 550°F water is 1.65 times the FCG rate in air. For steels with sulfur content > 0.005 weight percent, and for water environments

FIGURE 3.31. Schematic Model of Corrosion Fatigue Crack Growth, Time Domain Form (Eason, Nelson and Gilman 1990)

FIGURE 3.32. Schematic Model of Corrosion Fatigue Crack Growth, Fracture Mechanics Form (Eason, Nelson and Gilman 1990)

containing higher than normal concentrations of oxygen, sulfate, and other impurities, a portion of the data exhibits significantly higher crack growth over a range of loading conditions. The range of EAC-susceptible loading conditions is defined by the Threshold, Second Slope, and Upper Slope lines shown in Figure 3.31. Note that the intersections between the Upper Slope and Low Sulfur lines, and Second Slope and Upper Slope lines, occur at fixed values of the base FCG rate. In addition, the position of threshold FCG behavior depends on the R-ratio and the load rise time. This information can also be converted to the more familiar fracture mechanics format as depicted in Figure 3.32.

Thus, if one assumes the FCG rate predicted from the AE/FCG relationships provide an estimate of the EAC growth rate, then an estimate of ΔK could be obtained from Figure 3.32 provided estimates of the load rise time and R-ratio are available. An estimate of the crack size/shape could then be acquired, given information on the applied and residual stresses for component or structure under consideration. Clearly, in order to perform an FCG evaluation of a flaw found by other NDE methods, the same information would be needed. Therefore, it appears feasible to derive estimates of ΔK and possibly crack size/shape from AE monitoring data, but prudence would dictate that use of the AE/FCG relationships in this manner should be limited until additional on-reactor monitoring experience is obtained to validate the models (see Section 5).

3.4 SUMMARY

Phase I of the program focused primarily on laboratory testing to evaluate relatively basic issues of the AE/flaw growth relationship. These included the effects of material condition, temperature, R-ratio, fatigue cycle rate, flaw type, etc. By design, the program being application oriented, it did not include any study of fundamental AE generating mechanisms. Phase I culminated in development of the AE/flaw relationships for data evaluation. Phase II which followed evaluated the relationships on a test vessel where reactor environment (except nuclear radiation) was simulated, thus bringing the work closer to the ultimate intended application.

4.0 PHASE II - EVALUATION BY VESSEL TESTING

Theoretical transfer of laboratory test results to a full-size structure is complex and uncertain particularly in the case of AE. Phase II of the program was designed to be an experimental transfer step to validate laboratory relationships on a structure wherein conditions could be controlled and adjusted to provide a simulation of a range of reactor monitoring conditions. Additionally, it was intended to identify limitations in the AE technology for structural monitoring and develop improvements to overcome these limitations. Phase II testing was performed on a German test vessel in collaboration with the Materialpruefungsanstalt (MPA) of Stuttgart, Federal Republic of Germany (FRG) over the period September 1982 to September 1983. The test and results achieved are documented in considerable detail in a topical report (Hutton, et al. 1985); therefore, Phase II is only summarized here to avoid needless duplication.

4.1 TEST CRITERIA

The objective of Phase II was to determine the effectiveness of laboratory-developed relationships and methods for monitoring flaws in reactor components by testing on a structure which simulated the conditions of monitoring a reactor pressure system. Additionally, the objective included identification of limitations in the AE technology for structural monitoring and development of improvements to overcome these limitations. Key considerations in the work were to evaluate:

- reliability of flaw detection and location
- ability to detect AE signals in the presence of reactor background noise
- separation of crack growth AE signals from similar acoustic signals produced by innocuous sources (slag inclusions, insulation rubbing, etc.)
- ability to interpret AE data in terms of source significance
- effect of temperature on detection of AE from crack growth

In support of accomplishing the above objectives, a criteria was established to guide selection of a test vessel and test content. The requirements were:

- Vessel wall thickness shall be greater than 4 inches.
- Vessel diameter and length shall be sufficient to test AE detection over a distance relevant to reactor component monitoring - at least (10 feet.
- Vessel material shall be representative of U.S. reactor pressure vessel steel - A533 Grade B, Class 1.

- Vessel shall be cyclically and monotonically loaded by means of internal pressurization with water.
- Installed flaws shall be exposed to pressurized water at temperatures up to 550°F.
- Cyclic stressing at R-ratios (minimum pressure/maximum pressure) ranging from 0.1 to 0.7 shall be employed.
- Nominal stress levels shall be representative of those in operating reactor pressure vessels.
- Innocuous noise sources representative of reactor environment (e.g., coolant flow noise, insulation rubbing, slag inclusion, base metal-to-cladding interface noise, and electrical transients) shall be simulated.

4.2 TEST FACILITY SELECTION

Four approaches were considered in selecting the test vessel and test site. These included:

1. Utilize existing test facilities at another national laboratory site (Oak Ridge National Laboratory) and adapt an HSST vessel to the program needs.
2. Install a test facility at PNL capable of performing the required testing and procure a test vessel from a commercial shop - probably in Seattle, Washington.
3. Install a test facility at PNL capable of performing the required testing and procure a test vessel (HSST vessel) from Oak Ridge National Laboratory.
4. Participate in vessel tests planned as part of the Federal Republic of Germany's program to investigate pressure boundary integrity issues. There were two options on a vessel - the HDR test vessel at Kahl or a vessel (ZB-1) to be tested at MPA in Stuttgart which was ultimately moved to the Grosskraftwerk Mannheim (GKM) facilities to achieve high-temperature testing.

The choices for a test facility were analyzed from the standpoint of test features available, timing, and estimated cost. On the basis of this analysis, the decision was made to participate in the testing of the German ZB-1 vessel at the GKM facilities in Mannheim, FRG. Two major factors which influenced the choice were that it was the only approach that facilitated testing at high temperature (550°F) and it was the least expensive. Figures 4.1 and 4.2 show the test site and the test vessel in the bunker. Arrangements were made under the NRC-BMFT Safety Research Exchange Agreement. German (MPA and GKM) cooperation and support throughout the test was outstanding and contributed significantly to the success of the test.

FIGURE 4.1. Vessel Test Facility at Grosskraftwerk Mannheim (GKM), Germany

FIGURE 4.2. ZB-1 Vessel in Test Bunker at GKM

4.3 TEST DESCRIPTION

The ZB-1 vessel, fabricated from A508 steel, was 67 in. O.D. and 148 in. long with a 4.7-in. thick wall. An A533B steel section 59 in. x 27.6 in. x 4.7 in. containing I.D. and O.D. machined flaws (the machined grooves were fatigue precracked prior to testing) was inserted in the vessel wall. A second insert, designated KS07C, fabricated from degraded A508 German steel was also installed. This insert was 59 in. x 27.6 in. x 4.7 in. and was placed in the vessel wall on the side opposite the A533B insert.

Other special features designed into the test were:

- a slag inclusion placed into the A533B insert weld
- an I.D. clad section intended to contain areas of underclad cracking and poor clad-to-vessel bonding.

The AE monitoring and analysis system consisted of a special data acquisition front end developed and fabricated by PNL, a PDP 11/23 computer for data analysis, a waveform digitizer and recorder, and a PDP 11/03 computer for waveform pattern recognition analysis. A special feature of the data acquisition front end instrument was a high density tape cartridge recorder with a 67-Mbyte storage capacity. This facilitates making a backup record of all raw data as well as performing on line data analysis. Figure 4.3 shows the instrument system located in a trailer outside the test bunker.

Waveguide sensors made up of 0.125-in. diameter by 23-3/4-in. long stainless steel wire with a piezoelectric crystal and amplifier integral to one end were used to accommodate the high temperature test conditions. Part of these sensors (marked Cylindrical 1, 2, 3, and 4 in Figure 4.4) were pressure coupled to the vessel wall as they would be for reactor monitoring. The waveguide sensors were tuned to operate at 400 kHz frequency. All of the AE sensors mounted on the vessel appear in Figure 4.4.

A special set of commercial sensors with peak responses at 100 kHz were installed on the vessel surface for each hydrotest. The intent here was to provide a monitoring regime similar to what would normally be applied in a preservice hydrotest.

All sensors were characterized at the time of installation and periodically during the test.

Crack-opening-displacement (COD) gauges were installed on the three machined defects in the A533B insert. The smaller of the flaws was designated Flaw A and was located on the insert I.D. The middle sized flaw (Flaw B) was also on the I.D. Flaw C, which was the largest flaw, was located on the insert O.D. The flaws were sized to produce a wide range of flaw growth rate under the cyclic load test conditions. The COD gauges provided a continuous measure of crack growth which was then related to the AE information. Instrumentation located on the inside of the vessel is shown in Figure 4.5.

FIGURE 4.3. ZB-1 Vessel Test Instrumentation in Trailer at GKM Test Site

FIGURE 4.4. ZB-1 Test Vessel with AE Sensors in Place

**ZB-1 TEST VESSEL
INSTRUMENTATION INSIDE VESSEL**

FIGURE 4.5. Instrumentation Mounted on Inside Wall of ZB-1 Test Vessel

The ZB-1 vessel test plan was designed to provide a variety of test conditions (loading, temperature, R-ratio, noise sources, etc.). Water was used as an internal pressurizing medium to stress the vessel in both static and cyclic loading. The test matrix consisted of five hydrostatic tests separated by blocks of cyclic loading which total 28,000 cycles. Testing temperatures of 150°F and 550°F with an operating pressure of 250 bar (3675 psig) were used. During high temperature testing, the vessel was covered with a blanket insulation.

Cyclic loading was tension tension with planned changes in R-ratio. Cycle rate varied from a maximum of 1 cycle/78 seconds during 150°F testing to a minimum of 1 cycle/120 seconds during 550°F testing.

4.4 MAJOR TEST RESULTS

4.4.1 Hydrostatic Test Results

Six hydrostatic tests were performed as follows.

<u>Step</u>	<u>Overpressure, %</u>	<u>Temperature, °F</u>
1	0	150
4	2	150
7	4	150
9*	10	460
10	15	150
13	36	150

*Not planned as part of the original test matrix.

The AE results from the six hydrotests performed in the course of the ZB-1 test led to the conclusion that the effectiveness of AE monitoring during inservice hydrostatic testing to 1.15 x operating pressure is very questionable. Steps 4, 7, 9, and 10 produced very few AE indications identifiable with the known flaws. Without prior knowledge of the existence of the flaws, it is very doubtful that the indications would have alerted anyone to the presence of a flaw. In Step 13, AE was identified with Flaw B; however, the flaw was 83% through-wall at that time and the overpressure used was 1.36 x operating pressure. The stress intensity level required to activate Flaw B was about 70 ksi/ $\sqrt{\text{in}}$. Flaws A and C barely exceeded this level during the Step 13 hydrotest. Since no notable AE was detected for any of the flaws below this stress intensity level, small flaws do not appear to be detectable at low overpressure levels during an inservice hydrotest. As the level of inservice hydrotest overpressure increases, the flaw detection resolution could be expected to increase.

In a different context, the ZB-1 hydrotest produced results which support the validity of using AE monitoring to detect flaws in a vessel or pipe during a hydrotest when the container has not been previously stressed to the level produced by the hydrotest. This would apply to tests such as acceptance pressure testing of a new vessel. Step 1 was made up of two parts. Part 1 was performed with the original KS07 insert of degraded German pressure vessel steel in place. Part 2 was performed after the original KS07 was replaced with KS07R which was a sound piece of German pressure vessel steel. During Step 1, Part 1, AE showed strong evidence of a flaw in a weld across the middle of the KS07 degraded insert. This was subsequently shown to be correct by physical testing after the insert was removed from the vessel. Step 1, Part 2, performed after the KS07R insert was installed showed AE evidence of a flaw in the installation weld. Under subsequent cyclic loading, the flaw developed to the point that the vessel test was stopped somewhat short of the originally planned test format. Based on post-test examination of the ZB-1 vessel, it was estimated that the flaw in the weld (a crack) was about 0.20-in. deep at the time Step 1, Part 2 was performed. The significant element in this latter consideration is the fact that the flaws detected had been formed outside of the service environment and had not been previously stressed to a level similar to that produced by the hydrotest.

4.4.2 Special Test Features

There were several special features of the test. These features included:

- flow noise simulation to test the ability to detect AE in the presence of reactor coolant flow noise
- slag inclusion to evaluate separation of slag cracking from metal cracking
- weld cladding to evaluate detection of underclad cracking
- nozzle cracks to test flaw detection in a difficult geometry
- blind test to test detection and identification of an unknown flaw.

An electronic transducer used to inject noise simulating coolant flow noise was effective in testing the influence of reactor coolant flow noise on detection of AE from crack growth. It had little effect on AE detection with sensors tuned to 400 kHz; however, low frequency (150 kHz) sensors were saturated. The results were consistent with subsequent data obtained during hot functional testing at Watts Bar Unit 1 nuclear reactor in indicating that the flow noise is a manageable problem.

A slag inclusion placed in the A533B insert weld partially accomplished its purpose. It indicated that a small slag inclusion (4 x 0.12 in.) should not interfere with AE detection. Due to the small size, however, essentially no AE was detected which precluded assessing the similarity between AE which might be produced by a major slag inclusion and that from crack growth.

Weld cladding did not fulfill its intended purpose of evaluating AE detection of possible noise from defective cladding areas and detection of underclad crack growth due to the lack of flaws. Detection of underclad cracking should not be a problem, however, since the AE signals would travel through the base metal to the AE sensors and would not involve the highly attenuative cladding.

Cracking in the bore of blind nozzles on the vessel was not clearly indicated by AE. This, however, appears to be due to a unique geometry effect wherein the discontinuity of the nozzle bore interfered with propagation of signals to some of the AE sensors installed to monitor the vessel cylinder.

Unplanned cracking in the KS07R insert installation weld (Figure 4.6) provided an effective "blind" test of AE detection of crack growth which gave very positive results. AE detected and located the growing crack significantly before it was confirmed by ultrasonic testing. In Figure 4.7, a sample of AE source location data shows the AE response produced by the weld crack.

ZB-1 Vessel Test KS07R Insert Weld Crack

FIGURE 4.6. Unplanned Crack in KS07R Insert Weld in ZB-1 Test Vessel

AE Source Location, Step 9, Array 3 (65-80 LP), ZB-1

FIGURE 4.7. AE Source Location Data Indicating Growth of Unplanned Crack in Fabrication Weld - ZB-1 Test Vessel

4.4.3 AE Monitoring at 150°F

The first two-thirds of the ZB-1 vessel cyclic loading test was performed at 65°C (150°F). This consisted of four elements designated as Steps 2, 3, 5, and 6. The cyclic loading steps were alternately low (0.1) and high (0.6) R-ratio to produce crack front marking between steps as a result of the different crack growth rate. The maximum pressure for all low-temperature cyclic loading was 240 bar (3530 psig).

AE from growth of Flaws A and B was detected by both the AE array (Array 2) on the A533 insert and the AE array (Array 3) monitoring the vessel cylinder. The crack growth AE assumed consistent characteristics of high amplitude and of occurring at or near peak load.

Noise from a manhole cover assembly was an intermittent problem throughout the low temperature testing. Generally, this noise could be handled by filtering with parametrics (load position and signal amplitude). This, however, underscores the need for real-time AE identification by pattern recognition to isolate crack growth AE. Generally, parametrics such as load position in particular will be of little use in a general reactor monitoring situation.

There was some evidence of the KS07R weld crack becoming active during Step 6. The indications were not sufficiently prominent to draw attention at that time, however.

Location of AE sources was not always as precise as desired. The accuracy of point source location using a two-dimensional concept is generally considered to be within 2T. On a heavy section, complex structure, this uncertainty can be quite worrisome. This is an aspect of established AE technology that needs to be improved.

The need to exercise special care in mounting pressure-coupled waveguide AE sensors was evident. Performance of Array 3 (pressure coupled) was marginal until the waveguide-to-vessel interface pressure was corrected. This resulted in a significant improvement in array performance.

4.4.4 AE Monitoring at 550°F

High-temperature - 550°F (288°C) - test conditions were imposed during the last one-third of the cyclic loading. This constituted Steps 8, 9, and 11 for a total of 7300 load cycles. R-ratios of 0.4 and 0.6 were applied alternately during this period. The maximum cyclic pressure was 250 bar (3675 psig).

The dominant sources of acoustic emission during the high-temperature portions of the test were manhole noise during Step 8, and KS07R weld cracking during Steps 9 and 11. Flaw growth activity from A, B, and C was detected by all arrays during Steps 8 and 9, but was overshadowed by the large amount of activity from the KS07R weldment during Step 11. In the hot phase of the

test, Array 3 was most effective at detecting and locating activity from the KS07R weld cracking and least effective at detecting events from Flaws A, B, and C under the given circumstances. Overall, Array 3 (cylindrical array) performed better than Arrays 1 and 2 (quad arrays). The latter two showed a tendency to mislocate data originating outside of the array to show it inside the array. This was not the case with Array 3. The amplitude and load position characteristics of the AE signals from Flaws A, B, and C were similar to signals detected during 150°F testing. Signal attributes for events from the KS07R weld cracks were different than for Flaws A, B, and C. The differences in signal amplitudes were probably in part due to the large source-to-sensor distance. The substantially higher event rates from the KS07R weld cracks may have resulted from differences in material chemistry, heat treatment, and flaw geometry.

4.4.5 AE/Flaw Growth Interpretation

Flaw growth characteristics for Flaws A, B, C, and KS07R weld cracking have been determined from NDE, crack-opening-displacement, and fractographic measurements. This information was used to establish crack growth rates for comparison with predictions derived from the AE/flaw severity relationship. A final compilation of crack growth calculated from AE vs. measured crack growth is given in Figure 4.8. In view of the time period and the amount of AE data involved, the results are quite good.

FIGURE 4.8. Accuracy of Crack Growth Rate Estimate on AE Data - ZB-1 Test Vessel

The variation of predicted crack growth rates from the actual was within the variation observed during laboratory testing. Figure 4.8 shows that the model underpredicted the crack growth rate for Flaw A at 150°F test temperature. When the relationship was applied to 550°F data, the crack growth rate was consistently overpredicted by the near-field array as was expected. The far-field array results were not conservative in that the crack growth rate was underpredicted in half the cases. Clearly the source-to-sensor distance was largely responsible for this effect. It is significant to note that the relationship tends to compensate for the range of absolute number of AE signals detected by the two arrays for the same two installed flaws.

A test of the relationship on the weld cracking in the KS07R replacement patch was attempted and indicated a significant overprediction of the crack growth rate. The apparent cause is the large amount of crack interface noise produced as a result of the complex crack geometry.

The AE rate-crack growth rate relationship developed is considered to be useful for approximation at this time. It needs refinement but this can be accomplished most effectively through analysis of data obtained from reactor monitoring.

4.4.6 Crack Growth AE Signal Identification

One of the key requirements of this program was to develop a method of separating AE signals produced by crack growth from acoustic signals produced by innocuous sources. Signal pattern recognition using signal processing techniques to generate features related to spectral content (P. G. Doctor, et al. 1979) was chosen initially. This did not produce satisfactory results during the test. A second set of features consisting of the parameters of a 10th order autoregressive model of each waveform was developed out of analysis of initial ZB-1 data. This approach showed some improvements in identification of acoustic signals from the ZB-1 test, but the 76% correct classification still did not produce the performance level considered necessary ($\geq 90\%$ correct classification).

In the course of reassessing the problem, waveform by waveform source location made possible the identification of a consistent three-pulse signal pattern (Figure 4.9) present in practically all waveforms originating near known and suspected regions of crack growth. This is a function of the waveguide sensor response to excitation by a fast rise time, short duration signal. A signal classification method developed around this phenomenon has produced about 95% correct classification of recorded acoustic signals examined. This meets the required level of accuracy for an effective working tool.

FIGURE 4.9. AE Signal Identification by Three-Pulse Waveguide Sensor Response

4.5 CONCLUSIONS

The ZB-1 test results provide strong support to the feasibility of using AE for surveillance of flaws in reactor pressure boundaries. Key points of the test results are highlighted below.

The ability of an AE monitor system to perform over a long period under adverse conditions was demonstrated.

Generally, detection of AE from growing flaws was good at all stages of the test. One of the major achievements was detection of an unexpected crack growing in the KS07 replacement weld. This was clearly detected by AE well before it was confirmed by ultrasonics. Some of the AE sensors were 10 feet from the source.

Analysis of the test data to identify AE signals from crack growth has underscored the need for a method to isolate crack growth AE signals in real time. The pattern recognition algorithms applied for this purpose during the test performed discrimination correctly for many of the signals, but the overall performance was not acceptable. Post-test analysis of recorded waveforms, however, produced an approach based on the unique response of the

waveguide sensor to the fast rise time, short duration AE signal which achieved a correct classification level of about 95% for the ZB-1 data.

A test of the flaw evaluation model was performed. The relationship underpredicted the crack growth rate at 150°F test temperature and overpredicted the crack growth rate at 550°F test conditions. The high-temperature results were within the expected range based on the observed variations in laboratory data. The low-temperature results were below the expected range. This may be due to a change in the power-law relationship at low crack growth rates and/or that the relationship based on 70°F data should be corrected for use on 150°F data.

A test of the relationship on weld cracking in the KS07R replacement patch was attempted and indicated a significant overprediction of the crack growth rate, which is felt to be a function of the large amount of crack interface noise included in the data.

Results from the hydrostatic tests lead to three conclusions.

- For preservice hydrotesting to detect flaws where the vessel has not been previously stressed to operating conditions, AE monitoring should be an effective technique to detect and locate flaws because the flaws and surrounding material will be experiencing a new level of stress.
- Application of AE monitoring to inservice hydros limited to an overpressure of 1.15 x operating pressure appears to be of questionable effectiveness.
- In the case where higher hydro overpressures are used (i.e., in the order of 1.4-1.5 x operating pressure), AE monitoring appears to offer a usable degree of flaw detection resolution.

The AE event rate at a constant crack growth rate was not dependent upon temperature.

The results from the ZB-1 test showed that the AE technology developed under this program constitutes a tool that can be effectively applied to continuous monitoring of reactor components to detect cracking now. The primary refinement recognized at this time which would significantly simplify application and increase reliability is development of a sensor which will faithfully reproduce the waveform of the acoustic energy pulse propagated through the material. This would not be a trivial undertaking. It would require a nonresonant device with sensitivity at least equal to the current sensors and capable of withstanding the reactor environment. Such a sensor would facilitate improved source location and signal identification. Again, lack of such a sensor does not preclude useful application of the AE technology at this time, but it is an improvement needed to enhance the effectiveness and maturity of the technology.

Overall, the ZB-1 test is considered to be an unquestionable success. It provided valuable information to support continuous AE monitoring for flaw detection, and it showed limitations which need to be observed in application of the technology and ultimately eliminated. Much of this information could not have been practically obtained in any way other than a test such as ZB-1.

5.0 PHASE III - VALIDATION ON REACTOR SYSTEMS

Performance of the Phase II vessel test represented a limit of technology demonstration that could be effectively performed outside of the intended application environment; i.e., an operating reactor. To do further off-reactor testing would, for the most part, simply duplicate previous work. Although the vessel test attempted to simulate reactor conditions, it is impossible to literally duplicate all of the reactor monitoring environment such as background noise from coolant flow, radiation environment, containment penetrations as a noise source, the many sources of transient electrical signals, etc. It was recognized that the importance of the planned application of AE monitoring to reactor systems required that the technology validation be carried the final step of demonstration on an operating reactor(s) to complete the objective of the program.

5.1 APPROACH

A two-part approach to on-reactor validation of the AE technology evolved. One consisted of installation of AE monitoring equipment as part of a reactor construction phase (Watts Bar Unit 1) which facilitated monitoring pre-startup testing to provide baseline data for the reactor components being monitored and evaluating AE system response to coolant flow noise under a circumstance where the sensors inside of containment could be accessed during the test. The AE system will remain in place to monitor the same components during reactor operation.

The second part of the approach involves installation of AE monitoring equipment to monitor a specific flaw indication identified during inservice inspection (ISI) of an operating reactor (Limerick Unit 1) to determine if the flaw grows during subsequent reactor operation. Detection of flaw growth and an estimate of growth rate which is verified by independent means would provide the final validation; i.e., the technology is capable of detecting, locating, and evaluating flaw growth at the time it occurs during reactor operation. Data on AE from stress corrosion cracking and fatigue cracking from an operating plant is key to validating the technology developed under this program.

5.2 WATTS BAR UNIT 1 REACTOR

Discussions with Tennessee Valley Authority (TVA) technical staff members relative to the need for demonstration of AE technology on an operating reactor led to an arrangement for installing AE instrumentation on TVA's Watts Bar Unit 1 reactor. The locations monitored are illustrated in Figure 5.1. Inlet nozzle No. 2 was chosen because preservice ultrasonic examination had revealed small indications in the nozzle due to underclad cracking. The larger defects were repaired and only defects allowable under ASME Section XI Code remained in the nozzle. The section of vessel wall was monitored to help assess the feasibility of AE monitoring the vessel belt line fabrication weld with sensors at the closest readily accessible locations on

AE SENSOR LOCATIONS-WATTS BAR UNIT 1

FIGURE 5.1. Areas Monitored for AE at Watts Bar Unit 1 Reactor

the vessel. Sensors were installed on the No. 2 cold leg safety injection line for the purpose of monitoring a fracture specimen attached to the cold leg to demonstrate detection of AE in the presence of reactor background noise.

5.2.1 Objective

The rationale in installing the AE system during construction, aside from avoiding installation on a radioactivity "hot" system, was to monitor the preservice tests to establish baseline AE data which would then provide a reference for inservice monitoring. This is particularly significant in the case of the No. 2 inlet nozzle which had showed defect indications in preservice ultrasonic examination.

5.2.2 Cold Hydrostatic Test

The initial instrumentation was installed in 1981. The waveguide sensors were attached with magnets and temporary signal cables were run from the sensors to a location outside of containment. The work was performed under Contract B-D1803-A-L with TVA where the cost of the work was paid out of the NRC AE Program. This system was used to monitor the cold hydrostatic test of the pressure system in October 1981. Results from the test monitoring are documented in detail separately (Hutton et al. 1982). In summary, the results showed:

- AE produced by crack growth in a specimen attached to the safety injection pipe to No. 2 cold leg was detected during reactor testing.
- Spontaneous AE was detected from the No. 2 inlet nozzle during final pressurization.
- Preservice NDI information showed an indication (acceptable under the Code) which matched AE source indications.
- Evaluation of the flaw in the No. 2 inlet nozzle using an AE interpretation relationship agreed within a factor of two with an evaluation by ASME Section XI Code procedures.

Some areas of AE technology where the need for improvement was seen at that time were:

- Improved control of interference from background noise.
- Substantiation of AE sensor array configurations that are most effective in monitoring known areas of interest.

5.2.3 Hot Functional

After completion of the cold hydrostatic test, a second contract (B-D7540-A-L) was negotiated with TVA in 1982 to prepare for AE monitoring the same locations (Figure 5.1) during hot functional testing. In hot functional testing, all aspects of reactor operation are present except for the fuel load; orifices are used to simulate the coolant pressure drop across the fuel load. Hot functional testing was, therefore, an outstanding opportunity to evaluate ability to overcome background noise interference, AE system operation at reactor operating temperature, and ability to detect AE from cracking in a fracture specimen in the presence of the on-reactor environment. AE system installation and test results are discussed in detail in a topical report (Hutton et al. 1984) so the test will only be summarized here.

Since many of the AE sensors installed for cold hydrostatic testing disappeared during the subsequent construction work, it was necessary to install new sensors for hot functional testing. The sensor mounting fixtures were strapped in place using stainless steel straps where possible and bracketed to adjacent structure where banding was not practical. The signal cables from the sensors to containment penetration were enclosed in metal conduit to meet operational requirements. The AE monitor instrument was located in the auxiliary instrument room (a special cabinet was installed) and signal cabling was run from the outside of the containment penetration to the auxiliary instrument room.

Hot functional testing at Watts Bar Unit 1 was performed in July 1983. The test was AE monitored continuously during the coolant temperature/pressure from 150°F, 400 psig to 557°F, 2235 psig over the period July 2-12. The AE system was left on standby with the reactor at temperature until July 30 at which time it was reactivated to assure that the response was still consistent with the initial response at operating temperature (it was).

Summarizing, the results obtained from AE monitoring hot functional testing were very significant to the objective of continuous AE monitoring. Reactor system temperature and coolant flow noise were similar to that during reactor operation. Under these conditions, the following major items were demonstrated:

- Coolant flow noise can be overcome as a problem to on-line reactor monitoring. Current on-line monitoring at Limerick Unit 1 reactor has verified that this finding is true.
- AE signals from a fracture specimen mounted on the No. 2 safety injection pipe were detected under operating conditions.
- Spontaneous acoustic information was detected and located on the No. 2 inlet nozzle.
- Sensors and preamplifiers were not adversely affected by the exposure to the high-temperature environment.

5.2.4 AE System Calibration On-Reactor with Simulated AE

Delay in the startup of Watts Bar Unit 1 generating station resulted in a period of time during which the vessel was completely empty. This offered an unusual opportunity to enter the vessel and inject simulated AE signals at selected locations to calibrate the source location accuracy of the AE system. Such a calibration is important because the algorithms available at this time for practical application to AE monitoring on a reactor locate in two dimensions (X and Y) on the surface of the structure containing the AE sensors. On components where the wall thickness is relatively small, such as 2 or 3 inches on piping, this is not a serious concern. Considering, however, the circumstances of AE sensors mounted on the outer surface of a reactor vessel or nozzle sensing a growing crack at the inner surface of the structure which may have an 8 to 12 inch wall thickness, there is obviously a sizeable third (Z) dimension involved. The question being addressed is, "Does the two-dimensional source location operating on the outer surface of a heavy wall structure provide an acceptable indication of the location of the AE source?". Again, the test procedure and results' details are reported in a separate document (Hutton et al. 1988); hence, they are only summarized here.

Referring to Figure 5.1, the test work focused on two areas; the inlet nozzle and the section of the vessel wall. Signal attenuation by the vessel cladding and the clad-to-vessel interface in the 400 to 500 kHz frequency range being used to monitor for AE precluded use of the normal pencil lead

break signal source to inject signals at the inside of the vessel and nozzles. In real application of AE monitoring, this attenuation problem would not be present because one would be seeking to detect signals originating in the ferritic vessel wall. An electronic pulser with a transducer was used to inject acoustic signals at known points around the inside of the #2 inlet nozzle and on the inside surface of the vessel.

Signal source location for signals injected on the ID of the #2 inlet nozzle was accurate to the projected signal input location within 3 to 5 inches in the circumferential direction. In the front-to-rear direction, the location was accurate to within 5 to 7 inches for signals originating within the region under the AE sensor array, but the front-to-rear accuracy deteriorated progressively as the source point moved farther outside of the AE sensor array. There are methods of overcoming the front-to-rear location inaccuracy with additional sensors; however, it may be difficult to justify the additional hardware for the increase in source localization accuracy.

No usable data was obtained on the vessel wall. Signal attenuation in the cladding and at the clad-to-vessel wall interface was large enough to prevent proper detection of the injected signals at all four AE sensors mounted on the vessel wall. The fact that these test signals were not detected by all four sensors does not imply inability to detect AE signals originating from crack growth in the ferritic vessel wall in the belt-line region. It has been demonstrated in the intermediate scale vessel (ZB-1) test performed in Germany (Hutton et al. 1985) and in tests performed on an uninstalled reactor vessel at Hanford (Hutton et al. 1984, pp. 18-19) that AE signals originating in the ferritic vessel wall can definitely be detected at a distance of at least 15 feet using pressure-coupled waveguide AE sensors of the type installed on the Watts Bar vessel. The Watts Bar test result does, however, point out the need for more flexibility in the source location process. The AE system currently requires that a signal be detected by all four sensors in an array before it will be processed for location. In this instance, many of the injected signals were detected by two or three of the sensors in the array which is sufficient to perform a source location determination even though it may be less than optimum in definition.

The test results demonstrated that acceptable two-dimensional location of an acoustic signal source can be achieved on a reactor structure even in the relatively complex geometry of the nozzle-to-vessel joint.

5.2.5 Operational Monitoring

Plans still include AE monitoring of the locations shown in Figure 5.1 during operation of Watts Bar Unit 1 reactor. The waveguide AE sensors and preamplifiers plus the cabling from the sensors to the preamplifiers outside containment and on to the auxiliary instrument room where the instrument rack for the monitor instrument is located remain in place. Since the startup of Watts Bar Unit 1 has been significantly delayed (fuel loading currently projected to be late 1991), the AE monitor instrument has not been installed

but it is on-hand. Performance of the work will be by arrangement with the Tennessee Valley Authority.

5.3 LIMERICK UNIT 1 REACTOR

In the course of normal inspection performed at Philadelphia Electric Company's (PECO) Limerick Unit 1 reactor during the 1989 refueling outage, it was found that the recirculation nozzle-to-safe end weld VRR-1RD-1A N2H showed a flaw indication. The indication appears to be an intergranular stress corrosion crack 7 inches long with an average depth of 0.25 inches. It is located between 31.8 and 38.8 inches (257° to 314°) circumference measuring clockwise from top-dead-center looking with the flaw. The flaw indication was evaluated for continued service of the piping using fracture mechanics methods and a crack-arrest-verification specimen has been installed for assurance that the indicated flaw would not grow undetected. In addition, PECO elected to apply continuous AE monitoring to validate a direct method of determining if a flaw indication is growing during reactor operation. This project provides a long sought opportunity to validate AE monitoring technology developed under the NRC-sponsored AE/Flaw Characterization Program.

5.3.1 Objective

The objective of AE monitoring the nozzle-to-safe end weld at Limerick Unit 1 is to validate the AE technology developed for continuous monitoring of reactor components to detect and locate crack growth. Results from AE monitoring over a fuel cycle (≈18 months) will be compared with indications from ultrasonic examination of the weld at the end of the current fuel cycle for correlation.

5.3.2 Arrangements with PECO

PECO is funding the AE monitoring effort at Limerick Unit 1, and they also provided staff to assist with installation of the AE system and to perform routine surveillance of the AE system (monitor the system for correct operation, change out data tapes, send data tapes to PNL for analysis, etc.). In support of the project, NRC agreed to the use of an NRC-owned AE monitoring system which was developed under the AE/Flaw Characterization Program. PNL is performing data analysis and reporting results to PECO. NRC Research is also receiving the reports in return for use of the NRC-owned AE monitoring system.

5.3.3 Installation

Preparation and installation of the AE system at Limerick followed the requirements of the current version of a Code Case titled "Acoustic Emission for Successive Inspections, Section XI, Div. 1," which is in the ASME Code approval process.

As the first step in preparation, existing three-foot-long waveguide AE sensors were modified to provide nine-foot-long waveguides by welding on additional lengths of 0.125-inch diameter Type 304 stainless steel wire.

FIGURE 5.2. Sensitivity Test of Waveguide AE Sensors on Calibration Block

Construction of the waveguide AE sensors is shown in Figure 6.5. The AE system was then assembled in the laboratory at Battelle, and the sensors were mounted on a 4 in. x 12 in. x 12 in. steel calibration block one at a time to calibrate the sensor response characteristics and sensitivity. The sensors were pressure coupled to the surface of the block in a manner similar to that used to mount them on a reactor. A broad band acoustic signal was then generated by impinging a helium gas jet from a 30-psi source through a #18 hypodermic needle onto the surface of the block with the needle held 1/8 inch above the surface of the block and 1-1/2 inches from the sensor waveguide tip. The resulting spectral responses shown in Figure 5.2 are a logarithmic measure. The analyzer reference point is the top of the grid which corresponds to 0 dB on the logarithmic scale. Working from the top down, the logarithmic scale is in dB below reference (10 dB/division). The analyzer reference value is 1 milliwatt when a 50-ohm input is used. The corresponding voltage values can be determined by:

$$V_{\text{peak}} = \frac{0.244}{[\log^{-1}(\text{dBm}/20)] \times 0.707}$$

Thus, on a voltage scale, the reference corresponds to 317 millivolts peak with the analyzer on the 0 dB scale.

Figure 5.2 shows that the peak response of the sensors occurs at 400 to 425 kHz and that the sensitivity to the helium gas excitation is 0.08, 0.18, 0.18, and 0.11 millivolts_{peak} for sensors #1, #2, #3, and #4 respectively at the sensor output taking into account the 40 dB external gain used to produce the traces. The frequency range of the peak response and the overall response profile are very important to the resistance of the AE system to interference from noise such as coolant flow noise. Sensors with a response profile similar to those seen in Figure 5.2 have been used successfully in the presence of flow noise on a PWR, but there has not been the opportunity to test them previously on a BWR. The sensor sensitivity is acceptable particularly for nine-foot-long waveguide AE sensors.

The performance of the monitor system minus the sensors was checked using an Acoustic Emission Simulator, Model No. AES-1, from Acoustic Emission Associates, Laguna Niguel, California 92677. This is a special-purpose signal generator which facilitates imposing a time delay between similar signals out of two different channels to qualify Δ time determination by an AE monitor instrument. This, in turn, influences the signal source location capability of the instrument. For input delays of 100, 150, and 200 μ seconds, the measurement accuracy was within $\pm 2\%$. For inputs of 25 and 50 μ seconds, the measurement accuracy was within $\pm 5\%$. These input time delays represent the range of values expected to be significant in this test. The accuracy with the larger time delays is quite good, but the accuracy with the smaller time delays is marginal because the inherent measurement error is a larger portion of the total time delay.

The signal identification module was evaluated in accordance with the method described in ASME Code Case N-471, "Acoustic Emission for Successive Inspections, Section XI, Div. 1." This calls for installation of a representative waveguide AE sensor on a calibration block, connecting the sensor to the monitoring system, and exciting the sensor ten times by each of the following three methods:

1. Fracture a 0.3-mm, 2H pencil lead against the surface of the block in accordance with ASTM E976.
2. Strike the surface of the block with a 0.25-inch-diameter steel ball dropped from a uniform height.
3. Inject a multi-cycle (five cycles minimum) burst signal into the block with a transducer and a waveform generator.

The requirement in the Code Case is that the signal identification module flag at least 8 out of 10 lead fracture signals as crack growth AE signals and at least 8 out of 10 of the other types of signals as not being crack growth AE signals. The results obtained in this evaluation were:

- Pencil lead break -- 10 out of 10 identified as crack growth signals
- Ball drop -- 6 out of 10 identified as not being crack growth signals
- Pulser -- 10 out of 10 identified as not being crack growth signals

In the actual AE system installation, the initial emphasis was focused on installing the AE sensors on the pipe, installing signal cable from the sensors to a containment penetration where the leads could be connected to leads through the containment wall, and calibration of the installed sensors. All of this work required entry to the drywell. Installation was started March 22 and proceeded as a joint effort by PECO and Battelle personnel. The installed AE system is shown in Figures 5.3 through 5.10.

In Figure 5.3, the specific location of the AE sensors and the flaw indication are shown. Location of the flaw indication is taken from ISI report #89-003. Figure 5.4 shows the waveguide sensors mounted on the nozzle and safe-end. A force of at least 30 pounds is applied to the waveguide with the pressure bolt which is threaded into the mounting bracket (Figure 5.5). Since the waveguide is tapered from 0.125-inch diameter to 0.050-inch diameter at the tip, the 30 pounds force results in about 15,000 pounds per square inch interface pressure between the waveguide tip and the surface of the pipe. Experimental evaluation of interface pressure vs. acoustic coupling efficiency

FIGURE 5.3. Location of AE Sensors on Pipe Weld VRR-1RD-1A N2H - Limerick Unit 1

FIGURE 5.4. Waveguide AE Sensors Mounted Near Weld VRR-1RD-1A N2H - Limerick Unit 1

FIGURE 5.5. Waveguide AE Sensors Being Pressure Coupled to the Pipe Surface - Limerick Unit 1

FIGURE 5.6. Electronic Pulser Mounted Near AE Sensor No. 3 - Limerick Unit 1

in the 300 to 500 kHz frequency range has shown that 15,000 pounds per square inch is effective. Some improvement can be achieved with higher pressure, but it becomes a trade-off between the difficulty of achieving higher interface pressures vs. the limited gain to be had. An Inconel spring is included in the line-of-force to help maintain a uniform pressure in the presence of temperature changes. A high-temperature electronic pulser is mounted in the vicinity of the AE sensors (Figure 5.6) to provide a means of qualitatively testing AE sensor sensitivity during reactor operation.

As shown in Figure 5.7, after the waveguides exit the opening in the shielding wall, the outer end is supported in a location away from the opening. This removes the critical sensor head, which contains the piezoelectric crystal and a 20 dB gain amplifier, from the area of high temperature and possible neutron beams through the shielding annulus. It is important that the sensor head be maintained at no more than 200°F and away from neutron beams.

PECO personnel installed RG58 coaxial cables from the AE sensor heads to a containment boundary penetration where the wires were connected through a Cannon plug to two-wire pairs which were installed in the penetration plug. The RG58 cable picked up again at the outer end of the penetration plug and went to the preamplifiers (Figure 5.8) and to the AE monitor instrument (Figure 5.9). The AE monitor instrument is installed in the same cabinet with

FIGURE 5.7. Modified Waveguide AE Sensors Emerging from Weld Region - Limerick Unit 1

the crack-arrest-verification specimen instrumentation. A set of differential amplifiers has been added between the preamplifiers and the AE monitor instrument to help overcome a noise problem.

After the AE system is installed, there are two qualification steps to be performed. One evaluates the signal detection and source location function of the total system and the other evaluates the response sensitivity of the installed waveguide AE sensors.

The signal source location results obtained from cracking 0.5-mm pencil leads on the pipe surface to simulate an AE signal are shown in Figure 5.10. This is an accepted method of simulating an AE signal, and it is easily used in a field circumstance. Figure 5.10 is a roll-out of the pipe section and shows the AE sensors (black squares) plus the signal source determination (letters). The axial spacing of the sensors is 3-1/2 inches, and they are located at 90° intervals circumferentially around the pipe (ref. Figure 5.3).

FIGURE 5.8. AE System Preamplifiers Mounted in Junction Box in Reactor Enclosure, Elevation 283' - Limerick Unit 1

FIGURE 5.9. AE Monitor Instrument Mounted in a Cabinet in Reactor Enclosure, Elevation 283' - Limerick Unit 1

FIGURE 5.10. Test of Signal Source Location by AE System Using Signal Input from 0.5-mm Pencil Lead Breaks - Limerick Unit 1

Pencil leads were broken within 1 inch to 2 inch of each sensor in turn. This plot shows that the circumferential location accuracy is very good. The axial accuracy is not as good but still within the acceptable limits of two wall thicknesses (wall thickness at this point is 1.3 inches). The very narrow axial spacing of the AE sensors contributes to the location scatter in the axial direction; i.e., a given error in signal time-of-arrival determination has greater significance with the narrow spacing.

The helium gas jet technique similar to that used during preparation in the shop to measure AE sensor sensitivity on the calibration block was again used to evaluate sensitivity of the sensors installed on the reactor. Figure 5.11 gives the results. The output was taken directly from the sensor which omits two filter stages in the midamplifier; hence, the response curve appears rather poor relative to the similar data taken earlier on the calibration block in the laboratory (Figure 5.2) where the 40 dB gain midamplifier was used. In reality, the response sensitivity of the installed sensors was significantly better as indicated by the voltage values from the plots (0.25, 0.71, 0.4, and 0.56 millivolts_{peak} for sensors #1, #2, #3, and #4 respectively). This could be due to better coupling of the installed sensors and the thinner pipe wall (1.3 inches) compared to the calibration block (4 inches). We have found that the helium gas jet tends to transmit to the sensors better with thinner material.

FIGURE 5.11. Sensitivity Test of Waveguide AE Sensors Installed on Reactor Pipe - Limerick Unit 1

5.3.4 Operational Monitoring

The AE monitor system setup included linear preamplifiers with a linear main amplifier in the AE monitor instrument. A total of 90 dB electronic gain was included (20 and 40 dB in the preamplifiers and 30 dB in the main amplifier). Under these conditions, the system performance looked very acceptable initially; i.e., the background noise was about 2 volts_{peak} and the installed sensor sensitivity checks were satisfactory. Subsequently, however, a spike transient noise signal appeared on all sensors which was in excess of 10 volts_{peak} out of the monitor instrument. This condition precludes effective AE detection. The detection threshold of the AE instrument cannot be raised high enough to ignore the noise signals because it would also be far too high to detect AE signals. The repetition rate of the noise signal was such that it was filling up a tape cartridge (2.1 million signals) record in a few days and in a few instances the data rate became high enough to cause the monitor instrument to saturate and stop processing data. This noise problem was unexpected because a similar AE system has been installed at Watts Bar Unit 1 reactor and operated during hot functional testing of the plant without any noise problem. In spite of the very diligent efforts on the part of PECO

personnel and the Battelle specialist, the specific source of the noise signal could not be identified. The magnitude of the noise transient reduced substantially during reactor startup and has remained low enough to allow effective AE monitoring. The AE monitor instrument has been modified to eliminate recording any signals that appear simultaneously on all sensors (noise transients appear on all sensors simultaneously while AE signals from cracking will show a difference in time of arrival at the various sensors depending on the location of the cracking). The fact remains, however, that the AE system is vulnerable to noise signal pickup. The Watts Bar and Limerick installations are compared in Table 5.1. The most prominent difference in signal lead shielding is in the containment penetration arrangement where the Limerick installation has only two wires while at Watts Bar there are three wires which are twisted to provide shielding. Another difference that could be significant is the signal cabling inside of containment, which is a high-temperature coaxial cable contained in metal conduit at Watts Bar compared to standard RG58 cable in a tray at Limerick. This emphasizes the need for extreme care in maintaining good shielding in such a high-gain electronic system.

Another step taken to guard against the return of the noise transient consists of including a differential amplifier between the midamplifier and the AE monitor instrument. A differential amplifier (unity gain) is intended to cancel out signals which appear simultaneously on different signal lines. This is needed because if the noise transient returns, it can still lock up the monitor system buffer memory if the repetition rate is sufficiently high. The differential amplifier reduced background noise by about a factor of two.

Coolant flow noise has not presented any problem to operational monitoring. There was no perceptible change in background noise as coolant flow was raised to full flow. This is a result of using high-frequency (400 kHz) tuned sensors.

PECO personnel are managing the normal operation of the AE system. The data analysis process requires PECO personnel to remove the data tape cartridges at least once per month and send them to PNL for analysis. Instructions have been provided on loading and removing the tape cartridges associated with the AE monitor instrument. PNL analyzes the data for source location of recorded signals, number of signals from a given location, and signal classification. This information is then compiled in a report submitted to PECO once per month.

Upon request, PNL will provide PECO with specifications from which they can establish an in-house data analysis capability. This should become attractive if future additional AE monitoring of plant components evolves.

The installed AE system is functional and capable of detecting and locating AE signals. It was installed and checked within a very stringent time window. Coolant flow noise has not caused a problem of increased background noise level. This was a point of concern at the outset because we have not previously monitored a BWR during operation. The remaining point of

TABLE 5.1. Comparison of Watts Bar and Limerick AE Installations

<u>Item</u>	<u>Watts Bar</u>	<u>Limerick</u>
Sensor	3' long two-piece waveguide	9' long one-piece waveguide
Sensor tuning	375 kHz and 500 kHz	425 kHz
Sensor to cable connect	BNC	BNC
Cable inside containment	RG141 in conduit	RG58 in cable tray
Cable-to-penetration inside connection	BNC	Cannon plug
Penetration	Three-wire twisted with 2' of wire at inside end and 6' of wire at outside end of plug	Two-wire straight with 3' of wire at each end of plug
Cable-to-penetration outside connection	BNC	Cannon plug
Cable outside containment	RG58	RG58
Preamplifiers	20 dB at sensor and 40 dB outside penetration	20 dB at sensor and 40 dB outside penetration

concern is the fact that the AE system is subject to picking up electrical/electronic noise signals due to a weak point in the shielding (felt to be primarily at the containment penetration). Day-to-day operation of the AE system and replacement of the data storage cartridges for transfer of data to Battelle is being very capably handled by PECO personnel.

The results from this work will be reported separately as the AE data and data interpretation are verified by independent methods.

6.0 TECHNOLOGY TRANSFER

An important element of the total program is technology transfer which is defined as making the developed technology available to NRC Regulatory and the utility users. Although this simple definition may sound like a rather trivial requirement, it is not. Making the technology available involves not only describing the technology in open literature in sufficient detail for vendors and users to duplicate and expand on methodology developed as a part of this program; it also involves creating and obtaining approval of standards and code material which will guide the application of the technology and provide an organized basis for Regulatory and the utilities to judge where application of AE monitoring is acceptable and beneficial.

Another aspect of technology transfer involves publications and participation in technical meetings and committees. This presents the technology to a public forum with the opportunity for interested people to discuss the details of the developments.

6.1 ASTM STANDARD

An American Society for Testing and Materials (ASTM) Standard E 1139 titled "Continuous Monitoring of Acoustic Emission from Metal Pressure Boundaries" was approved in 1987. It appears in the 1987 and later editions of the Annual Book of Standards. This standard provides guidelines for the application of continuous AE monitoring to detect cracking in metal pressure boundaries.

6.2 ASME SECTION XI CODE APPENDIX AND CODE CASE

Development of a Nonmandatory Appendix to the American Society of Mechanical Engineers (ASME) Section XI Code to address continuous, inservice AE monitoring of reactor components was started in January 1987. The Appendix titled "Acoustic Emission Monitoring of Nuclear Reactor Pressure Boundaries During Operation" was approved by the Subgroup on Nondestructive Examination in January 1988, but subsequently it was tabled by the Subcommittee on Nuclear Inservice Inspection at the same meeting. At the recommendation of the Subcommittee, a Code Case version of the material (Case N-471, Acoustic Emission for successive Inspections, Section XI, Div. 1) was prepared and was fully approved in May 1990.

The Code Case includes a Supplement that provides detail on special equipment and techniques developed under this program to facilitate on-line AE monitoring of reactor systems. This includes the waveguide AE sensor, frequency filtering, signal pattern recognition, and sensor mounting.

6.3 INFORMATION SEMINARS

Three seminars have been presented for NRC Regional and NRR personnel. The first held in October 1986 at Region I offices in King of Prussia was attended by Region I personnel only. The second seminar conducted in October 1988 at the Region I offices in King of Prussia was attended by Regions I, II, and IV personnel plus NRR representatives. A third seminar was held in February 1988 for Region III staff in Glen Ellyn, Illinois.

6.4 PROTOTYPIC INSTRUMENT SYSTEM

Development of an AE instrument system suited to the unique requirements of reactor component monitoring was an essential part of the program. The instrument system developed is described in this section and design drawings and software are in Volume 2 of this report. The primary special needs are long-term (1 to 2 year) operation with minimum attention, capability to store large quantities of data, and incorporating a method for identifying AE signals from crack growth as unique from other acoustic signals. The prototypic instrumentation system incorporates special techniques and methods developed during the program to that point. A flowchart of the acquisition, analysis, and display path is shown in Figure 6.1.

The prototypic instrument system consists of two parts: a data acquisition subsystem and a data analysis subsystem. Rationale for this approach is that the data analysis subsystem, being independent, could be used to service more than one data acquisition subsystem. The data acquisition subsystem performs a complete function of acquiring data and storing it on tape for record purposes. It can operate independently or in unison with the data analysis subsystem to give real-time data analysis. As improved and/or different analytical methods become available, the basic data recorded on tape is available for re-analysis.

6.4.1 Data Acquisition

Two basically identical data acquisition subsystems were fabricated. One is an 8-channel unit (Figure 6.2) which is portable for use in monitoring a specific flaw location. The other is a 24-channel unit (Figure 6.3) which is intended for semi-permanent installation to monitor several locations at a single reactor. The discussion which follows focuses on the 24-channel unit but is applicable also to the 8-channel unit.

The data acquisition subsystem is a 24-channel unit divided into six arrays that 1) acquires and stores raw AE data and associated parameters and 2) transmits all data to the DEC 11/73 computer analysis subsystem. The front-end unit is packaged in its own 6-foot-high environmental rack-mount enclosure. It is a microprocessor-based system that controls two identical 12-channel AE acquisition chassis. Two small integral CRTs are used for data display in quasi real time. All operator interaction occurs through a keyboard or by front-panel switches. Data storage is on a high-capacity digital tape cartridge.

FIGURE 6.1. Flowchart of Prototypic AE Monitor System

FIGURE 6.2. Eight-Channel AE Monitor Instrument Developed by Pacific Northwest Laboratory

FIGURE 6.3. Twenty-Four Channel AE Monitor/Analysis Prototype Developed by Pacific Northwest Laboratory

The two identical analog chassis are each comprised of six modules. Four of the modules are signal processors, one is peak time/pulse height (PT/PH), and the remaining one is RMS. Each signal processor handles four channels with three modules devoted to array processing and the fourth one to "guard" channels. The PT/PH module processes three channels, one for each of the three array signal processors. The RMS module uses all 12 channels coming to that chassis. An optional signal recognition module has been built which, when used, replaces and PT/PH module.

6.4.1.1 Signal Processors

The signal processor modules each accommodate four sensor channel inputs. Each module incorporates amplification and signal detection. Gain is set with on-board switches over a range from 15-30 dB in 1-dB steps. This provides for a system gain ranging from 75 to of 90 dB (microamp = 20 dB, midamp = 40 dB, signal processor = 15 to 30 dB). Signal detection is accomplished by comparing the AE signal to an operator-controlled DC voltage threshold. The resulting output is a digital signal that has a true value (logic state of 1 or 5 VDC) for the time period that the AE signal is above the threshold. A single front-panel threshold adjustment is used for all four channels of each module.

There is also a buffered analog AE signal output which is used for operator viewing and for input to the signal recognition module and/or the pulse height/peak time module.

In each module there is also a device that performs an RMS measurement on the analog AE signal. The resulting proportional DC voltage is outputted to the RMS module.

Each signal processor module provides DC power to its associated midamps. Each channel has a low voltage detector for monitoring a malfunction. If an alarm state occurs, a front-panel LED lights and a flag is sent to the system controller.

6.4.1.2 Array Processors

The array processors are digital cards that use the signal processor outputs to measure delta times. The delta times determine the difference in time of arrival of an AE signal at the various sensors in an array. The boards are jumper configurable so that an array can consist of one to four channels. Timing resolution is one microsecond. To keep from reprocessing the same signal, a hardware lockout feature is utilized. It tracks the signal duration and prevents reactivation of the timing circuits until the signal ends. The delta-time outputs sit on a microprocessor buss and will cause a processor interrupt for a reading only when all channels of the array have detected a signal. The delta times are required to perform point source location which is a task done by the DEC 11/73 analysis subsystem. The digital array processor cards reside on the same buss as the system controller.

6.4.1.3 Peak Time/Pulse Height

This module(s) sits in the analog chassis and performs measurements on three analog channels from three separate arrays. It consists of three identical circuits with a common digital output buss. The measurement principle is to digitize the front portion of the AE signal and to store the results in memory. An on-board single chip microcomputer is then used to analyze the memory to find the peak value (pulse height) and the time it took to obtain the peak (peak time). There are two modules providing for a measurement on each of the six total possible arrays.

6.4.1.4 RMS

The root-mean-square (RMS) module(s) resides in the analog chassis and monitors the RMS outputs from the signal processors. There are two identical modules, each accommodating 12 channels for a total of 24. Every AE sensor channel is continuously monitored for RMS changes. Each module has an on-board single chip microcomputer that compares each channel RMS value to a DC threshold. Number of occurrences above the threshold along with time above the threshold are reported to the system controller over a digital buss. The thresholds are front-panel adjustable with accompanying LED indicators for indications of alarm levels above the threshold.

6.4.1.5 Signal Recognition

The signal recognition module is a substitute for a PT/PH module. During setup, the operator will select which module is to be present. The recognition module output is two flags. One flag indicates a signal was processed, and the second whether the template was matched. The electronics use a single chip microcomputer for comparison of timing between the occurrence of secondary pulses. Refer to Section 6.4.3 for more detail on theory.

6.4.1.6 Parametrics

The system can measure eight non-AE parameters (coolant pressure, temperature, etc.) in terms of DC voltage over a 0-10 volt range. The number of parametric channels is selected by the operator during setup. Each time a valid AE event occurs (i.e., all sensors in a given array detect the AE signal), the parametric values will be inputted to the data and stored.

6.4.1.7 External Digital

A parallel input is provided for acquiring 16 bits of external digital data. This input is enabled by the operator during setup. If selected, the digital word would be inputted every time an AE event was recorded.

6.4.1.8 Data Storage

All data acquired by the front-end subsystem is stored on a 1/4-inch digital tape cartridge. Total storage capacity is 67 Mbytes, consisting of 16 tracks with each track having 4K blocks and 1K bytes for each block. This allows for over 1.8 million AE events to be recorded. Data format is custom and consists of a preamble code followed by logical ordering of specific data parameter types. Since the occurring AE data rate can exceed the tape recording rate, a data memory buffer is implemented. Data is stored in the buffer as it occurs, and it is off-loaded to the tape during non-event periods. Over 1000 AE events can be stored in the buffer.

6.4.1.9 Analysis Subsystem Interface

The data acquisition subsystem communicates with the DEC 11/73 computer analysis subsystem over a 16-bit parallel digital link. The parallel buss allows for a fast data transfer rate. The same data sent to the tape storage unit is also sent to the 11/73 interface. To ensure proper transfer, handshaking is employed. In the event the 11/73 is not present or malfunctions, the data acquisition system will still continue to acquire and record data.

6.4.1.10 Front-end Controller

The controller is a Z-80 8-bit microprocessor residing on an industry standard buss (STD buss). On this buss also resides the array processors, digital counters, I/O, and A/D cards. Software resides in a nonvolatile EPROM and is written in assembly language. The code has been optimized for real-time performance and makes extensive use of hardware interrupts. When an AE event occurs, the array processor issues an interrupt which causes the controller to branch into a data acquisition cycle. Data is acquired from both the analog chassis and the digital chassis. The controller writes the data into a memory buffer and then jumps into a transfer mode. The transfer mode extracts buffer data and writes it to 1) the cartridge tape, 2) DEC 11/73 analysis subsystem, 3) video itemized display subprocessor, and 4) video graphics display subprocessor. At periodic time intervals, the controller will automatically generate an acquisition cycle. This data is uniquely tagged and is used to ensure system integrity during quiet periods when there is no acoustic activity. It is also used to ensure regular recording of external parametric data values. Without this feature, during periods of little or no AE activity, there could be unacceptable gaps in the parametric data record.

To initiate a test, the operator communicates with the controller through a keyboard. Menus appear on a video monitor that allow the operator to exercise various functional aspects of the subsystem. Prior to test initiation, a setup menu must be completed. This is an "answer the prompts" approach and becomes the basis of a header associated with each AE event. Once a test is started, the keyboard is disabled and communication is with front-panel switches for controlling different video displays. There are two

video monitors, each controlled by slave processors which both reside in a common chassis separate from the controller chassis.

In addition to controlling data acquisition and data transfer/storage, the controller can also read and display cartridge tapes. This provides for a self-check feature when the data acquisition subsystem is used in a stand-alone mode.

6.4.1.11 Data Display

Two 5-inch black and white video monitors are used for operator display. One monitor is devoted entirely to a limited graphics display of AE source locations. A separate Z-80 microprocessor monitors stored data and performs source location calculations for zoned geometry areas. This is a limited version of what the analysis subsystem does, which is to calculate actual point location. The graphics display can be switched to any one of the six possible arrays. The display will also be different depending on the number of array sensors (quad, triad, linear, or single).

The other monitor is used for real-time data display and for operator interactions. During a test, it will display the acquired data parameters for each event regardless of which array. A separate Z-80 microprocessor is used for these calculations and display formatting. The display is also used prior to testing for operator control and setup. A menu scheme is used in which the operator interacts with the system via a keyboard and makes functional choices from those displayed on the monitor.

6.4.2 Data Analysis Subsystem

The analysis subsystem was developed to maximize speed and information accessibility with minimal operator intervention. The subsystem is provided with a small number of clearly marked switches and indicators to allow a plant technician easy access to information without requiring detailed knowledge of acoustic emission techniques and instrumentation. Information concerning acoustic emission activity is presented in the form of simple graphical displays on a CRT screen. The choice of parameters displayed, the reactor area of interest, and information screening is achieved through front panel switches so that with minimal training and without computer experience, the operator can characterize activity according to location, signal amplitude, rise time, and pattern recognition class. Front-panel lamps direct the operator to monitoring and serviceable instrument conditions such as a full data tape or empty hardcopy paper bin.

For unattended monitoring, several automated features were incorporated. Hardcopies of a select group of principal displays are periodically generated allowing acoustic signal evolution to be examined. A 67-Mbyte cartridge tape unit used for off-line analysis also serves as a backup on-line storage device when primary storage is filled. The system can also be remotely accessed via a modem.

The analysis subsystem is built around the DEC Q-bus using the KDJ11 central processing unit. For the sake of speed, the software makes extensive use of VM: device, which is extended RAM memory accessed as a disk. Several I/O cards provide communication to the external devices, which include the SYS 84 front end, the HCD 75 tape drive, various front-end switches and LED indicators, the graphics hardware, terminal, and the modem. A floppy disk interface module may be installed for software updating or diagnostics. Functions of the various modules incorporated in the system are described in Table 6.1 - Module Functions.

Signal analysis and display software is written as a series of modules which carry out the tasks of interacting with the data acquisition subsystem; reading, translating, and analyzing data; making and updating displays; and interacting with the operator. The programs are written both in DEC assembler and RT11 FORTRAN. The graphics display routines utilize the REGIS-oriented MGSP software package available from Multiware. A description of the functions performed by the software routines is included in Volume 2 of this report.

6.4.2.1 Software Program Descriptions

The main program is NEWMAN.FOR, a FORTRAN routine responsible for calling and coordinating the routines listed below.

6.4.2.1.1 Utilities. Utilities for initializing pointers, and opening and closing communication lines and files, are contained in the file UTIL.MAC. The various functions are performed by resident subroutines. The subroutines and their functions are:

- GETDAT: Sets interrupts and points for reading raw data from the System 84. Stores addresses of macro/FORTRAN shared variables. Initializes write pointers.
- PANSET: Sets front panel interrupts.
- ENABLE: Enables interrupts for the front panel and the System 84.
- PNTSET: Set interrupt pointers.
- CHANNEL: Open all channels used by the system. Open template files.
- CLOSIT: Disable System 84 interrupts. Close virtual disk files. Purge or save data file.
- CLOSUP: Close template files.

TABLE 6.1. Module Functions

<u>Module</u>	<u>Function</u>
DKJ11-AA	Central processing unit. Configured for boot address 173000.
MCV11-DC	CMOS memory module comprised of 32 kilobytes of battery-backed RAM. The purpose is to provide storage space for program and test parameters.
NS23R	National Semiconductor, 1 Megabyte RAM module. Two modules provide enough memory to operate the system from a virtual disk.
SCD-DLV11J	<p>Sigma Research, 4-port serial I/O board. Ports are allocated as follows:</p> <p>Port 1: terminal port for model at 9600 baud Port 2: graphics display at 19.2 baud Port 3: - (at 9600 baud) Port 4: printer at 9600 baud</p> <p>The base address is 176500, base vector 300.</p>
ADAC 1632	16-bit parallel I/O module to read data from the System 84 front end. Interrupt line allocations are RQST A = RESET and RQST B = DATA READY. Base address is 167770, base vector 300.
ADAC 1664	64-bit parallel I/O module used to communicate with the HCD-75 cartridge tape. The link utilizes a custom circuit residing on a separate board, described in a separate section. Base address is 164100.
ADAC 1616 CCI	A contact closure module used to detect closure of the ENTER and CLEAR front-end switches. Closure of either switch generates an interrupt, which causes all switch settings to be read via the DRV 11 module. Base address 164010, base vector 270.
ADAC 1616 HDO	Solid-state relay module used to control panel lights and annunciators. Base address is 164000.
DRV 11	16-bit parallel I/O module used to communicate with the front panel switches. Base address is 167760.

6.4.2.1.2 Data Read and Transfer, Plot Setup, Front Panel Interrupts.

- CCIREAD: Reads data from 1616CCI card and passes data to FORTRAN. Installs graphics template and filters. Resident subroutine MACBLK creates master common block shared by macro and FORTRAN.
- READ: Interrupt service routine which reads from the System 84, stores the data in a low memory ring buffer, and writes the data to file VM:RAWDAT.DAT. Contains the subroutines RESET to close the file and end the test, and CCIISR which responds to the CLEAR and ENTER front panel switches. Sets the system status variable and disables interrupts.

6.4.3 Signal Recognition Design

The purpose of the signal recognition module is to identify AE signals produced by crack growth. For the techniques being utilized which require use of a waveguide sensor, the criteria requires that the waveform produced by a crack-growth event have three characteristic pulses separated by specified time intervals (see Figure 4.9). There is, however, no restriction on either the maximum signal amplitude or the relative amplitudes of the three pulses; hence, the signal recognition module must be able to accommodate possible variations in these parameters.

Two approaches to module design were recognized. The approach used during development of the signal recognition technique was to compare digitized waveform envelopes with a template representing a typical three-pulse response. Effects of variations in maximum signal amplitude were removed by normalizing the signals. The result of the comparison was the maximum value of the cross-correlation function, and only those signals were classified as crack events whose cross-correlation value exceeded a set minimum. Requirements for field implementation of this approach would be a signal digitizer and a microprocessor with analysis software. An input buffer would also be desirable to prevent loss of data when the AE event rate is high, since the signal processing is relatively slow.

As an alternative, an inexpensive analog module was designed and built which allows AE source classification with greater speed and comparable accuracy. The concept is based on a "triggers and windows" principle illustrated in Figure 6.4. The time delays between the three important peaks in a crack-growth signal are determined by the longitudinal and shear speeds and the waveguide length. These delays are therefore constant for a given waveguide sensor, and it is only necessary to detect peaks with the proper delays in order to verify a signal produced by crack growth. In its simplest form, a detection threshold is set moderately high but below the peak amplitudes so that the signal will cross it only near a peak. The first threshold crossing triggers a timer; and after a set delay, the device begins looking for a second threshold crossing. If a second crossing occurs within a time window containing the expected pulse arrival time, and whose width is set to account for stochastic variation in the signal, a flag is triggered. If a third crossing with the proper delay corresponding to the time lag between the

FIGURE 6.4. Pattern Recognition Trigger-Window Sequence

first and third signal peaks is similarly detected, the signal is flagged as a crack event.

To accommodate various signal amplitudes, five thresholds were used, and the above criteria applied on each threshold level. The upper threshold value can be set by an operator to any value less than or equal to 10 volts which is the range of the peak height detector. The remaining thresholds are spaced at equal intervals below the upper value.

Two other potential problems were anticipated based on earlier visual examination of the waveforms, and appropriate modifications were made to the instrument. Waveforms were observed which were composed of many signal peaks similar to a beat-type effect, and other signals of persistent high amplitude. In both cases, the detection threshold of the module may be crossed within the window even though the signals do not show the desired pattern. To avoid this, the set delay following the first threshold crossing was replaced with a shorter delay plus an additional window, resulting in the following sequence: threshold crossing, delay, window 1, window 2. Window 1, which encloses a range of times just prior to arrival of the second peak of a crack signal, requires that there be no threshold holding; and window 2 encloses the expected arrival time of the second pulse in a crack-type event where a threshold crossing is expected. A continuous high-amplitude signal or a signal with many peaks, however, would trigger a second threshold crossing within window 1. In order for a signal to be flagged as a crack event, no

threshold crossing is allowed in window 1, but a crossing is required in window 2. The same approach was implemented for the third pulse. The second anticipated problem was the possible absence of the third pulse in the crack signal. The amplitude of the third pulse is always less than that of the first peak since the third peak is the internal reflection of the longitudinal pulse (first peak). In addition, the third peak amplitude may be further reduced depending on the reflection characteristics at the waveguide tip. Since the third peak may be present but not detected, an additional flag was used to indicate that a signal with two pulses in the proper positions was detected. The output of the signal recognition module is, therefore, a "3" if all three pulses arrived at the proper times and were detected and a "2" if a signal did not meet the criteria. These values are input as a parametric and are stored with the other data obtained from each detected event.

6.4.4 Cabling and Preamplifiers

Two separate signal amplification devices exist between the sensor and the AE instrument: 1) microamp and 2) midamp. The microamp is packaged integral with the sensor as part of the waveguide. In addition to gain, the microamp provides impedance matching between sensor and output coaxial cable. The midamp is located between the sensor/microamp and AE instrument. It adds additional gain and sharp signal filtering. Due to the long cable runs of several hundred feet, the midamp concept is used to preserve the signal-to-noise ratio.

6.4.4.1 Microamp

The microamp is a 20 dB (gain = 10) differential amplifier. It is a compact PC board device (1-1/4" x 1") allowing for integral packaging with the sensor. This close proximity has several advantages. First, by eliminating any cable on the sensor output, capacitance is minimized. This improves high frequency response and associated amplitude at the monitoring frequencies of interest (≈ 400 kHz). The sensor output impedance is not low and is significantly affected by cable capacitance which affects higher frequencies. Second, noise is minimized by immediately amplifying the sensor signal. This equates to the total system signal-to-noise ratio being set by the microamp. The electronic noise of the first amplification stage defines the minimum detectable signal.

The microamp gain stage is composed of only discrete active devices designed for low noise and high input impedance. It is a differential design that provides for a high common mode rejection (useful in electrically noisy applications). The differential pair devices were selected from a manufacturer with the pair on the same substrate for equal temperature tracking and electrical performance. The differential gain stage is fixed at 20 dB (gain = 10) with its output connected to a coaxial cable driving buffer. The buffer has unity gain and high current drive to handle the cable capacitance which accumulates with increasing distance.

DC power is supplied over the same coax that transmits the detected signal out. The DC voltage (+24 VDC) is inductively decoupled from the coax. The output would appear on an oscilloscope as a +24 VDC level with AE signals superimposed. The microamp is capable of operation at high temperatures which would occur inside reactor containment.

6.4.4.2 Midamp

The midamp is an intermediate device to provide additional gain and selective filtering. For reactor monitoring of a location within containment, the midamp would be located just outside containment. It consists of four small PC boards packaged inside a metal box (4" x 2-3/4" x 1-3/4").

The first stage is a second-order fixed high-pass filter. It severely attenuates frequencies below the desired AE monitoring frequency. Following the filter is a single-ended 20 dB (gain = 10) amplifier. The preceding filtering prevents the amplifier from being driven into saturation by low frequency sources (vibration, cavitation, etc.). The amplifier output drives a tuned circuit which can be jumpered to either 425 kHz, 500 kHz, or 600 kHz. The bandpass of this tuned circuit is not very sharp, having a 100 kHz passband ($Q = 4$). This tuning network is optional and can be bypassed. The last stage is another 20 dB amplification stage with a cable driver output.

DC power is supplied on the coax signal cable. The midamp inductively decouples the DC voltage at its output side and inductively adds it back at the input side for the microamp.

6.4.4.3 Cabling

Cabling requirements are for 50 ohm coax. Inside containment a high-temperature type such as RG-141 A/U, which has a teflon insulation, is required. For outside containment, RG-58 U or equivalent is acceptable.

6.4.5 Waveguide AE Sensor

The environment imposed on the AE sensor for continuous monitoring of reactor components during reactor operation is very demanding. The surface temperature of most components is about 600°F, the ambient air temperature inside containment is about 175°F, and gamma radiation levels in the order of 5.0×10^4 R/hour can be expected. In addition, drilling and tapping holes in the surface of a reactor vessel or piping is not an acceptable method of mounting AE sensors or sensor mounting fixtures.

An AE sensor consisting of a metal waveguide with an integral piezoelectric crystal and preamplifier mounted on one end of the waveguide and with the other end of the waveguide pressure coupled to the surface of the component being monitored has been demonstrated to be suitable for use in the operating reactor environment. The sensor design is shown in Figure 6.5. The length of the waveguide can be adjusted to suit a given situation with some qualification. Signal attenuation in the 1/8-in. diameter waveguide has been

FIGURE 6.5. Design of Waveguide AE Sensor Developed at Pacific Northwest Laboratory

measured to be 0.45 dB/ft. in the 400 to 500 kHz frequency range. The longest waveguide applied to date was 17 ft. 8 in., which represents a linear factor of 2.5 reduction in acoustic signals over the length of the waveguide. Sensitivity was still adequate to detect AE signals. Waveguide AE sensors currently installed on a nozzle at Limerick Unit 1 reactor are 9 ft. long and calibration following installation showed the sensitivity was quite adequate for the detection of AE signals.

The piezoelectric crystal and preamplifier (this is the microamplifier discussed in Section 6.4.5.1) assembly at the outer end of the waveguide is shown in Figure 6.6. MIL Standard components are used in the microamplifier to maximize resistance to temperature damage. It should be able to withstand long-term service at temperatures up to 200°F. Inductive tuning is incorporated into the microamplifier to facilitate discriminating against acoustic signal frequencies below a selected value; in most reactor component monitoring, this value would be about 450 kHz. The tuning is an important feature to overcome the problem of interference from background noise such as coolant flow noise. This design of waveguide AE sensor has been used in

FIGURE 6.6. Waveguide AE Sensor Head

monitoring vitrified waste tests where they were exposed to about 5.0×10^4 R/hour of 1 to 1.3 MeV gamma radiation for 120 days without any evident deterioration.

In the course of the research program, it was discovered that the waveguide AE sensor also offered another valuable feature. The unique response of the waveguide sensor to an AE signal from crack growth provides a means of identifying these signals and is discussed in Section 6.4.3.

7.0 SUMMARY AND CONCLUSIONS

The program has taken the AE technology from the status in 1976 where the consensus was that AE monitoring could never be utilized for on-line surveillance of reactors systems to the present status where on-line monitoring is a reality by having solved major application problems and demonstrated the validity of these solutions. Problem solutions that were developed include:

- an effective high-temperature AE sensor
- a method of coupling AE sensors to the reactor component which is efficient to transmit AE signals and also acceptable to nuclear reactor practice
- a method to overcome interferences from the background noise (primarily coolant flow noise) and still retain sufficient sensitivity to detect AE signals
- an instrument system suitable for long-term monitoring and capable of providing mass data storage
- a method of identifying AE signals from crack growth in the midst of other acoustic noise signals
- a relationship for estimating crack growth rate based on AE rate.

The program has attained international stature through participation in the Phase 2 vessel test with the German Materialpruefungsanstalt (MPA) Laboratory and presentations at numerous international technical meetings.

The program has gone beyond development and demonstration of the technology to promote effective and beneficial application of the technology. This has involved preparation and approval of an ASTM Standard E1139 and an ASME Section XI Code Case #471 which guide and support the use of continuous AE monitoring. Finally, the methodology and technical details are fully disclosed to benefit users.

The accomplishments under the AE/Flaw Characterization Program benefit the NRC and the nuclear industry in several ways. It provides an alternate method for surveillance of identified flaws to detect growth. This enhances safety by continuous surveillance and increases regulatory flexibility in dealing with such cases. The need for manual inspection of service-sensitive piping and nozzles is reduced which supports ALARA by reducing personnel exposure. Continuous monitoring for early detection of cracking in the primary pressure boundaries results in increased operational safety by timely detection and location of active flaw sites. The inherent leak detection capability is more sensitive and timely than current leak detection methods, thus, increasing operational safety.

8.0 REFERENCES

Bryan, R. H., P. P. Holz, J. G. Merkle, G. C. Smith, J. E. Smith, and W. J. Stelzman. 1978. Test of 6-in.-Thick Pressure Vessels, Series 3: Intermediate Test Vessel V-7B. NUREG/CR-0309, prepared by Oak Ridge National Laboratory for U.S. Nuclear Regulatory Commission, Washington, D.C.

Bryan, R. H., D. A. Canonico, P. P. Holz, S. K. Iskander, J. G. Merkle, J. E. Smith, and W. J. Stelzman. 1979. Test of 6-in.-Thick Pressure Vessels, Series 3: Intermediate Test Vessel V-8. NUREG/CR-0675, prepared by Oak Ridge National Laboratory for U.S. Nuclear Regulatory Commission, Washington, D.C.

Childress, C. E. 1969. Fabrication History of the First Two 12 In. Thick ASTM A-533 Grade B, Class 1 Steel Plates of the Heavy Section Steel Technology Program, Documentary Report 1. ORNL-4313, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Childress, C. E. 1971. Fabrication Procedures and Acceptance Data for ASTM A-533 Welds and a 10 In. Thick ASTM A-543 Plate of Heavy Section Steel Technology Program, Documentary Report 3. ORNL-4313-3, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Doctor, P. G., T. P. Harrington, and P. H. Hutton. 1979. Pattern Recognition Methods for Acoustic Emission Analysis. NUREG/CR-0910, prepared by Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, Washington, D.C.

Eason, E. D., E. E. Nelson, and J. D. Gilman. 1990. "Reactor Pressure Vessel Steel Fatigue Crack Growth Rates in Air and Water Environments," IAEA Specialists' Meeting on Subcritical Crack Growth, Moscow, May 14-18, 1990.

Edler, S. K., ed. 1982. Reactor Safety Research Programs Quarterly Report, April-June 1982. NUREG/CR-2127, Vol. 2, prepared by Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, Washington, D.C.

Exxon Nuclear Company, Inc. 1973. NDT-Acoustics Preservice of Calvert Cliffs Nuclear Power Plant Unit 1. XN-149, Vol. 1, Richland, Washington.

Exxon Nuclear Company, Inc. 1976. Final Report, NDT-Acoustics Inservice Inspection of Pilgrim Nuclear Power Station Unit 1 Reactor Vessel Nozzles N4A and N2B. XN-387, Richland, Washington.

Gibbons, W. S. and B. D. Hackney. 1964. Survey of Piping Failures for the Reactor Primary Coolant Pipe Rupture Study. GEAP-4574, General Electric Company, San Jose, California.

Hartman, W. F. 1986. "Is It Time for Acoustic Emission Surveillance of Operating Nuclear Reactors?" Journal of Acoustic Emission, 1(5).

Hiser, A. L., F. J. Loss, and B. H. Menke. 1984. J-R Curve Characterization of Irradiated Low Upper Shelf Welds. NUREG/CR-3506, prepared by Materials Engineering Associates, Inc. for U.S. Nuclear Regulatory Commission, Washington, D.C.

Hutton, P. H. and E. B. Schwenk. 1977a. Program to Develop Acoustic Emission - Flaw Relationships for Inservice Monitoring of Nuclear Pressure Vessels, Progress Report for February 1, 1977 to July 1, 1977. NUREG/CR-0250-2, prepared by Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, Washington, D.C.

Hutton, P. H. and E. B. Schwenk. 1977b. Program to Develop Acoustic Emission - Flaw Relationships for Inservice Monitoring of Nuclear Pressure Vessels, Progress Report for July 1, 1976 to February 1, 1977. NUREG/CR-0250-1, prepared by Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, Washington, D.C.

Hutton, P. H., E. B. Schwenk, R. J. Kurtz, and C. Pavloff. 1978. Program to Develop Acoustic Emission - Flaw Relationships for Inservice Monitoring of Nuclear Pressure Vessels, Annual Report for July 1, 1976 to October 1, 1977. NUREG/CR-0123, prepared by Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, Washington, D.C.

Hutton, P. H., T. T. Taylor, J. F. Dawson, R. A. Pappas, and R. J. Kurtz. 1982. Acoustic Emission Monitoring of ASME Section III Hydrostatic Test. NUREG/CR-2880, PNL-4307, prepared by Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, Washington, D.C.

Hutton, P. H., J. F. Dawson, M. A. Friesel, J. C. Harris, and R. A. Pappas. 1984. Acoustic Emission Monitoring of Hot Functional Testing, Watts Bar Unit 1 Nuclear Reactor. NUREG/CR-3693, PNL-5022, prepared by Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, Washington, D.C.

Hutton, P. H., R. J. Kurtz, R. A. Pappas, J. F. Dawson, L. S. Dake, and J. R. Skorpik. 1985. Acoustic Emission Results Obtained from Testing the ZB-1 Intermediate Scale Pressure Vessel - An Interim Report. NUREG/CR-3915, PNL-5184, prepared by Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, Washington, D.C.

Hutton, P. H., M. A. Friesel, J. F. Dawson, and J. C. Harris. 1988. Acoustic Emission System Calibration at Watts Unit 1 Nuclear Reactor. NUREG/CR-5144, PNL-6549, prepared by Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, Washington, D.C.

Kurtz, R. J., P. H. Hutton, and T. E. Hall. 1987. Testing of Flawed Pipe Repairs. NP-5237-LD, Research Project T303-2, prepared by Battelle, Pacific Northwest Laboratories for the Electric Power Research Institute and the BWR Owners Group.

McElroy, J. W. 1982. Topical Report, Acoustic Emission Crack Detection Methods for Light Water Reactor Application. DOE Report C00-35208-4, PECO/ADAB Report ADAB-2-44, pp. 19-22, Philadelphia Electric Co., Philadelphia, Pennsylvania.

Office of Nuclear Research, U.S. Nuclear Regulatory Commission. 1988. Fifteenth Water Reactor Safety Information Meeting, Volume 2. NUREG/CP-0091, Vol. 2, pp. 338-341, Washington, D.C.

Parry, D. L. and D. L. Robinson. 1970. Incipient Failure Detection by Acoustic Emission - A Development and Status Report. IN-1398, pp. 38-42, Idaho Nuclear Corporation, Idaho Falls, Idaho.

Phillips, C. A. G. and R. G. Worwick. 1968. A Survey of Defects in Pressure Vessels Built to High Standards of Construction and Its Relevance to Nuclear Primary Circuit Envelopes. United Kingdom Atomic Energy Authority, AASB(S)R 162.

DISTRIBUTION

<u>No. of Copies</u>		<u>No. of Copies</u>	
	U.S. Nuclear Regulatory Commission Division of Technical Information and Document Control 7920 Norfolk Avenue Bethesda, MD 20014		B. E. Rodgers TVA Nuclear Maintenance BR5N49A Chattanooga, TN 37402-2801
	J. Muscara NRC/NRR Mail Stop NS 217C		R. C. Emory Maintenance Engineering and Technical Section MOB1L Watts Bar Nuclear Plant PO Box 2000 Spring City, TN 37381
	C. Z. Serpan NRC/NRR Mail Stop NS 217C	21	<u>Pacific Northwest Laboratory</u>
3	D. B. Fethers, 62A4-1 Nuclear Engineering Dept. Philadelphia Electric Co. 965 Chesterbrook Blvd. Wayne, PA 19087-5961		DM Boyd JF Dawson SR Doctor MA Friesel PH Hutton (3) RJ Kurtz (3) DK Lemon
2	J. McElroy Philadelphia Electric Co. PO Box 650 Valley Forge, PA 19482		RL Moffitt JR Skorpik JC Spanner TT Taylor
5	D. D. Nunn Vice President, Nuclear Projects TVA LP6N20E Chattanooga, TN 37402-2801		Technical Information (5) Publication Coordination
	J. A. Scalice Plant Manager Watts Bar Nuclear Plant PO Box 2000 Spring City, TN 37381		

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
(Assigned by NRC. Add Vol., Supp., Rev.,
and Addendum Numbers, if any.)

NUREG/CR-5645
PNL-7479

2. TITLE AND SUBTITLE

Acoustic Emission/Flaw Relationships for
Inservice Monitoring of LWRs

3. DATE REPORT PUBLISHED

MONTH YEAR

October 1991

4. FIN OR GRANT NUMBER

FINs B2088, B2913

5. AUTHOR(S)

P. H. Hutton, R. J. Kurtz, M. A. Friesel, J. R. Skorpik,
J. F. Dawson

6. TYPE OF REPORT

Technical

7. PERIOD COVERED (Inclusive Dates)

July 1976-Oct. 1988

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Pacific Northwest Laboratory
Richland, WA 99352

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Division of Engineering
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

The program concerning Acoustic Emission/Flaw Relationships for Inservice Monitoring of LWRs was initiated in FY76 with the objective of validating the application of acoustic emission (AE) to monitor nuclear reactor pressure-containing components during operation to detect cracking. The program has been supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research with supplemental support from the Tennessee Valley Authority. Research and development has been performed by Pacific Northwest Laboratory, operated for the Department of Energy by Battelle Memorial Institute. The program has shown the feasibility of continuous, on-line AE monitoring to detect crack growth and produced validated methods for applying the technology. Included are relationships for estimating flaw severity from AE data and field applications at Watts Bar Unit 1 Reactor, Limerick Unit 1 Reactor, and the High flux Isotope Reactor. This report discusses the program scope and organization, the three program phases and the results obtained, standard and code activities, and instrumentation and software developed under this program.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

nondestructive testing, acoustic emission, inservice monitoring,
flaw detection, reactor pressure boundaries

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)

Unclassified

(This Report)

Unclassified

15. NUMBER OF PAGES

16. PRICE

**UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555**

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, \$300

SPECIAL FOURTH-CLASS RATE
POSTAGE & FEES PAID
USNRC

PERMIT No. G-67

ISBN 0-16-035990-2

9 780160 359903