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FINAL REPORT

**An Experimental Study of
Advanced Continuous Acoustic Emission Monitoring
of BWR Components**

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September 1980

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Technology for Energy Corporation

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An Experimental Study of
Advanced Continuous Acoustic Emission Monitoring
of BWR Components

John W. McElroy

Philadelphia Electric Company
September 1980

William F. Hartman
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SECTION 1

Summary

A program to evaluate the capabilities of continuous, on-line acoustic emission (AE) monitoring as applied to nuclear reactor pressure boundary components has been ongoing at the Philadelphia Electric Company's Peach Bottom Nuclear Generating Station Unit 3 since April of 1977. This final report covers the contract period from October, 1977 through February 28, 1979.

The program consisted of installing, maintaining, and monitoring AE sensors located on primary piping, nozzles, and valves in the BWR system. The analysis of the AE data obtained on the structural and operational integrity was correlated to the results of supplementary nondestructive testing techniques used during the in-service inspection, performed at refueling outages.

The purpose of the program was to develop the on-line surveillance acoustic emission technique in order to identify areas of possible structural degradation, thereby resulting in reduced off-line inspection time because of selected concentrated inspection activity. The result of reducing inspection time was to reduce accumulated radiation exposure to inspecting personnel and to reduce the amount of critical plant outage time by identifying the critical inspection areas during operation.

The program demonstrated the capability of acoustic emission instrumentation to endure the nuclear reactor environment. The acoustic emission sensors withstood 12 months of reactor operation at temperatures of 400°F and greater in high radiation fields. The preamplifiers, also mounted in the reactor environment, operated for the 12-month period in 100% humidity, 250°F conditions. The remaining cable and AE instrumentation were operated in controlled environments.

This program demonstrated that on-line AE monitoring in an operating nuclear power station was feasible, especially for selected regions of certain components. The AE monitoring data obtained during operation was used to identify locations where further detailed ISI was performed. The subsequent ISI data obtained corroborated locations found to be free of AE indications and locations found to contain insignificant AE indications. No significant AE activity occurred at any locations. This fact remained as the major deficit in the program. The correlation between no AE activity and no ISI indications and insignificant AE activity and insignificant ISI indications at several locations with no data indicative of significant AE activity and no data indicative of significant ISI indications was developed as a basis for continuing the program past the present contractual period.

In addition to the acoustic emission investigation of the nozzles and piping, an investigation was performed to develop techniques of on-line surveillance of the main steam safety relief valves. A surveillance system utilizing a microprocessor was designed to remotely monitor the acoustic levels within the valves to detect minute leakage. The monitoring system would alarm if the acoustic levels within certain bandwidths exceeded a preset level. The developmental program associated with the valve leakage included various statistical data manipulation procedures. The data analysis resulted in utilizing a plot of the ratio between adjacent bandwidth acoustic levels; one bandwidth was the one containing pertinent frequencies of leakage, the other bandwidth was the one containing frequencies characteristic of normal operating noise. By putting the two adjacent bandwidth acoustic levels into a ratio form, the system was capable of distinguishing between valve leakage and normal environmental noise increases.

SECTION 2

Introduction

The research effort is a continuation and expansion of previous programs sponsored by the U. S. Department of Energy and the Philadelphia Electric Company. The entire effort began in 1975 when feasibility studies were being performed at Yankee Atomic Power Plant in Vermont and at Peach Bottom Atomic Power Station in Pennsylvania. The success of these studies led to the research program conducted at Peach Bottom from October 1976 to October 1977 under DOE contract EN-77-C-02-4087. That program was continued under the DOE contract EN-78-C-02-4592 covering the period October 1977 through February 1979. This final report covers all work performed under the later contract.

Improvements in nuclear electrical energy production can produce significant benefits for the utility industry, the nuclear industry and the general public. A forced outage of a large Light-Water-Reactor nuclear unit can cost up to \$500,000 per day in replacement power costs alone. Therefore, significant incentive exists for improving plant productivity by improving availability of the operating power stations. On-line surveillance by acoustic emission (AE) monitoring can contribute to improving availability by providing early detection of degenerative conditions, especially those which may lead to eventual plant shutdown. The risk of major damage is thereby reduced, needed maintenance may be better planned, and data collected could assist in identifying the root causes of a problem and thereby prevent repetition. An additional benefit, reduced occupational radiation exposure, may occur by reducing current manual inspections because on-line AE surveillance should identify which locations require inspection.

Acoustic emission signals can emanate from a variety of different types of sources. For pressurized systems, two basic types of AE signals can be detected, namely, discrete-event signals and steady-leak signals. The former type is often called "burst emission" and is normally associated with crack

growth and plastic deformation. Normally, this type of AE corresponds to the current ASTM definition of AE: "a transient elastic wave, generated by the rapid release of energy from a localized source within a material." However, discrete-event AE can also be caused by impacting and frictional effects.

Reliable detection and confident interpretation of AE signals which occur under operating conditions may be hampered by background noise or electrical interference. While operating parameters, such as pressure, temperature, flow rates and vibration levels may remain nominally constant, AE can be produced by a growing discontinuity or area of degradation. The rate of such structural degradation can be slow and intermittent, and can depend strongly on operating conditions. The need for studying these interrelationships in the actual environs of an operating plant was considered in the formulation of the AE surveillance techniques developed in this research program.

On-line AE monitoring of selected BWR valves, piping and nozzles is being performed at Philadelphia Electric Company's Peach Bottom No. 3 Atomic Power Station. The objective is to assess the feasibility of routine implementation of AE surveillance methods for increasing plant availability. The results of the measurement tasks of the surveillance methods are being compared with the results of in-service ultrasonic and radiographic examinations occurring during refueling outages.

SECTION 3

Objective

The chief objective of this program is to assess the feasibility of routine implementation of AE surveillance of crack propagation and leak detection as a technique to assist in reducing the radiation exposure to plant workers and for increasing the availability of Light-Water Reactor systems. The successful development and utilization of AE surveillance could result in:

- The reduction and minimization of manual in-service inspection (ISI) requirements.
- The alteration or minimization of operating practices that have a deteriorating effect on plant systems and components.
- The scheduling of maintenance on deteriorated components during planned maintenance outages rather than during unanticipated forced outages.
- The reduction in manpower radiation exposure and outages time by localizing areas of components requiring repair or inspection prior to containment entry.

The proposed objective is being accomplished through the successful completion of the following six tasks:

1. Continuation of On-Line Monitoring Effort to Refueling Outage
2. Refueling Outage Inspections and Installation
3. Main Steam Safety Relief Valve Exploratory Testing
4. Expanded On-Line Monitoring Effort
5. Data Analysis and Documentation
6. Refueling Outage Correlations

A detailed discussion into the work to be accomplished under each task is contained in (1). The semi-annual report, (1), also contains results from the work completed in Tasks 1 and 2. This final report covers the work accomplished in Tasks 3 through 6.

SECTION 4

Results

4.1 Tasks 1 and 2

This work was completed and the results are included in the semi-annual report (1).

4.2 Task 3: Main Steam Safety Relief Valve Exploratory Testing

Task 3 required certain tests to be performed on the main steam safety relief valves. The objective of these tests was to optimize sensor sensitivity to the dominate acoustic frequencies present during relief valve pilot stem leakage. These tests were to determine the relationship between the acoustic amplitude at the optimum frequency and the relief valve pilot stem leak rate.

A commonly used safety relief valve in BWR plants is a pilot-operated type, having a two-stage pilot valve section and a main valve section. A schematic of the valve is shown in Figure 1. In operation, the pilot stage is held closed by the internal pressure acting over the pilot stage seat area. As the pressure increases, bellows expansion reduces the abutment gap between the stem and disc yoke. When the stem abuts against the yoke, further pressure increase lifts the disc from its seat, admitting steam to the operating piston of the second stage, causing it to open. This vents the chamber over the main valve piston to the downstream side of the valve, thus creating a pressure differential across the main valve piston and the valve opens.

The pilot-disc seat is vulnerable to deteriorating processes which lead to increasing leakage across the seat into the chamber of the second stage. A small leak has no immediate major bad effect, because the pilot valve discharge line can accommodate the flow without substantial pressure buildup. However, as a small leak grows, the second-stage pressure increases until, upon reaching a critical value, the second-stage disc opens; even though the pilot disc was not lifted.

4.2 Task 3: Main Steam Safety Relief Valve Exploratory Testing (Cont'd)

Since valve malfunction does not occur until a leak worsens to that critical leak rate, early detection of leakage can allow corrective measures to be scheduled, thereby reducing the loss of plant availability.

Several types of surveillance techniques have been considered for detection of the valve leakage. Sensing the temperature near the discharge line is one possibility. This approach is generally considered to be unreliable and to have poor sensitivity. Direct measurement of the pressure in the second stage chamber could be accomplished by fitting the valve with a pressure transducer.

Although this invasive technique can be accommodated in a laboratory experiment, its implementation for plant use would require substantial additional safety analysis and testing of the valve's integrity. The monitoring of the leak noise transmitted to the external wall of the pilot stage is an attractive non-invasive method. A review of on-line leak detection methods for reactor systems indicated that AE monitoring has the greatest potential for providing desirable features not found in other methods. Among these are high sensitivity and potential for determining leakage rate.

A valve having a leaking pilot-disc seat was tested under laboratory conditions in order to determine the correlation of the leak noise with leak rate or second-stage pressure. The leak rate, downstream temperature and the RMS voltage of the acoustic signal are plotted against the second-stage pressure in Figure 2. Very soon after exceeding a second-stage pressure of 200 psi, the valve "pops." At that time, the second-stage pressure is about 400 psi, but, for all practical purposes, the valve of 200 psi may be taken as the critical pressure, since the pressure increases rapidly from 200 to 400 psi. The AE sensor was clamped onto the external housing of the pilot works. The signal was band-pass filtered in the range 5-8 kHz. The downstream temperature was measured by a thermocouple in the vicinity of the "pilot valve discharge line." As the second stage pressure increased from 40 to 200 psi, the leak rate increased 59%, the temperature increased 9% and the AE RMS voltage increased 370%. Therefore, the sensitivity

4.2 Task 3: Main Steam Safety Relief Valve Exploratory Testing (Cont'd)

of the acoustic detection is excellent. Furthermore, if the leak were intermittent, the acoustic detector would respond accordingly while the thermocouple would not.

Eleven safety relief valves at Peach Bottom Unit 3 were instrumented for AE leak surveillance. The signal from each valve's acoustic sensor was conditioned prior to being channeled to a multiplexer. The RMS values of the signals are sampled at a rate of one per second per channel. Each sample consists of two values, namely, the RMS values in two different bandwidths. In this way, substantial changes in background noise can be distinguished from leak noise. A microprocessor memory stores the last sixty pairs of RMS values for each channel, as well as the last one hundred hourly averages of each, together with the hourly variance. Also, every hourly average value is preserved on a floppy disk data storage device.

Figures 3a and 3b are examples of the type of data display that is presented by the surveillance system. The top plot of Fig. 3a shows the history of the minimum RMS value of the signal in the bandwidth 4-7 kHz. This is formed by plotting the smallest of 3600 RMS values measured each hour. This measurement is very sensitive to sustained leak noise or an overall increase in the continuous background noise; but it is not affected by impact noise, such as insulation chattering on the pipes. The bottom plot of Fig. 3a gives the hourly history of the average RMS values (4 days x 24 hours = 96 datapoints). It can be seen that this particular valve had an average signal of about 4.4g for three days, with the corresponding minimum value being about 2.9g. These values are between 5 and 10 times greater than the corresponding values for the other valves, during the same time period. Figure 3b shows that the ratio of the RMS value in the bandwidth 4-7 kHz to that in the bandwidth 1-4 kHz is 1.3, a ratio value which is about twice the ratio values for the other valves. It was concluded that Valve #11 was probably leaking, although not to a great extent.

4.2 Task 3: Main Steam Safety Relief Valve Exploratory Testing (Cont'd)

Subsequent records show that this valve's acoustic signal has not increased much, but from time to time it still exhibits behavior similar to that shown in Figs. 3a and 3b.

Since the refueling outage, completed in May 1978, three valves have shown acoustic indications of leaking, but none has given signs of increasing leak rate. Correspondingly, there have been no plant operational problems caused by leaking safety relief valves.

4.3 Task 4: Expanded On-Line Monitoring Effort

Several Boiling-Water Reactor (BWR) plants have reported incidences of crack growth on certain components. Typical of these is the intergranular stress-corrosion cracking in 304 stainless steel recirculation-bypass piping and core-spray piping. The majority of these cracks have been found on the inside surface of the pipe, in the heat-affected zone adjacent to the weld. These cracks are largely due to sensitization of the piping, local stress levels and local coolant oxygen levels. Cracks have also been found in the BWR feedwater-nozzle cladding, blend-radii area. These cracks are believed to be initiated by high cycle thermal fatigue due to mixing of hot and cold flow streams, and propagate due to stresses induced during startups and shutdowns. The crack area is adjacent to the feedwater spargers, which have also experienced vibrational fatigue in several plants. For further background details refer to (1).

Nuclear power plant owners recognize that on-line monitoring may enhance the reliability of standard in-service nondestructive evaluation by detecting growing insignificant defects. In order to minimize loss of plant availability, on-line detection and evaluation of growing defects is of economic interest. If, prior

4.3 Task 4: Expanded On-Line Monitoring Effort (Cont'd)

to a scheduled shutdown, surveillance data provide estimates of the probability of cracks in these pipes, then preparation can be made for additional inspections and/or repairs. Furthermore, surveillance data could also save time by localizing areas of concern thereby concentrating inspection activity.

During refueling outages of the Peach Bottom BWR Unit 3, AE sensors, calibration devices, vibration transducers and thermocouples are removed, refurbished, recalibrated, and reinstalled on:

- 1) Segment of the "B-loop" recirculation bypass line (RB), containing two welds,
- 2) Segment of the "B-loop" core-spray line (CSB), containing four welds,
- 3) Feedwater "A" nozzle (FWA), monitoring nozzle-to-vessel weld,
- 4) Feedwater "E" nozzle (FWE), monitoring nozzle-to-vessel weld; and, during a subsequent refueling outage,
- 5) Segment of the "A-loop" core-spray line (CSA), containing four welds.

All sensing devices, preamplifiers and cables had to be compatible with the environmental conditions within the reactor containment. The AE sensors were designed for continuous operation at 315°C. The sensors were calibrated using reciprocity techniques to have peak sensitivity at 315°C, at 500 kHz of 0.4 mV/Pascal (-85d ref: 1V/microbar). The circular housing is threaded, such that a thin stainless steel cap can be placed over the ceramic face plate. A high-temperature couplant material, Dow-Corning Heat Sink Compound No. 340, was used between the face plate and the threaded cap. This is particularly true for the AE sensors and vibration transducers, both of which are in direct contact with the reactor components. Reference should be made to (1) for more detailed description of the monitoring system.

4.3 Task 4: Expanded On-Line Monitoring Effort (Cont'd)

Each AE sensor is connected to a differential preamplifier. Having a fixed differential gain of 40 dB, the preamplifier has a flat frequency response over the bandwidth, 30-3000 kHz. (The monitoring bandwidth is bandpass-limited to 400-800 kHz in the main amplifiers.) The signal output from each preamplifier is connected to a high-temperature coaxial cable, which runs to a penetration connector. The same type of coaxial cable is used for each vibration transducer. These transducers are conventional charge-mode quartz accelerometers having a sensitivity of 1.0 picocoulomb per g. Each transducer is designed to function at operating temperatures up to 600°F and is connected by a hard-line ceramic insulated cable to a nearby charge converter.

The installed AE sensors were periodically checked for degradation or change in the coupling efficiency by using impact calibrators and measurement of background noise. The calibrators are remotely activated devices, each consisting of a high-temperature solenoid that drives a pin with a hardened tip onto the surface of the part. The resulting impact signal appears different on each type of pipe and at each sensor on that pipe; but it is assumed to always produce approximately the same stress wave at a given sensor. Figure 4 shows the performance of the sensors throughout the program from the impact calibrator tests. The sensors show continued efficiency of impact detection throughout the program. The background noise measurements on the majority of sensors varied from .025 to .032 Pascals at 500 kHz. This additional data confirmed sensor efficiency.

The locations of some of the sensing devices are shown in Figures 5 and 6. The AE sensors are mounted "dry" without the use of couplant material. On each component, guard sensors are mounted at the extremities of the region of interest. Whenever a guard sensor is

4.3 Task 4: Expanded On-Line Monitoring Effort (Cont'd)

activated prior to an array sensor, the AE system is inhibited from processing a corresponding data set. That is, only information resulting from AE waves originating within the region of interest will be accepted.

Except for six guard sensors, all AE sensors and vibration transducers are attached to the various components using stainless steel straps and special mounting brackets. The guard sensors on the reactor vessel wall are mounted using a hold down lever that attaches with straps to the nozzle. In addition to the AE sensors, transducers and simulators, each component has a thermocouple attached in the vicinity of the sensor array.

The AE instrumentation is installed in the control room area. The system contains 24 channels of amplification, with corresponding detectors and processors. The sensors are arranged into five one-dimensional arrays. For each detected AE event, the measured parameters are: the event number, arrival time difference, sensor identity codes in the sequence of arrival, the peak voltage of the signal's envelope at the first-arrival sensor, the real-time, the array number, the temperature of that array, the maximum vibration amplitude at that array in the previous one second interval, and a voltage corresponding to the core flow. All data are permanently recorded in digital form on data disks.

The array sensors are used in adjacent one-dimensional patterns. This scheme results in a simplified data-display format: namely a segmented line showing the locations of the 14 sensors. The coordinates assigned to each sensor are listed in Table 1. These coordinates are fictitious but the spacing between sensors in each array is actual.

TABLE 1

ARRAY SENSORS

<u>MSCD Channel No.</u>	<u>X (cm) Coord.</u>	<u>Array No.</u>	<u>Sensor Ident.</u>
1	0	1	RB1
2	27.3	1	RB2
3	52.7	1	RB3
4	70.0	2	CSB1
5	156.5	2	CSB2
6	192.1	2	CSB3
7	210.0	3	FA1
8	275.4	3	FA2
9	347.8	3	FA3
10	411.9	3	FA4
11	431.9	4	FE1
12	497.3	4	FE2
13	569.7	4	FE3
14	633.8	4	FE4
15	650.0	5	CSA1
16	736.5	5	CSA2
17	772.1	5	CSA3

4.3 Task 4: Expanded On-Line Monitoring Effort (Cont'd)

The computerized monitoring and data processing system has been operating at the plant since March 23, 1977. Several times weekly, real-time displays of the AE data are hard copied and are logged for reference. The real-time data displays depict the AE activity over the entire monitored portion of each component. Using a Data Analysis Mode of the system, it is possible to display only the AE activity that has occurred in a particular region of a component. This is an appropriate way of viewing the source-location data whenever AE activity appears clustered within a region. Examples of such displays will be given below for each monitored component.

4.4 Task 5: Data Analysis and Documentation

This task is divided into four separate discussions, corresponding to four of the components which were instrumented.

1. Recirculation Bypass Line

The relative positions of the sensors and welds are shown in Fig. 5b. Of all the components being monitored, the recirculation bypass line (RB) has given the least amount of AE, about 700 AE events annually, compared with an average of 4000 events annually on each of the other components. Up to December, 1978, the monitoring of RB has not given any indications of AE sources near either of the two welds which are within the monitored region. Most of the AE has originated in a 10 cm section, that is centered at 48 cm. The activity in this section increases with variations in both temperature and vibration level. For example, it was observed that the AE event distribution shown in Fig. 7 accumulated during two separate 6 day periods when the vibration levels ranged from three to five times greater than the average level.

4.4 Task 5: Data Analysis and Documentation (Cont'd)

1. Recirculation Bypass Line (Cont'd)

The ultrasonic examinations of the RB welds during both the 1977 and 1978 in-service inspections (ISI) produced no reportable indications of discontinuities. During the next ISI, the pipe will also be inspected at the 48 cm location.

It is worth noting that on February 23, 1978, guard sensor, S₁₅, showed an occasional sustained increase in background noise.

(The ambient signal level is normally between 50 and 70 mv and the threshold is set at 100 mv.) The signal consistently exceeded 600 mv. The next day considerable burst emission and an overall increase in background signal was observed on all RB sensors. Prior to the background signals all exceeding the set threshold, 2700 events were rejected as invalid data sets on RB. Eventually, the background signals increased to 400 mv. On the oscilloscope, burst emission was observed to be occurring at a rate equal to five times the recirculation pump speed.

It was concluded that the AE behavior probably indicated leakage associated with a pump problem. Plant operations personnel were informed of the observation. It was not until March 5, 1978, that excessive drywell leakage caused the plant to shutdown. Maintenance during the outage found the recirculation cross-connect valve to be leaking and the recirculation discharge pump seals to be worn. Following repairs, and start-up, the background noise on RB returned to normal. This experience clearly demonstrated the sensitivity of AE leak detection in the bandwidth 400-800 kHz.

2. Core-spray B Line

Accepted AE events are occurring at a rate of approximately 4000 per year on the "B-loop" core-spray line (CSB). The monitored segment of CSB contains four welds, whose relative positions are shown in Figure 5a. There have been very few emissions from the vicinity of Weld #3. However, the general area of Welds #1 and #2, which are 15 cm apart, has been the location of many emissions.

4.4 Task 5: Data Analysis and Documentation (Cont'd)

2. Core-spray B Line (Cont'd)

Approximately 75% of all the AE recorded for CSB occurred during noticeable variations in the temperature of the pipe and/or the reactor coolant flow rate. An example of this is shown in Fig. 8. The activity appears to be centered at W_2 ; however, because of the large diameter of the pipe, there are overlapping zones of possible source locations for Welds #1 and #2. This uncertainty is further complicated by the fact that the heat-affected zones of the welds are the anticipated sources. Therefore, the source distribution shown in Fig. 8 could be predominantly associated with Weld #1. The time record of the occurrence of the events is given in Fig. 9. The time interval in which the sharp increase in activity occurs corresponds to, within several hours, a time interval when the recorded temperature of CSB decreased 80°F and then returned to normal operating temperature.

During a controlled plant run-down, twenty events occurred near Weld #4. Although relatively little AE has been otherwise detected there, it is important to note AE activity that is associated with cooling. During cooling of the component, crack closure stresses would cause cracking of corrosion particles, thereby giving repetitive AE signals.

The ISI results for the "B" core-spray line are available only from the 1978 refueling outage, because the line was freshly installed during the previous refueling outage. The results can be summarized as follows:

Weld #1 - 29 ultrasonic indications, of which 7 were 100% DAC or larger, but were classified as geometric reflectors. The remaining 22 were classified as insignificant.

4.4 Task 5: Data Analysis and Documentation (Cont'd)

2. Core-spray B Line (Cont'd)

Weld #2 - radiographically examined - no reportable indications

Weld #3 - radiographically examined - no reportable indications

Weld #4 - some insignificant indications

The AE results are reasonably compatible with the above, especially for Welds #1, #3 and #4. Because of the uncertainty in the location zone, the amount of AE associated with Weld #2 is not known; but it is regularly active. Of course, the sensitivity of the radiographic examination differs with the ultrasonic which differs with the AE.

3. Feedwater Nozzle "A"

This component has exhibited two areas of repetitive AE activity, namely, 0°-40° azimuth and 165°-240° azimuth. The first of these has contributed approximately 65% of the AE on FWA. Although a repetitive source, it sometimes takes several months before appreciable activity occurs. For the most part, this area of activity does not occur when temperature or coolant flow changes. The second area is more active when temperature variations occur. The 1977 ISI report lists two monitor ultrasonic indications for FWA. No indications were reported in 1978.

4. Feedwater Nozzle "E"

Three areas of activity have been recorded. They are shown in Fig. 10 for two 7-day periods. They have the following azimuthal ranges: 0°-35°, 90°-145° and 250°-300°. The last of these was more active during the first six months of monitoring than recently. The area, centered near 30° in Fig. 10, could correspond to a 1/10" long ultrasonic indication reported in the 1977 ISI results. The

4.4 Task 5: Data Analysis and Documentation (Cont'd.)

4. Feedwater Nozzle "E" (Cont'd.)

report lists the location at 36° nozzle azimuth. Figure 11 shows the rapidity with which the AE occurred at that location. The 1977 ultrasonic examination of this nozzle weld gave a total of nine insignificant indications. In 1978, a total of 20 insignificant indications were reported.

Closer comparisons of the AE and ultrasonic results need to be performed. This will be a major objective during the next ISI.

4.5 Task 6: Refueling Outage Correlations

Only one refueling outage took place during the period of this contract. Figure 12 shows the in-service inspection (ISI) nondestructive test results for the April 1978 inspection and compares them with the previous January 1977 inspection results. No insignificant indications were found on any component involved in this contract. This correlates with the results of AE monitoring during the operating cycle. Several locations of insignificant indications were found on the relevant components during ISI. These same locations had a corresponding amount of insignificant AE activity accumulated during the operating cycle. At many locations no indications were observed whatsoever at the time of ISI. These locations also had no AE activity accumulated during the operating cycle.

From the refueling outage, there has been excellent correlation between the results of ISI inspection and accumulated AE activity to a point. There were locations where AE results indicated no need for further inspection because no AE activity accumulated--later confirmed by ISI. There were locations where AE results indicated insignificant indications requiring further inspection--later confirmed by ISI. There has not been to date any location which had a significant amount of AE activity accumulated or any location of significant indications from ISI to either

4.5 Task 6: Refueling Outage Correlations (Cont'd)

prove or disprove the ability of the AE technique to be responsive to significant activity during plant operation and thereby provide a basis by which the areas of no AE activity can be accepted and not inspected.

SECTION 5

Conclusions

Discussion

AE event clusters were observed on all components. Three AE sources are considered to be minor indications of discontinuities. In general, AE activity increased whenever the vibration levels increased or when cooling of the component occurred. Thus far, the AE results are similar to the results of both ultrasonic and radiographic in-service inspections. Although there is not yet any clearly defined quantitative relation between the AE data and the ultrasonic results, there is a correlation which is very promising.

The two monitored welds on the recirculation bypass lines have not produced any regular emission. The 1977 and 1978 ultrasonic examination of these welds produced "no reportable indications" of discontinuities. One weld on the "B" core-spray line has no AE associated with it. Radiographic examination of this weld produced "no reportable indications" of discontinuities. Regular AE activity was found to cluster in the vicinity of two core-spray welds which had a large number of reportable ultrasonic indications. Also, clustered AE has been recorded on the FE nozzle weld near the region of 1/10" long ultrasonic indication.

These general correlations suggest the following argument: those areas showing little or no AE activity need not be examined according to the requirements of routine in-service inspection.

If such an argument can become well received, then on-line AE surveillance can greatly improve the efficiency of traditional ISI techniques.

This can be illustrated by recognizing that, for the four components discussed herein, the AE data would lead to specific recommendations for examination by conventional methods. As of December 1, 1978, of the six

welds on the RB and CSB lines, only three would be cited for detailed ultrasonic examination. Also, only about 50% of FE and FA would be further examined. In other words, the amount of ultrasonic inspection could be reduced by one half; or, a more detailed examination of selected areas could be performed without additional occupational radiation exposure. Progress towards implementing this type of procedure will continue, as additional on-line AE data is eventually compared with further ISI results.

This program has demonstrated that on-line AE monitoring in an operating nuclear power station is feasible, especially for selected regions of certain components. Using the results of AE monitoring of safety relief valves, guidelines are available for selecting which valves should be bench tested or refurbished. Regarding piping and nozzle welds, a correlation with other NDT results in providing for the formulation of rationales that can potentially reduce in-service inspection requirements, leading to improving plant availability. Furthermore, these, in turn, relate to reductions in amounts of occupational radiation exposure.

SECTION 6

Recommendations

The most extensive, in-plant, continuous AE monitoring in a BWR plant is that being conducted under this contract. Over 50 AE sensors are installed on a recirculation bypass line, two core spray lines, two feedwater nozzles and 22 safety related valves. Two state-of-the-art computerized instrumentation systems are being used to regularly condense and correlate the AE data. This experimental in-plant program, which is beginning its third year, offers a genuinely unique basis on which to build towards eventual routine use of on-line AE monitoring. However, continued improvement of both the techniques of monitoring, the process of data interpretation, and irrefutable correlation with ISI require information which is obtainable only through experiments on mockup components, and on an operating component that has a high incidence of significant flaw growth.

In Japan, the Tokyo Electric Power Company (TEPCO) and the Tokyo Shibaura Electric Company (TOSHIBA) have performed some AE experiments on mockup BWR components. Recognizing the direct relevance to the DOE-PECO program, detailed discussions were recently conducted with the Japanese at their facilities. The principal investigator and chief consultant of this DOE-PECO program visited both TEPCO and TOSHIBA and observed their experimental setup. Several day-long technical sessions were held to exchange information. Considerable scrutiny was given to the available results of the Japanese tests on the mockup components. Review of the Japanese work was enlightening, especially with respect to its shortcomings. The Japanese investigators could not be influenced to perform the relevant experiments to overcome these shortcomings because of funding allocation problems. It is recommended that a test program that fills the voids and relates directly to the following essential questions:

1. What are the probable characteristics (rise time, amplitude, frequency content, pulse shape) of a stress wave emission (AE) emanating from flaw activity within the component?
 - a. The AE source (machine notch or fusion lock) should be at elevated temperature. It is reasonable to expect the features of AE from flaw growth to be temperature dependent.
 - b. Precautions should be taken to distinguish the AE emanating from flaw growth from that generated by plastic deformation of the weld attaching specimen and nozzle.
 - c. Distinction should be made between the AE generated by crack growth and that associated with crack closure.
2. What are the AE detection capabilities for sensors positioned on the outer surface of the pressure vessel wall relative to sensors positioned at various places on the outer surface of the component?
3. What is the source location sensitivity and accuracy for different sensor arrays placed on the various component geometrics?

It is also recommended that continuous on-line monitoring be performed on a component that has a high incidence of significant flaw growth. This is recommended to fill the only void of correlation between ISI results and AE activity accumulated during the operating cycle. By providing data which demonstrates the correlation between the indication of significant flaw growth between the ISI results and AE monitoring, the ability of the AE technique to provide the basis to waive needless inspection requirements can be accepted.

There exists no component on the BWR system which has a high incidence of significant flaw growth. Therefore, it is recommended that other operating facilities be investigated in order to locate a component with similar operating conditions but with a high incidence of significant flaw growth. Successfully demonstrating the ability of the AE system on such a component will provide the necessary correlation required for this program.

Figure 1 - Main Steam Safety Relief Valve Schematic

LEAK RESPONSE FROM MSSRV
FIGURE 2

FIGURE 3a
PECO VALVE LEAK MONITOR

FIGURE 3b

PECO VALVE LEAK MONITOR

VALVE #11 BANDWIDTH B 20-JUN-78

09 · 14 · 25

RELATIVE TIME IN DAYS

FIGURE 4
TRANSDUCER SENSITIVITY TRENDS

FIGURE 5a
CORE SPRAY LINE
INSTALLATION

FIGURE 5b
RECIRCULATION BY-PASS
LINE INSTALLATION

FIGURE 6
FEEDWATER NOZZLE
INSTALLATION

FIGURE 7
RECIRCULATION BY-PASS LINE
AE EVENT DISTRIBUTION

FIGURE 8
CORE SPRAY LINE
AE EVENT DISTRIBUTION

FIGURE 9
CORE SPRAY LINE
TIME RECORD OF
AE EVENT ACTIVITY

FIGURE 10
FEEDWATER NOZZLE 'E'
AE EVENT DISTRIBUTION

FIGURE 11
FEEDWATER NOZZLE "E"
TIME RECORD OF AE
EVENT ACTIVITY

Figure 12 - ISI Ultrasonic Inspections

January 1977				April 1978		
Recirc. Bypass	No Recordable Indications	Insignificant Indications (0-100% DAC)	Geometric Reflections	No Recordable Indications	Insignificant Indications (0-100% DAC)	Geometric Reflections
<u>Weld 8 Visual</u>	x			x		
0° Lam Scan	x			x		
45° Scan	x			x		
45° T Scan	x			x		
60° Scan	x			x		
<u>Weld 9 Visual</u>	x			x		
0° Lam Scan	x			x		
45° Scan	x			x		
45° T Scan	x			x		
60° Scan	x			x		
<u>Feedwater "A"</u>						
<u>Mechanized</u>						
21° RC	x			x		
0° Lam Scan	x			NP		
0° Weld Scan			2	NP		
45° Scan	x			NP		
60° Scan	x			NP		
<u>Feedwater "E"</u>						
<u>Mechanized</u>						
21° RC		5			20	3
0° Lam Scan	x			x		
0° Weld Scan		*		NP		
45° Scan		2		x		
60° Scan		1		x		

- Not Performed

* 1/10" long slag-like indication at 36° nozzle azimuth at subsurface intersection of nozzle boss and vessel outside surface under the nozzle to vessel blend radius weld (2" metal path).

Figure 12 - ISI Ultrasonic Inspections (Cont'd)

January 1977				April 1978		
Core Spray "B"	No Recordable Indications	Insignificant Indications (0-100% DAC)	Geometric Reflections	No Recordable Indications	Insignificant Indications (0-100% DAC)	Geometric Reflections
<u>Weld 23 Visual</u>	x			x		
0° Lam Scan	x			x		
45° T Scan	x			x		
45° Scan	x				1	
60° Scan	x				3	
<u>Weld 24 R</u>						
Radiograph	x			x		
<u>Weld 25 R</u>						
Radiograph	x			x		
<u>Weld 26 Visual</u>						
0° Lam Scan	x			x		
45° T Scan	x			x		
45° Scan	x				11	5
50° Scan	x				18	2
Core Spray "A"						
<u>Weld 23 Visual</u>						
0° Lam Scan	x			x		
45° T Scan	x			x		
45° Scan	x				1	
60° Scan	x				3	
<u>Weld 24 R</u>						
Radiograph	x			x		
<u>Weld 25 R</u>						
Radiograph	x			x		
<u>Weld 26 Visual</u>	x			x		

SECTION 7

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