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March 30, 1984

Docket Nos. 50-277
50-278

Dr. Thomas E. Murley, Administrator
Office of Inspection and Enforcement
Region I
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, PA 19406

SUBJECT: Preliminary Report of the Fracture Mechanics Analyses for Weld Acceptability on the Weld Imperfections Identified by Philadelphia Electric Company in Licensee Event Report 2-83-24/1T.

REFERENCES:

1. Licensee Event Report Narrative Description to Dr. T. E. Murley, NRC, from R. S. Fleischmann, PECo., dated October 24, 1983.
2. Licensee Event Report 2-83-24/1T Attachment to Dr. T. E. Murley, NRC, from M. J. Cooney, PECo., dated November 7, 1983.
3. Letter, dated February 17, 1984, from T. T. Martin, NRC, to S. L. Daltroff, PECo., (Inspection No. 50-277/84-05; 50-278/84-05).

Dear Dr. Murley:

On October 24, 1983, Philadelphia Electric Company Engineering and Research Department notified the Electric Production Department that certain Class I piping radiographs were improperly read by Eastern Testing and Inspection, Inc. (ETI) during previous outage modification work. These discrepancies were identified during an audit of ETI by Philadelphia Electric Company's Construction Division Quality Control Section. This notification by the Construction Division identified seven welds with radiographs that had been improperly



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1/1

examined. The identification and location of these seven welds are as follows:

| No. of Welds | Peach Bottom Unit No. | Location | Weld ID Number |
|---|---|---|---|
| 3 | 2 | CRD Scram Discharge Volume | 655-2-12<br>655-2-39<br>655-2-40 |
| 3 | 3 | Reactor Water Cleanup System Between Inner Isolation Valve and Drywell Penetration | 686-3-7<br>686-3-8<br>686-3-11 |
| 1 | 2 | Reactor Water Cleanup System Between Inner Isolation Valve and Drywell Penetration | 686-2-15A |

Upon identification of these seven radiographic interpretation deficiencies, the ETI radiographs were re-interpreted and unacceptable indications were confirmed by a Philadelphia Electric Company NDE Level III. The weld deficiencies were characterized as having indications of incomplete fusion, porosity, or crater cracks, or a combination thereof. Defects were estimated to be 0.010" to 0.060" in depth and 0.5" to 3.0" long.

Philadelphia Electric Company subsequently requested Stone and Webster Engineering Corporation to evaluate the effect of these indications on fitness of the weldments for continued service. Stone and Webster Engineering Corporation's initial evaluation determined that it would take more than 1000 cycles to propagate each of the seven aforementioned defects to twice their current depth. These defects, even at twice their current depth, would be acceptable for continued operation. Therefore, these seven reported radiographic indications were determined as acceptable for continued operation for the life of the plant.

Immediately following the discovery of these seven discrepancies, Philadelphia Electric Company reviewed the 131 welds that had been radiographed by ETI in safety-related systems at the Peach Bottom facility. Philadelphia Electric Company's review identified sixteen additional welds with previously unidentified defects. The identification and location of these sixteen additional welds are as follows:

| No. of Welds | Unit No. | Location | Weld ID Number |
|---|---|---|---|
| 8 | 3 | CRD Scram Discharge Volume | 655-3-3<br>655-3-4<br>655-3-5<br>655-3-8<br>655-3-16<br>655-3-24<br>655-3-27<br>655-3-30 |
| 5 | 2 | Core Spray System | 389-2-1<br>389-2-4<br>389-2-7<br>389-2-11<br>389-2-15 |
| 1 | 3 | Core Spray System | 389-3-7 |
| 2 | 2 | Feedwater System | 381-2-8<br>381-2-14 |

Following identification of these sixteen additional radiographic interpretation deficiencies, Philadelphia Electric Company requested Stone and Webster Engineering Corporation to perform fracture mechanics analyses for weld acceptability determination for all twenty-three welds with imperfections.

Licensee Event Report Narrative Description dated October 24, 1983, (Dr. T. E. Murley, NRC, from R. S. Fleischmann, PECo.) and Licensee Event Report 2-83-24/1T attachment dated November 7, 1983, (Dr. T. E. Murley, NRC from M. J. Cooney, PECo.) notified the Nuclear Regulatory Commission of these findings, and a commitment was made to submit to the NRC a summary report of the fracture mechanics evaluation upon

completion of Stone and Webster Engineering Corporation's analyses.

On January 26, 1984, the NRC requested that PECo provide, as quickly as possible, an update as to the status of the fracture mechanics analyses. In addition, the NRC requested that Philadelphia Electric Company perform microdensitometer readings to confirm the size and depth of the weld imperfections.

In response to the NRC's request to perform microdensitometer readings, Philadelphia Electric Company and the NRC selected five welds for confirmatory examination.

Microdensitometer readings were performed on February 14, 1984. Unfortunately, the readings were performed with a machine that had not been calibrated. Philadelphia Electric Company is in the process of re-performing the readings with a calibrated densitometer. These readings are expected to confirm the findings of February 14, 1984, and are expected to be complete and reported within ten (10) days.

A review of the location of each of the 23 welds with imperfections revealed that all the welds were located in the following four systems of the Peach Bottom Atomic Power Station:

- Scram Discharge Volume
- Reactor Water Cleanup System
- Core Spray System
- Feedwater System

Conference calls were made to the Regional Office of the NRC reporting on the preliminary fracture mechanics evaluation performed on the 'worst case' weld imperfection in each of the four systems. These findings were reported to the NRC, Region I, on February 1, 2, 10 and 17, 1984 for the Scram Discharge Volume, Reactor Water Clean-Up, Core Spray and Feedwater systems, respectively.

The following is a summary of the description of the welds that were determined by analysis to be the 'worst case' imperfections within each of the four systems:

| System | Weld No. | Material |
|---|---|---|
| Scram Discharge Volume | 655-2-39 | 6 inch schedule 80 pipe ASTM/A-106/Grade B Carbon Steel |
| Reactor Water Cleanup | 686-3-8 | 6 inch schedule 80 pipe ASTM/A-312/Type 316L Stainless Steel |
| Core Spray | No particular weld was determined 'worst case' | 10 inch and 12 inch schedule 80 pipe ASTM/A-316/Type 304 Stainless Steel |
| Feedwater | No particular weld was determined 'worst case' | 10 inch schedule 100 pipe ASTM/A-106/Grade B Carbon Steel |

Additional details of the worst case weld, the methods utilized during the evaluation, and the findings of these evaluations are described in the attached 'Radiographic Indications and Fitness of Affected PBAPS, Units 2 and 3, Systems for Continued Service' (Preliminary Evaluation).

The findings of the report are summarized on Table 2, 'Minimum Number of Loading Cycles for Propagation of Postulated Cracks to an Acceptable Limit'.

In accordance with Table 2, the weld defects that can operate safely with the least number of loading cycles (240) until the propagation of the crack would approach the acceptable limit of the analysis are both in the Feedwater system of Peach Bottom Atomic Power Station Unit 2.

Please be advised that the preliminary fracture analyses performed on each of these two welds were based on code allowable stresses, when in fact, the actual stresses on these welds are considerably less than code allowable. Stone and Webster Engineering Corporation, the consultant performing the fracture mechanics analyses on each of the twenty-three (23) welds, has been provided with the actual stresses on both Feedwater System welds. It is expected that the actual stress fracture mechanics analyses on these two welds will reveal that these Feedwater System welds can experience the calculated number

of loading cycles projected during the design life of Peach Bottom Atomic Power Station and still be acceptable for continued operation.

In addition, as the calculated number of loading cycles for the worst weld defect within the other three systems, scram discharge volume, core spray and reactor water cleanup are in excess of the design life of Peach Bottom Atomic Power Station, Philadelphia Electric Company, based on the attached preliminary evaluation, has concluded that the weld defects within these systems are also acceptable for the design life of the plant. A final determination will be made by Philadelphia Electric Company upon receipt and review of Stone and Webster Engineering Corporation's final report on the fracture analyses.

Based on conversations with our consultant, Philadelphia Electric Company expects to receive a final report on the fracture analyses of all twenty-three (23) welds by Stone and Webster Engineering Corporation by May 1, 1984. Submission of this final summary report to the NRC will take place following our review of that document.

Should you require additional information, please do not hesitate to contact us.

Very truly yours,

W. T. Ullrich
Superintendent
Nuclear Generation Division

Attachment

cc: A. R. Blough, Site Inspector
NRC Document Control Desk

ATTACHMENT

PHILADELPHIA ELECTRIC COMPANY

# RADIOGRAPHIC INDICATIONS AND FITNESS OF AFFECTED PBAPS, UNITS 2 AND 3, SYSTEMS FOR CONTINUED SERVICE

(Preliminary Evaluation)

for

PEACH BOTTOM ATOMIC FOWER STATION

UNITS 2 AND 3

DOCKET NUMBERS 50-277 and 50-278

Submitted To

THE UNITED STATES NUCLEAR REGULATORY COMMISSION

March 1984

# RADIOGRAPHIC INDICATIONS AND FITNESS OF AFFECTED PBAPS, UNITS 2 AND 3, SYSTEMS FOR CONTINUED SERVICE

(Preliminary Evaluation)

## 1.0 INTRODUCTION

A number of radiographic indications in the scram discharge volume, reactor water cleanup, core spray, and feedwater systems were reported by Philadelphia Electric Company (PECo). A preliminary fitness for service analysis has been performed by Stone & Webster Engineering Corporation (SWEC) based on fatigue crack propagation methodology consistent with ASME Section XI (Ref. 1) rules. The results of this preliminary analysis are briefly summarized below. Each of the following four subsections includes stress analysis, cyclic crack growth analysis, and acceptance evaluation.

The effect of temperature, environment, and residual stresses is taken into account as appropriate. Both constant (self-similar growth) and variable (per Ref. 2) flaw depth to length ratios are used.

## 2.0 SCRAM DISCHARGE VOLUME (SDV)

### 2.1 Stress Analysis

Unit 2 contains three welds with a total of six indications. Two of the welds are girth butt welds joining 90-deg L.R. elbows to straight pipe, and the third is a girth butt weld joining straight pipe members.

Unit 3 contains 8 welds with a total of 17 indications. Five of the welds are girth butt welds joining 90-deg L.R. elbows to straight pipe, two join 45-deg elbows to straight pipe, and one is a butt weld joining an 8 in. x 12 in. weldolet to a tank.

The following cyclic stress contributions are evaluated:

1. Stresses due to the free end displacement caused by thermal expansion (based upon Code, Ref. 3, allowable stresses)

2. Internal pressure stress

3. Transient through wall thermal stress due to fluid temperature change (the linear part ($\Delta T_1$)).

These stress contributions are conservatively considered to act concurrently.

The stress components normal to the face of the indication are evaluated. These stresses are separated into membrane and bending components. In the case of elbow/pipe girth butt welds, the maximum permissible moment load is determined from the Code relationship (Ref. 3, Para. 104.8.3, Eq. 13a)

$$\frac{iM}{Z} \leq S_A$$

The maximum membrane plus bending stress is then calculated from the relationship given by Rodabaugh, Iskander, and Moore (Ref. 4) where the maximum stress at the end of an elbow is

$$S_{max} = 1.48 \left(\frac{Rt}{r^2}\right)^{-0.4} \frac{M}{Z}$$

where R, t, and r are the bend radius, nominal thickness, and mean radius of the elbow. M/Z is the permissible nominal stress $S_A/i$. This value of $S_{max}$ is separated into membrane and bending as follows:

$$S_m = S_A /i \quad \text{and} \quad S_b = S_{max} - S_A /i$$

The pressure stress (Ref. 3, Para. 102.3.2, d) is taken as entirely membrane and normal to the girth weld.

The $\Delta T_1$ contribution is determined through a time-dependent heat transfer analysis. This stress is compressive at the inner surface and tensile at the outer surface.

The free end displacement stresses and the pressure stresses are adjusted by the ratio of the nominal thickness over the minimum thickness ($t_{min}$) where:

$$t_{min} = 0.875t_n - A$$

$t_n$ is the nominal thickness and A is the corrosion allowance.

The stresses for the worst case (weld No. 655-2-39) are shown in Table 1.

2.2 Cyclic Crack Growth Analysis

The fatigue crack growth analysis addressed the worst case, i.e., weld #655-2-39 for which a 60-mil-deep indication was reported. In the calculation, a 0.125-in.-deep crack was postulated to allow for any uncertainties in the indication sizing.

Fatigue crack growth rate data of Ref. 1 developed for ferritic materials in high-temperature water environment were used in the crack propagation analysis.

Using the above conservatively estimated stresses, the crack depth of the propagating postulated crack was evaluated as a function of a number of the load cycles for constant and variable crack depth to length ratios.

2.3 Acceptance Evaluation

The acceptance criterion of IWB-3612(a) of Ref. 1, i.e.,

$$K_I \leq K_{Ia}/\sqrt{10}$$

was used as suggested in paragraph IWB-3620.

Here $K_I$ is the maximum stress intensity factor and $K_{Ia}$ is the crack arrest toughness given by Figure A4200-1 in Appendix A of Ref. 1.

Since the indications were reported in the weld metal, fracture toughness of the weld metal, SFA5.1 E7018, was used in the analysis.

The evaluation has determined that the above criterion is met even when the number of loading cycles is greater than 1000 (see Table 2).

3.0 CORE SPRAY (CS)

3.1 Stress Analysis

The Unit 2 CS contains five welds with eight indications. Four of these welds are at pipe/elbow girth butt welds and the fifth is a girth butt weld adjacent to the RVP nozzle safe end (on the pipe side). The stresses for this latter weld are shown in Table 1. These stresses are determined based on Code permissible moments and in addition are determined from the loads taken from the PECo stress analysis.

The Unit 3 CS contains one weld with two indications. The weld is a pipe/elbow girth butt weld. The indications are such that for the purpose of this preliminary evaluation, the Unit 2 result suffices to cover Unit 3.

The CS system does not experience any operating transients (cyclic stress) due to CS operation. The CS system experiences RPV pressure cycling and free end expansion stresses caused by RPV expansion/contraction while the CS pipe remains essentially at ambient temperature.

3.2 Cyclic Crack Growth Analysis

The depth of the reported indications varied between 5 and 25 mils. An initial postulated crack depth of 1/16 in. was assumed in this analysis. The calculations were performed for stresses based on the stress analysis submitted by PECo and stresses based on the Code allowable values.

Crack growth rate data from Ref. 5 and high-temperature properties of 308 weld stainless steel are used in this analysis.

### 3.3 Acceptance Evaluation

The acceptance criterion of IWB-3640 (Ref. 1) which covers austenitic piping applies to the core spray pipe welds. According to this criterion, the allowable end-of-evaluation period flaw depth to thickness ratio in this case is 0.75. It has been determined that under the most conservative assumptions, i.e. using stresses based on Code permissible moments, the number of cycles satisfying the above criterion exceeds 12,000 (see Table 2).

## 4.0 REACTOR WATER CLEANUP (RWCU)

### 4.1 Stress Analysis

THe RWCU system of Unit 2 contains one weld with two indications. The weld is a pipe/penetration girth butt weld.

The RWCU system of Unit 3 contains three welds with five indications; two of those welds are pipe/elbow girth butt welds and the stresses are determined in the same manner as for the SDV system. The third weld is a girth butt weld joining two 45-deg elbows. These elbows are out of plane and thus the longitudinal stresses are considered entirely membrane. The maximum stress is determined through the series solution of Rodabaugh and George (Ref. 6) and is found to be:

$S_{max} \leqslant 2.5\ M/Z$ in the direction normal to the weld.

The computed cyclic stresses are shown in Table 1.

### 4.2 Cyclic Crack Growth Analysis

The analysis was performed for two worst cases:

1. The deepest postulated crack of 0.125 in. (weld 686-2-15A) and

2. The highest stress level (weld 686-3-8).

The latter was found to be a limiting case.

Crack growth rate data used for the RWCU system were the same as for the core spray. Results presented below refer to weld 686-3-8.

### 4.3 Acceptance Evaluation

Using the acceptance criterion described in Section 3.3, it was found that at least 40,000 cycles are required for a postulated crack to grow to the maximum allowable depth of 0.25 in., i.e., to 75 percent of the thickness (Table 2).

## 5.0 FEEDWATER STARTUP BYPASS (FW)

### 5.1 Stress Analysis

The Unit 2 bypass contains two welds with three indications. One weld is at a girth butt weld joining a welded end valve (900 lb) to the large end of a 10 in. x 8 in. Schedule 100 reducer. The expansion stresses are based on the maximum Code permissible moment at the small end of the reducer, adjusted upward by 12 percent based on PECo stress analysis results to account for the change in moment arm. The longitudinal pressure stress contains a bending component caused by the greater thickness of the valve.

The other weld is at a pipe/elbow girth butt weld. The stresses are based on the permissible Code moments and are computed in the manner described under the SDV system.

The stresses for both of these welds are shown in Table 1.

### 5.2 Cyclic Crack Growth

Postulated cracks with an initial depth of 1/16 in. (reported depth was 0.010 to 0.015 in.) were assumed to grow due to the cyclic loading into the weld metal, SFA5.1 E7018.

The crack growth rate for ferritic materials in water reactor environment was used in this analysis.

### 5.3 Acceptance Evaluation

The acceptance criterion given in Section 2.3 required evaluation of $K_{Ia}$ from Figure A4200-1 in Appendix A of Ref. 1.

For the weld metal, SFA 5.1 E7018, the temperature difference, $T-RT_{NDT}$, was estimated and the crack arrest fracture toughness, $K_a$, was found from Figure A4200-1.

Based on the crack propagation results, it was shown that the postulated cracks will remain with the allowable limits for at least 240 cycles based on the assumption of self-similar growth and 550 cycles based on the assumption of variable flaw depth to length ratio (Table 2).

## 6.0 CONCLUSIONS

The preliminary results are summarized in Table 2, together with the estimated number of loading cycles over a 40-year period.

Based on these results, the following conclusions can be drawn:

1. In the scram discharge volume, core spray, and reactor water cleanup systems, the number of loading cycles required to propagate postulated cracks to an acceptable limit considerably exceeds the estimated total number of cyclic events over the plant lifetime.

2. Based on cyclic stress data presented in Section 5.1, postulated cracks in the feedwater startup bypass line are not expected to reach the allowable limit within at least 18 years of plant operation. A more realistic assessment based on variable crack depth to length ratio shows that postulated cracks will remain within the allowable limit over the plant lifetime.

# REFERENCES


1. Rules for Inservice Inspection of Nuclear Power Plant Components, ASME Boiler and Pressure Vessel Code, 1983 and Winter 1983 Addenda.

2. Newman, J. C. and Raju, I. S., Engineering Fracture Mechanics, v. 15, No. 1-2, p. 182, 1981.

3. Power Piping, American National Standard, ANSI B31.1-1973 with Addenda to and including Summer 1975.

4. Rodabaugh, E. C.; Iskander, S. K.; and Moore S. E.; End Effects on Elbows Subjected to Moment Loadings, ORNL/Sub-2913/7, 1978.

5. Bamford, W. H., Journal of Pressure Vessel Technology, v. 101, p. 73, February 1979.

6. Rodabaugh, E. C. and George, H. H., Trans. ASME, v. 79, 1957.

TABLE 1

APPLIED STRESSES FOR THE PRELIMINARY FATIGUE ANALYSIS

| System | Weld Number | Pressure Stress $\sigma_m$(ksi) | Pressure Stress $\sigma_b$(ksi) | Expansion Stress $\sigma_m$(ksi) | Expansion Stress $\sigma_b$(ksi) | Thermal Stress $\sigma_m$(ksi) | Thermal Stress $\sigma_b$(ksi) | Sum of Cyclic Stress $\sigma_m$(ksi) | Sum of Cyclic Stress $\sigma_b$(ksi) |
|---|---|---|---|---|---|---|---|---|---|
| RWCU | 686-2-15A | 5.56 | 0 | 4.58 | 0 | 0 | 0 | 10.14 | 0 |
| RWCU | 686-3-7 | 5.56 | 0 | 7.80 | 10.90 | 0 | 0 | 13.36 | 10.90 |
| RWCU | 686-3-8 | 5.56 | 0 | 9.39 | 11.87 | 0 | 0 | 14.95 | 11.87 |
| RWCU | 686-3-11 | 5.56 | 0 | 10.49 | 0 | 0 | 0 | 16.03 | 0 |
| SDV* | 655-2-12 | | | | | | | | |
| SDV* | 655-2-39 | 5.91 | 0 | 13.70 | 15.40 | 0 | 0 | 19.61 | 15.40 |
| SDV* | 655-2-40 | | | | | | | | |
| SDV | 655-3-3 | | | | | | | | |
| SDV | 655-3-4 | | | | | | | | |
| SDV | 655-3-5 | | | | | | | | |
| SDV | 655-3-8 | | | | | | | | |
| SDV | 655-3-16 | | | | | | | | |
| SDV | 655-3-24 | | | | | | | | |
| SDV | 655-3-27 | | | | | | | | |
| SDV | 655-3-30 | | | | | | | | |
| CS | 389-2-1 | 5.26 | 0 | 8.19 | 0 | 0 | 0 | 13.46 | 0 |
| CS* | 389-2-1 | 5.26 | 0 | 19.51 | 0 | 0 | 0 | 24.77 | 0 |
| CS | 389-2-4 | | | | | | | | |
| CS | 389-2-7 | | | | | | | | |
| CS | 389-2-11 | | | | | | | | |
| CS | 389-2-15 | | | | | | | | |
| CS | 389-3-7 | | | | | | | | |
| FW* | 381-2-8 | 7.71 | 0 | 18.62 | 20.05 | 0 | 0 | 26.33 | 20.05 |
| FW* | 381-2-14 | 7.71 | 8.67 | 17.13 | 0 | 0 | 0 | 24.84 | 8.67 |

*Expansion stresses are based on code allowables

TABLE 2

MINIMUM NUMBER OF LOADING CYCLES FOR PROPAGATION OF POSTULATED CRACKS TO AN ACCEPTABLE LIMIT

| Unit | System | Calc. Number of Cycles to Acceptable Limit: Const $\frac{a}{l}$ | Calc. Number of Cycles to Acceptable Limit: Variable $\frac{a}{l}$ | Estimated Number of Cycles per Year | Estimated Number of Cycles Over 40 Years |
|---|---|---|---|---|---|
| 2 and 3 | Scram discharge volume | >1,000 | >1,000 | 15 | 600 |
| 2 and 3 | Core spray | >12,000 | 50,000 | 13 | 520 |
| 2 and 3 | Reactor water cleanup | >40,000 | >>40,000 | 28 | ≃1,100 |
| 2 | Feedwater startup bypass | 240 (~18 yrs) | 550 (~42 yrs) | 13 | 520 |