WCAP 10501

TECHNICAL BASES FOR ELIMINATING LARGE PRIMARY LOOP PIPE RUPTURES AS THE STRUCTURAL DESIGN BASIS FOR CALLAWAY & WOLF CREEK PLANTS

W. K. Ma

S. A. Swamy

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APPROVED: J. N. Chirigos, Manager
Structural Materials
Engineering

APPROVED: E R Johnson
E. R. Johnson, Manager
Structural and Seismic
Development

APPROVED: John J McInerney
J. J. McInerney, Manager
Mechanical Equipment and
Systems Licensing

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1.0 INTRODUCTION

1.1 Purpose

The current structural design basis for the reactor coolant system (RCS) primary loop requires that pipe breaks be postulated as defined in the approved Westinghouse Topical Report WCAP 8082, Reference 1. In addition, protective measures for the dynamic effects associated with RCS primary loop pipe breaks have been incorporated in the Callaway and Wolf Creek plants design. However, Westinghouse has demonstrated on a generic basis that RCS primary loop pipe breaks are highly unlikely and should not be included in the structural design basis of Westinghouse plants (see Reference 2). The purpose of this report is to demonstrate that the generic evaluations performed by Westinghouse are applicable to the Callaway and Wolf Creek plants. In order to demonstrate this applicability, Westinghouse has performed a comparison of the loads and geometry for the Callaway and Wolf Creek plants with envelope parameters used in the generic analyses (Section 3.0); fracture mechanics evaluation (Section 4.0); determination of leak rates from a through-wall crack (Section 5.0), and fatigue crack growth evaluation (Section 6.0). Conclusions are presented in Section 7.0.

1.2 Scope

This report applies to the Callaway and Wolf Creek plants reactor coolant system primary loop piping. It is intended to demonstrate that specific parameters for the Callaway and Wolf Creek plants are enveloped by the generic analysis performed by Westinghouse in WCAP-9570 (Reference 3) and accepted by the NRC as noted in a letter from Harold Denton dated May 2, 1983 (Reference 4).

1.3 Objectives

The conclusions of this report (Reference 3) support the elimination of RCS primary loop pipe breaks for the Callaway and Wolf Creek plants. In order to validate this conclusion the following objectives must be achieved:

a. Demonstrate that Callaway and Wolf Creek plants parameters are enveloped by generic Westinghouse studies.

b. Demonstrate that margin exists between the critical crack size and a postulated crack which yields a detectable leak rate.

c. Demonstrate that there is sufficient margin between the leakage through a postulated crack and the leak detection capability of the Callaway and Wolf Creek plants.

d. Demonstrate that fatigue crack growth is negligible.

1.4 Background Information

Westinghouse has performed considerable testing and analysis to demonstrate that RCS primary loop pipe breaks can be eliminated from the structural design basis of all Westinghouse plants. The concept of eliminating pipe breaks in the RCS primary loop was first presented to the NRC in 1978 in WCAP 9283 (Reference 5). This Topical Report employed a deterministic fracture mechanics evaluation and a probabilistic analysis to support the elimination of RCS primary loop pipe breaks.

This approach was then used as a means of addressing Generic Issue A-2 and Asymmetric LOCA Loads. Westinghouse performed additional testing and analysis to justify the elimination of RCS primary loop pipe breaks. As a result of this effort, WCAP 9570 was submitted to the NRC. The NRC evaluated the technical merits of this concept and prepared a draft SER in late 1981 endorsing this concept. Additionally, both Harold Denton and the ACRS have endorsed the technical acceptability of the Westinghouse evaluations. Specifically, in a May 2, 1983 letter (Reference 4) Harold Denton states that ". . . it is technically satisfied with Westinghouse Topical Report 9570 Rev. 2 . . . ." Additionally, the ACRS stated in a June 14, 1983 letter (Reference 6) that "... there is no known mechanism in PWR primary piping material for developing a large break without going through an extended period during which the crack would leak copiously."

The NRC funded research through Lawrence Livermore National Laboratory (LLNL) to address this same issue using a probabilistic approach. As part of the LLNL research effort, Westinghouse performed extensive evaluations of specific plant loads, material properties, transients, and system geometries to demonstrate that the analysis and testing previously performed by Westinghouse and the research performed by LLNL applied to all Westinghouse plants including Callaway and Wolf Creek plants (References 7 and 8). The results from the LLNL study were released at a March 28, 1983 ACRS Subcommittee meeting. These studies which are applicable to all Westinghouse plants east of the Rocky Mountains, determined the mean probability of a direct LOCA (RCS primary loop pipe break) to be 10^{-10} per reactor year and the mean probability of an indirect LOCA to be 10^{-7} per reactor year. Thus, the results previously obtained by Westinghouse (Reference 5) were confirmed by an independent NRC research study.

The above studies establish the technical acceptability for eliminating pipe breaks from the Westinghouse RCS primary loop. The LLNL study has been shown applicable to the Callaway and Wolf Creek plants by inclusion of plant specific data. This report will demonstrate the applicability of the Westinghouse generic evaluations to the Callaway and Wolf Creek plants.

2.0 OPERATION AND CHEMICAL STABILITY OF THE PRIMARY COOLANT SYSTEM

The Westinghouse reactor coolant system primary loop has an operating history (over 400 reactor years) which demonstrates its inherent stability characteristics. Additionally, there is no history of cracking in RCS primary loop piping. In addition to the fracture resistant materials used in the piping system, the chemistry of the reactor coolant is tightly controlled and variations in temperatures, pressure and flow during normal operating conditions are insignificant.

As stated above, the reactor coolant chemistry is maintained within very specific limits. For example, during normal operation oxygen in the coolant is limited to less than [ ]+ This stringent oxygen limit is +a,c,e
achieved by controlling charging flow chemistry and maintaining hydrogen in the reactor coolant at a concentration of [ ]+ The +a,c,e
oxygen concentration in the reactor coolant is verified by routine sampling and chemical analysis. Halogen concentrations are also stringently controlled by maintaining concentrations of chlorides and fluorides at or below [ ]+ This concentration is assured by controlling charging flow +a,c,e
chemistry and specifying proper wetted surface materials. Halogen concentrations are also verified by routine chemical sampling and analysis.

In order to ensure dynamic system stability, reactor coolant parameters are stringently controlled. Temperature during normal operation is maintained within [ ]+ by control rod position. Pressure is controlled by +a,c,e
pressurizer heaters and pressurizer spray, to a variation of less than [ ]+ for steady state conditions. The flow characteristics of the +a,c,e
system remain constant during a fuel cycle because the only governing parameters, namely system resistance and the reactor coolant pump characterisitcs are controlled in the design process. Additionally, Westinghouse has instrumented typical reactor coolant systems to verify the flow characteristics of the system.

The reactor coolant system, including piping and primary components, is designed for normal, upset, emergency and faulted condition transients. The design requirements are conservative relative to both the number of transients and their severity.

3.0 PIPE GEOMETRY AND LOADING

A segment of the primary coolant hot leg pipe is shown in Figure 1. This segment is postulated to contain a circumferential through-wall flaw. The inside diameter and wall thickness of the pipe are 29.0 and 2.45 inches, respectively. The pipe is subjected to a normal operating pressure of [ ]+ psig. The design calculations indicate that the junction of the [ ]+ is most highly stressed. At this location the axial load, F, and the total bending moment M, are [ ]+ kips (including the axial force due to pressure) and [ ]+ in-kips, respectively. Figure 2 identifies the loop weld locations. The material properties and the loads at these locations resulting from Deadweight, Thermal Expansion and Safe Shutdown Earthquake are indicated in Table 1. The method of obtaining these loads can be briefly summarized as follows:

+a,c,
+a,c,
+a,c,
+a,c,

The axial force F and transverse bending moments, M_y and M_z, are chosen for each static load (pressure, deadweight and thermal) based on elastic-static analyses for each of these load cases. These pipe load components are combined algebraically to define the equivalent pipe static loads F_s, M_{ys}, and M_{zs}. Based on elastic SSE response spectra analyses, amplified pipe seismic loads, F_d, M_{yd}, M_{zd} are obtained. The maximum pipe loads are obtained by combining the static and dynamic load components as follows:

$$F = \left|F_s\right| + \left|F_d\right|$$

$$M = \sqrt{M_y^2 + M_z^2}$$

where

$$M_y = \left|M_{ys}\right| + \left|M_{yd}\right|$$

$$M_z = \left|M_{zs}\right| + \left|M_{zd}\right|$$

The corresponding geometry and loads used in the reference report (Reference 3) are as follows: inside diameter and wall thickness are 29.0 and 2.5 inches; axial load and bending moment are [ ]+ inch kips. +a,c,e The outer fiber stress for Callaway and Wolf Creek plants is [ ]+ ksi, +a,c,e while for the reference report it is [ ]+ ksi. This demonstrates +a,c,e conservatism in the reference report which makes it more severe than the Callaway and Wolf Creek projects.

4.0 FRACTURE MECHANICS EVALUATION

4.1 Global Failure Mechanism

Determination of the conditions which lead to failure in stainless steel must be done with plastic fracture methodology because of the large amount of deformation accompanying fracture. A conservative method for predicting the failure of ductile material is the [

]+ This methodology has been shown to be applicable to ductile piping through a large number of experiments, and will be used here to predict the critical flaw size in the primary coolant piping. The failure criterion has been obtained by requiring [+a,c,e

]+ (Figure 3) when loads are applied. The detailed development is provided in Appendix A, for through-wall circumferential flaw in a pipe with internal pressure, axial force and imposed bending moments. The [+a,c,e

]+ for such a pipe is given by: +a,c,e

[ ]+ +a,c,e

+a,c,e

The analytical model described above accurately accounts for the piping internal pressure as well as imposed axial force as they affect [ ]+ Good agreement was found between the analytical predictions and the experimental results [9].

4.2 Local Failure Mechanism

The local mechanism of failure is primarily dominated by the crack tip behavior in terms of crack-tip blunting, initiation, extension and finally crack instability. Depending on the material properties and geometry of the pipe, flaw size, shape and loading, the local failure mechanisms may or may not govern the ultimate failure.

The stability will be assumed if the crack does not initiate at all. It has been accepted that the initiation toughness, measured in terms of J_{IN} from a J-integral resistance curve is a material parameter defining the crack initiation. If, for a given load, the calculated J-integral value is shown to be less than J_{IN} of the material, then the crack will not initiate. If the initiation criterion is not met, one can calculate the tearing modulus as defined by the following relation:

$$T_{app} = \frac{dJ}{da} \quad \frac{E}{\sigma_f^2}$$

where

T_{app} = applied tearing modulus

E = modulus of elasticity

σ_f = [ ]+ (flow stress)

a = crack length +a,c,e

[ ]+ +a,c,e

In summary, the local crack stability will be established by the two step criteria:

$J < J_{IN}$

$T_{app} < T_{mat}$ if $J > J_{IN}$

4.3 Results of Crack Stability Evaluation

Figure 4 shows a plot of the [ ]+ as a function of throughwall circumferential flaw length in the [ ]+ of the main coolant piping. This [ ]+ was calculated for Callaway and Wolf Creek plants data of a pressurized pipe at [ ]+ with ASME Code minimum [ ]+ properties. The maximum applied bending moment of [ ]+ in-kips can be plotted on this figure, and used to determine the critical flaw length, which is shown to be [ ]+ inches. This is considerably larger than the [ ]+ inch reference flaw used in Reference 3. +a,c,e +a,c,e +a,c,e +a,c,e +a,c,e +a,c,e

[ ]+ +a,c,e

[

]+ Therefore, it can be concluded that a postulated [ ]+ inch through-wall flaw in the Callaway and Wolf Creek loop piping will remain stable from both a local and global stability standpoint. +a,c,e

5.0 LEAK RATE PREDICTIONS

Leak rate calculations were performed in Reference 3 using an initial throughwall crack [ ]+. The computed leak rate was [ ]+ based on the normal operating pressure of [ ]+ psi. [ ]+ +a,c

This computed leak rate [ ]+ significantly exceeds the smallest detectable leak rate for the plant. The Callaway and Wolf Creek plants have RCS pressure boundary leak detection system which is consistent with the requirements of Regulatory Guide 1.45 and can detect leakage of 1 gpm in one hour. There is a factor of [ ]+ between the calculated leak rate and the Callaway and Wolf Creek plants leak detection systems. +a,c

6.0 FATIGUE CRACK GROWTH ANALYSIS

To determine the sensitivity of the primary coolant system to the presence of small cracks, a fatigue crack growth analysis was carried out for the [ ]+ region of a typical system. This region was selected +a,c, because it is typically one of the highest stressed cross sections, and crack growth calculated here will be conservative for application to the entire primary coolant system.

A finite element stress analysis was carried out for the [ ]+ of a plant typical in geometry and operational characteristics to +a,c, any Westinghouse PWR System. [

]+ All +a,c, normal, upset and test conditions were considered, and circumferentially oriented surface flaws were postulated in the region, assuming the flaw was located in three different locations, as shown in Figure 5. Specifically, these were:

Cross Section A: [ ]+ +a,c,e
Cross Section B:
Cross Section C:

Fatigue crack growth rate laws were used [

]+ The law for stainless steel was +a,c,e derived from Reference 11, with a very conservative correction for R ratio, the ratio of minimum to maximum stress during a transient.

$$\frac{da}{dn} = (5.4 \times 10^{-12})\ K_{eff}^{4.48} \text{ inches/cycle}$$

where $K_{eff} = K_{max}\ (1-R)^{0.5}$

$R = K_{min}/K_{max}$

+a,c,e

The calculated fatigue crack growth for semi-elliptic surface flaws of circumferential orientation and various depths is summarized in Table 2, and shows that the crack growth is very small, regardless [

]+ +a,c,e

7.0 CONCLUSIONS

This report has established the applicability of the generic Westinghouse evaluations which justify the elimination of RCS primary loop pipe breaks for the Callaway and Wolf Creek plants as follows:

a. The loads, material properties, transients and geometry relative to the Callaway and Wolf Creek RCS primary loop are enveloped by the parameters of WCAP 9570.

b. The critical crack length at the worst location in the RCS primary loop is [ ]+ This is significantly greater than the +a,c
[ ]+ inches stable crack used as a basis for calculating leak rates +a,c
in WCAP 9570.

c. The leakage through a [ ]+ crack in the RCS primary loop is [ ]+ +a,c
based on WCAP 9570. The Callaway and Wolf Creek plants have a RCS pressure boundary leak detection system which is consistent with the requirements of Regulatory Guide 1.45 and can detect leakage of 1 gpm in one hour. Thus, there is a factor of [ ]+ between the calculated +a,c
leak rate and the Callaway and Wolf Creek plants leak detection systems.

d. Fatigue crack growth was determined for postulated flaws and was found to be extremely small over plant life and, therefore, is considered insignificant.

Based on the above, it is concluded that RCS primary loop pipe breaks should not be considered in the structural design basis of the Callaway and Wolf Creek plants.

8.0 REFERENCES

1. WCAP 8082 P-A, "Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop," Class 2, January 1975.

2. Letter from Westinghouse (E. P. Rahe) to NRC (R. H. Vollmer) dated May 11, 1983.

3. WCAP 9570, Rev. 2, "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-Wall Crack," Class 3, June 1981.

4. Letter from NRC (H. R. Denton) to AIF (M. Edelman) dated May 2, 1983.

5. WCAP 9283, "The Integrity of Primary Piping Systems of Westinghouse Nuclear Power Plants During Postulated Seismic Events," Class 2, March, 1978.

6. Letter from ACRS (J. J. Ray) to NRC (W. J. Dircks) dated June 14, 1983.

7. Letter from Westinghouse (E. P. Rahe) to NRC (W. V. Johnston) dated April 25, 1983.

8. Letter from Westinghouse (E. P. Rahe) to NRC (W. V. Johnston) dated July 25, 1983.

9. Kanninen, M. F., et al., "Mechanical Fracture Predictions for Sensitized Stainless Steel Piping with Circumferential Cracks" EPRI NP-192, September 1976.

10. Bush, A. J., Stouffer, R. B., "Fracture Toughness of Cast 316SS Piping Material Heat No. 156576 at 600°F", W R&D Memo No. 83-5P6EVMTL-M1, Class 2, March 7, 1983.

11. Bamford, W. H., "Fatigue Crack Growth of Stainless Steel Piping in a Pressurized Water Reactor Environment" Trans. ASME Journal of Pressure Vessel Technology Vol. 101, Feb. 1979.

+a,c,e

APPENDIX A

+a,c

TABLE 1

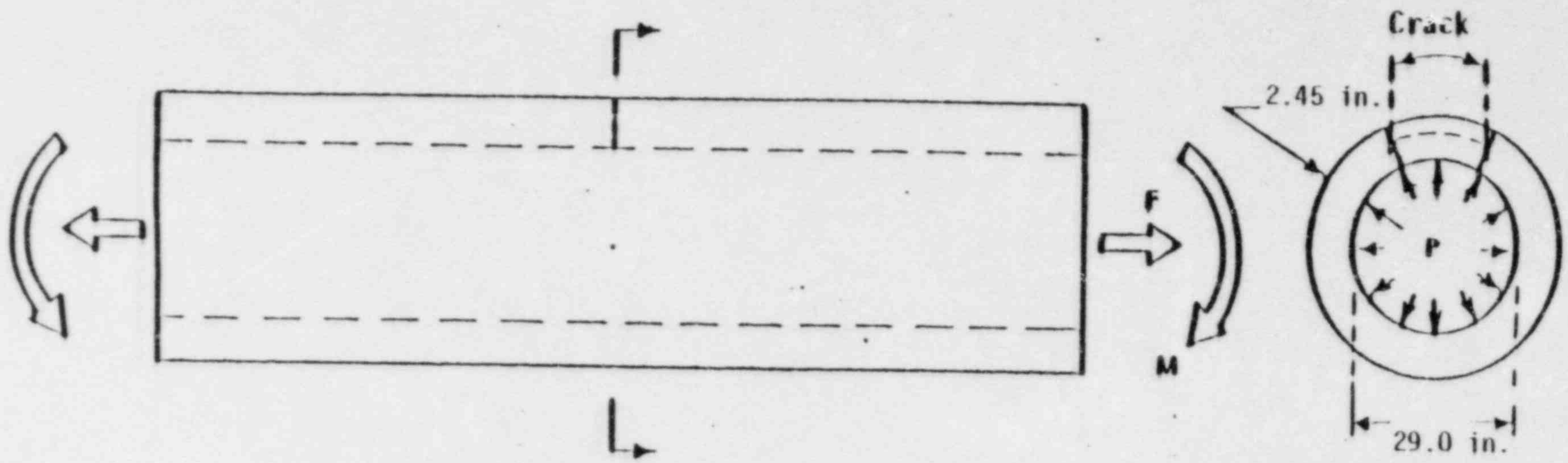
PRIMARY LOOP DATA FOR CALLAWAY AND WOLF CREEK PLANTS

+a,c,e

TABLE 2

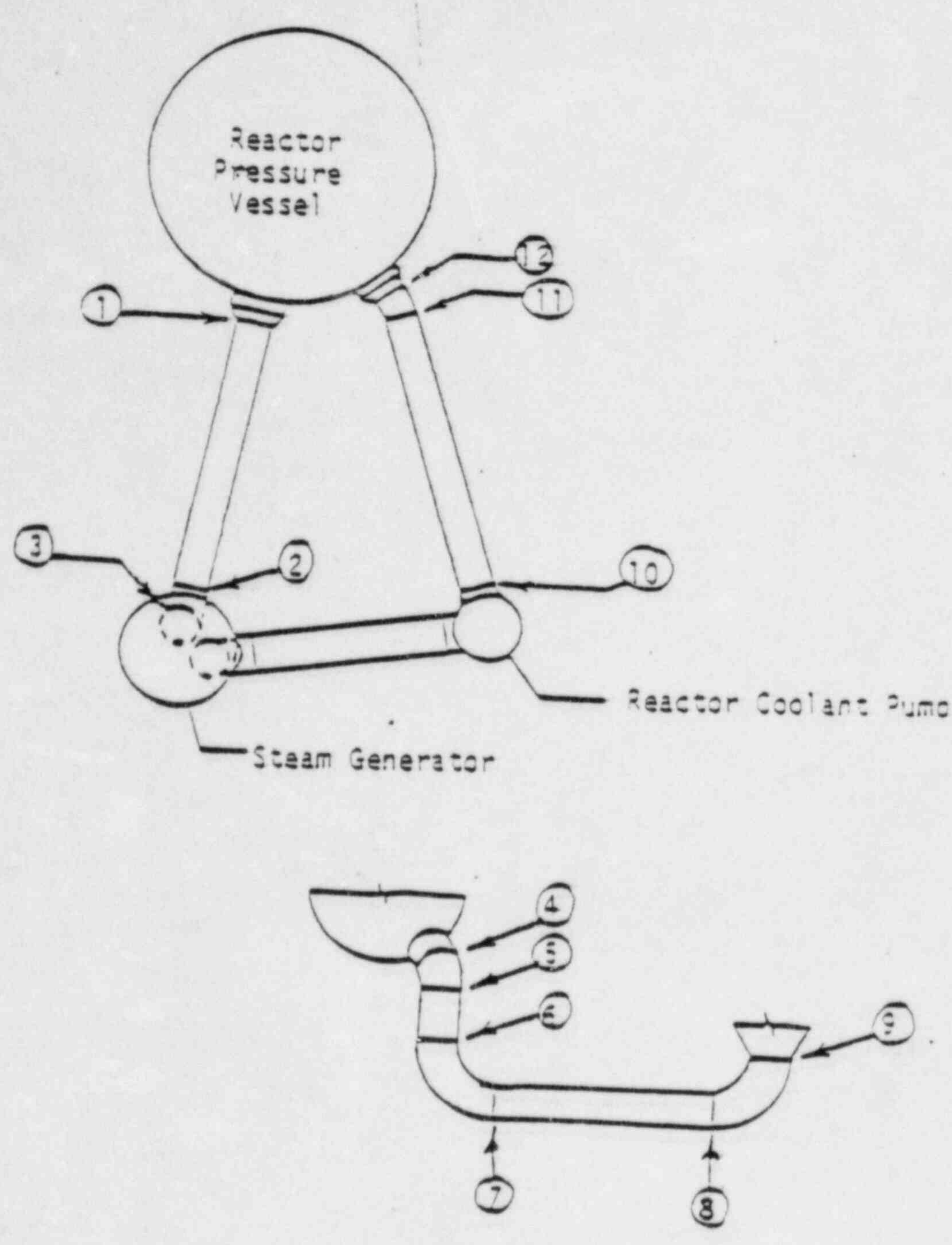
FATIGUE CRACK GROWTH AT [ ]+ (40 YEARS) +a,c,e

Initial Flaw (In)	FINAL FLAW (IN)			
	[ ]+	[ ]+	[ ]+	+a,c,
0.292	0.31097	0.30107	0.30698	
0.300	0.31949	0.30953	0.31626	
0.375	0.39940	0.38948	0.40763	
0.425	0.45271	0.44350	0.47421	



[ ]+a,c,e

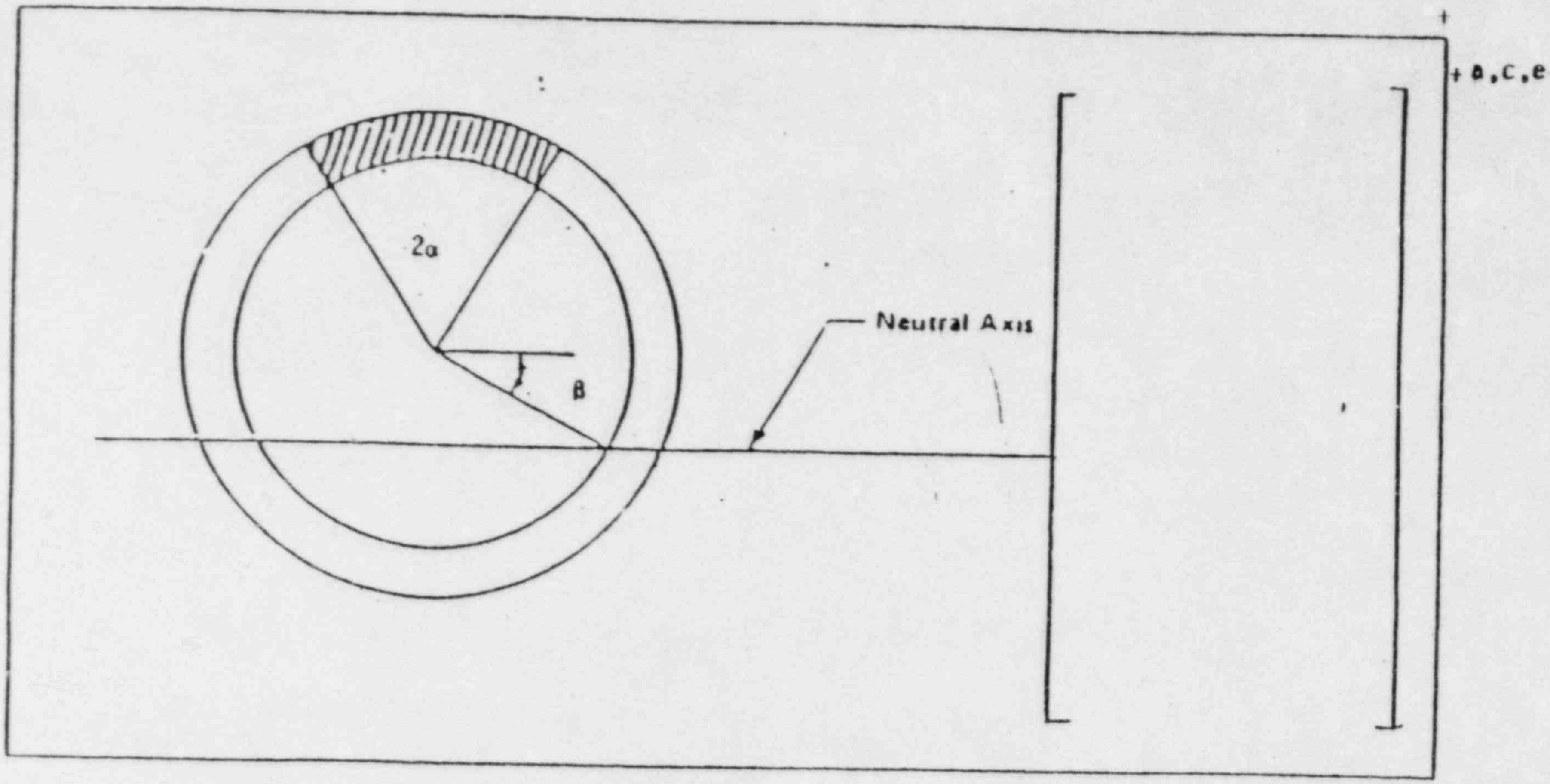
FIGURE 1 REACTOR COOLANT PIPE



a,c,e

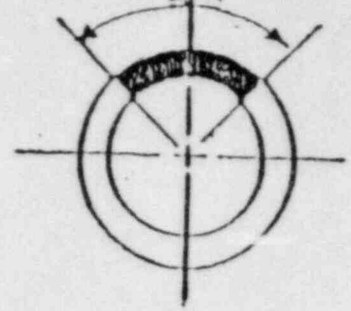
FIGURE 2 SCHEMATIC DIAGRAM OF PRIMARY LOOP SHOWING WELD LOCATIONS

Figure 3 [ ]$^{+a,c,e}$ Stress Distribution



PREDICTED[ ]$^{+a,c,e}$

]$^{+a,c,e}$



FLAW GEOMETRY

CRACK LENGTH , INCHES

FIGURE 4 CRITICAL FLAW SIZE PREDICTION

+a,c,e

FIGURE 5 CROSS SECTION OF [ ]

+a,c

Crack Growth Rate da/dN Micro inches/cycle

Stress Intensity Factor Range ΔK_I (ksi$\sqrt{in}$)

FIGURE 6 REFERENCE FATIGUE CRACK GROWTH CURVES FOR [ ] +a,c,e

+a,c,e

FIGURE 7 REFERENCE FATIGUE CRACK GROWTH LAW FOR INCONEL 600 IN A WATER ENVIRONMENT AT 600F.

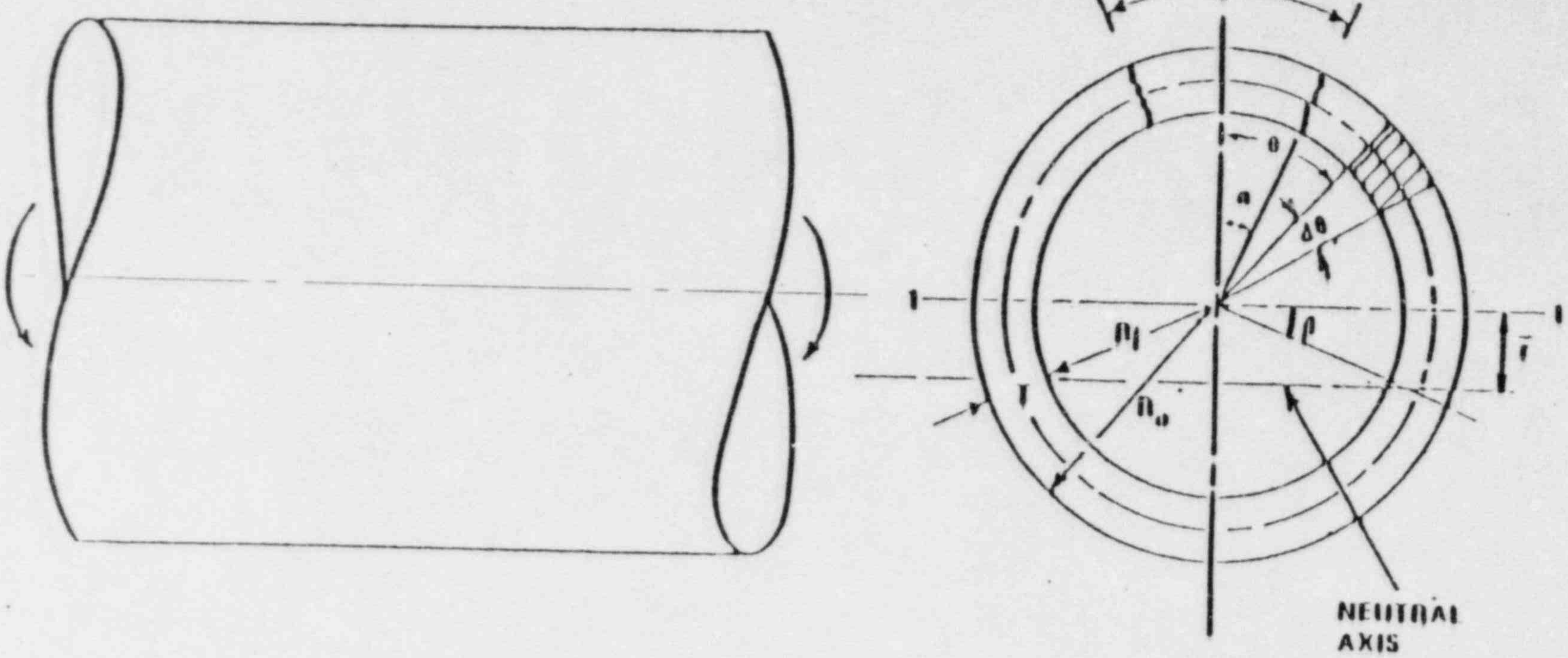


FIGURE A-1 P!PE WITH A THROUGH-WALL CRACK IN BENDING