Mailing Address
Alabama Power Company
600 North 18th Street
Post Office Box 2641
Telephone 205 783-6090

R. P. McDonald
Senior Vice President-
XXXXXXXXXXXX
Flintridge Building

Alabama Power
the southern electric system

June 18, 1984

Docket Nos. 50-364

Director, Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. S. A. Varga

Joseph M. Farley Nuclear Plant - Unit 2
Response to NRC Questions - Reactor Vessel Surveillance
Capsule Report and Associated Technical Specification Change Request

Gentlemen:

Pursuant to NRC Staff request for information of May 3, 1984 and NRC letter dated May 14, 1984, Alabama Power Company provides the enclosed response.

This response serves to clarify both the Reactor Vessel Surveillance Capsule Report, which was transmitted to the NRC by Alabama Power Company letter dated November 10, 1983, and the associated technical specification change request dated February 10, 1984.

If there are any questions, please advise.

Yours very truly,

R. P. McDonald

RPM/CJS:ddr-D9
Attachment
cc: Mr. L. B. Long
Mr. J. P. O'Reilly
Mr. E. A. Reeves
Mr. W. H. Bradford

A001
1/1

Enclosure

NRC Staff Request for Information
Heatup/Cooldown Curves
Joseph M. Farley Nuclear Plant Unit 2

1) NRC Request

Provide the nickel composition for all plate materials in the reactor vessel beltline.

Alabama Power Company Response

The nickel composition of the plate material in the reactor vessel beltline is tabulated below:

Component	Heat No.	Plate No.	Ni (wt %)
Inter Shell	C6309-2	B7203-1	.60
Inter Shell	C7466-1	B7212-1	.60
Lower Shell	C6888-2	B7210-1	.56
Lower Shell	C6293-1	B7210-2	.57

2) NRC Request

Provide pressure temperature limit curves that comply with the explicit closure flange material temperature requirements of the amended (May 27, 1983) Appendix G, 10CFR50, or provide the information described in Item 3.

Alabama Power Company Response

The pressure temperature limit curves submitted to the NRC by letter dated February 10, 1984 do not reflect the 120°F (normal operation) and 90°F (hydrostatic testing) requirements of Appendix G to 10CFR50 for the flange area since Appendix G also allows a lower temperature to be used, if properly justified. Alabama Power Company's response to question 3, below, provides the required justification.

3) NRC Request

Provide the analysis that shows that the closure flange region is less limiting than the beltline region.

Alabama Power Company Response

Westinghouse has performed an analysis to demonstrate that the closure flange region is less limiting than the beltline region. As stated in 10CFR50 Appendix G, "the margins of safety for those regions (i.e., closure flange regions) when they are controlling are equivalent to those required for the beltline when it is controlling". The Westinghouse analysis was originally performed for the Comanche Peak Nuclear Plant - Units 1 and 2. Westinghouse has evaluated the Comanche Peak analysis (Attachment 2) and has determined that its methodology and results are conservatively bounding for the Farley Nuclear Plant - Unit 2. Westinghouse has determined that the Farley closure flange region is less limiting than the beltline region as demonstrated in response to NRC questions 3a, 3b, 3c, 3d and 3e.

3a) NRC Request

Include as a minimum the following information:

a) A description of the finite element analysis used to determine the stresses within the closure flange region.

Alabama Power Company Response

A two dimensional finite element model of a typical 4 loop reactor vessel closure head flange and vessel flange geometry was used in the analysis. The WECAN finite element program was used to develop the model. A discussion of why the 4 loop model can be applied to the 3 loop Farley Nuclear Plant - Unit 2 is provided in response to NRC question 3b. The finite element model was used to obtain temperature and stress gradients induced by the heatup and cooldown transients. Separate iterations of the finite element model were performed to determine the bolt-up, pressure and thermal stresses. Figure 1 of Attachment 2 shows the cross sections analyzed for the closure flange regions. Figure 2 of Attachment 2 shows the mechanical boundary conditions and Figure 3 of Attachment 2 illustrates the thermal boundary conditions of the finite element model. A summary description of how the model was developed and the rationale for the mechanical and thermal boundary conditions is provided in Sections 2.0, 2.1 and 2.2 of Attachment 2.

3b) NRC Request

Include as a minimum the following information:

b) Indicate the peak bolt-up, pressure and thermal stresses determined by the finite element analysis at the inside and outside surface locations of the flange to head and flange to shell junctions.

Alabama Power Company Response

The peak bolt-up, pressure and thermal stresses determined by the finite element analysis are provided in Tables 1 through 4 of Attachment 2. Stress values are provided in these Tables for both the inside and outside surface locations of the flange to head and flange to vessel cross-sections. Included are both Heat-up and Cooldown transients and associated longitudinal and circumferential stress values. A summary of the methodology for developing the stress values is provided in Section 3.0 of Attachment 2. It is noted that cross-sections 1 and 2 are for the flange to head junctions and cross-section 3 is for the flange to vessel junction as shown in Figure 1 of Attachment 2. The tabulated values are considered representative for the Farley Nuclear Plant - Unit 2. Stress values are lower for Farley Unit 2 for the following reason:

In Reactor Vessel Design, the vessel wall, head and flange dimensions are sized in such a manner that the total mechanical stresses due to pressure and bolt-up are virtually identical. Thermal stresses are a function of the vessel wall thickness. The Comanche Peak (4 loop) vessel, which was used in the analysis (Attachment 2), has a thicker reactor vessel wall than Farley - Unit 2. The Comanche Peak Vessel exhibits higher thermal inertia and thereby has higher thermal stresses than the thinner walled 3 loop reactor vessels like Farley - Unit 2.

3c) NRC Request

Include as a minimum the following information:

c) Indicate how the bolt-up, pressure and thermal stresses were combined to determine the maximum applied stress intensity factors.

Alabama Power Company Response

A safety factor of 2.0 was applied to the stress intensity factor for primary stresses (bolt-up and pressure stresses) as required by ASME Code Section III, Appendix G.

The formula for the combination of primary and secondary (thermal) stress intensity factors (K_1) is as follows:

(K_1) total = 2 (K_1) primary + (K_1) secondary

In order to conservatively neglect the reduction of either the primary or secondary stress intensity factors by a corresponding negative stress intensity factor value, negative stress intensity factors were assumed to be equal to zero.

3d) NRC Request

Include as a minimum the following information:

d) Indicate the flaw geometry used to calculate the maximum applied stress intensity factors.

Alabama Power Company Response

The flaw assumed in the analysis is a 0.625 inch deep surface flaw with an aspect ratio of 1:6. The long dimension of the flaw is assumed to be in the longitudinal direction for the calculation of stress intensity factors for longitudinal flaws (both inside and outside surfaces) and to be in the circumferential direction for the calculation of stress intensity factors for circumferential flaws (both inside and outside surfaces). The methods of ASME Code Section IX, Appendix A, 1983 were used to generate the fracture analysis results.

3e) NRC Request

Include as a minimum the following information:

e) Indicate the maximum applied stress intensity factors for the flange to head and flange to shell junctions.

Alabama Power Company Response

The maximum stress intensity factors, for the three closure flange cross-sectional areas analyzed, are tabulated in Tables 5 through 12 of Attachment 2. The tabular maximum stress intensity factors for the flange to head and flange to shell junctions are 52.79 ksi $\sqrt{in}$ and 64.74 ksi $\sqrt{in}$ respectively. The tabulated values are considered to be representative for Farley Unit 2. The stress intensity factors are lower for Farley Unit 2 for basically the same reasons stated in response to question 3c above. A dimensional analysis was performed by Westinghouse to verify that the Comanche Peak analysis is conservative for Farley - Unit 2.

Table 10 of Attachment 2 indicates that the maximum total stress intensity factor (KI) of 64.74 ksi $\sqrt{in}$ occurred for a hypothetical outside surface circumferential flaw, at the flange to shell juncture during a cooldown transient. The thermal stresses at the Farley Unit 2 (3 loop) flange to shell juncture can be approximated by comparing thermal stresses of two cylinders; one cylinder with a thickness of 9.125 inches (3 loop) and the other cylinder with a thickness of 10.75 inches (4 loop). From Figures A.3-5 and A.3-6 of "Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components (Pressurized, Water Cooled Systems)," U.S. Department of Commerce, December 1, 1958 and February 27, 1959 Addenda, it can be shown that the thermal stresses for the thinner vessel are approximately 75 percent of the thicker vessel. For Farley Unit 2, the resulting maximum total stress intensity factor for primary and secondary stresses is 61.63 ksi $\sqrt{in}$ for the same hypothetical outside surface circumferential flaw, at the flange to shell juncture during a cooldown transient. This maximum stress intensity factor for Farley Unit 2 is considered to be relatively small.

3f) <u>NRC Request</u>

Include as a minimum the following information:

f) Indicate the nondestructive examination methods that will be used during inservice examination to determine that the critical flaw size, which was used in determining the maximum applied stress intensity factors, is not within the flange to head and flange to shell junctions.

<u>Alabama Power Company Response</u>

Nondestructive examinations currently used for inservice inspection of the reactor flange-to-vessel and the flange-to-head welds are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, Division 1 - Subsection IWB, "Requirements for Class 1 Components of Light-Water Cooled Power Plants" 1974 Edition, Summer 1975 Addenda and APCo's, "Augmented Reactor Vessel Examination Program" transmitted to the NRC by letter dated October 26, 1983. Table IWB-2600 requires volumetric examination of flange-to-shell welds and flange-to-dome welds. The 1974 Edition, Summer 1975 Addenda of Section XI specifies the boundaries for volumetric examination to include the weld and adjacent base material for a distance equal to one-half the weld thickness on both sides of the weld. The areas to be inspected are graphically represented in the Attached Figures 1 and 2.

Volumetric coverage of the reactor vessel flange-to-upper shell weld and specified adjacent base material is accomplished by two ultrasonic scan routines. Coverage from the flange side of the weld involves use of angled longitudinal waves from the flange seal surface. Beam angles are selected based on their ability to provide coverage of the weld and specified adjacent base material to the extent practical and provide near normal incidence to the plane of the weld. Refracted beam angles in the range 0° to 16° are typically used for these examinations. Examinations from the shell side of the weld involve 0°, 45°, and 60° refracted angle beam coverage from the vessel inside diameter surface. Angle beam scanning is performed in two directions parallel to the weld and perpendicular to the weld from the shell side. Access for the shell side examinations is limited to the Ten Year ISI outage when the core barrel is removed from the reactor vessel.

Volumetric examination of the reactor flange-to-head weld and specified adjacent base material is accomplished by 0°, 45° and 60° refracted angle coverage from the head outside surface. Angle beam scanning is performed in two directions parallel to the weld and perpendicular to the weld from the dome side.

It is the judgement of Alabama Power Company that the inservice examination methodology and techniques will detect critical flaws for the areas examined.

3g) <u>NRC Request</u>

Include as a minimum the following information:

g) Indicate whether the nondestructive examination methods identified in (f) have been evaluated to demonstrate that the examination methods are capable of locating and sizing flaws of the geometry used for calculating the maximum applied stress intensity factors. Indicate the results of the evaluation.

Alabama Power Company Response

Flaws assumed for this analysis were 0.625 inch deep planar surface flaws with 1:6 aspect ratios. The flaw may be oriented circumferentially or axially with respect to the vessel or head and may lie on the OD or ID surface.

The fact that the postulated flaws are surface related is significant from a detection probability point of view. Incipient cracks starting at right angles to a given surface (OD or ID) provide favorable conditions for detection via ASME Code specified 45° shear wave ultrasonic examinations from the opposite surface. Circumferential flaws are oriented favorably for detection during axial scanning. Axial flaws are oriented favorably for detection during circumferential scans. Circumferentially oriented flaws in the vessel flange weld region also provide favorable conditions for detection during ultrasonic examinations from the flange seal surface.

It is noted that the maximum stress intensity factor occurs for a postulated outside surface circumferential flaw at the flange to vessel juncture during a cooldown transient. As mentioned above, in response to NRC question 3f, the volumetric examinations of the shell side of the flange-to-vessel juncture are performed from the inside surface, thereby enhancing the probability of flaw detection. For the flange to head juncture the maximum stress intensity factor occurs for a postulated inside surface longitudinal flaw during a cooldown transient. The volumetric examinations of the reactor closure head flange is accomplished from the outside surface, thereby enhancing the probability of flaw detection. Beam angles selected for these particular scans provide near normal incidence to the anticipated flaw plane thereby further enhancing the probability of detection. Application of near surface examination methods in the form of full mode 45° or shallow angle techniques significantly increases the probability of detecting flaws at the examination surface, i.e., the vessel inside and the head outside.

While the qualitative assessment indicates that detection probabilities are reasonably good for flaws postulated in this analysis, certain unknown factors such as clad effects, defect roughness, orientation, and transparency due to high compressive stresses influence the ability to detect and ultimately provide a realistic estimate of the flaw size with current techniques. Defect sizing by ultrasonic methods has been the subject of several recent studies. To date, no single method has been identified which consistently provides precise sizing data. Typically several different methods must be applied and the most conservative results used in any analysis that might be necessary.

No quantitative information concerning detection and sizing capabilities of the techniques currently applied during examinations of closure flange junctions has been developed based upon qualification demonstrations, nor are such demonstrations specifically required by existing codes and standards. However, the above features of the examinations may be considered to establish that flaws of the type postulated in this analysis which fall within the volumes subject to examination are likely to be detected.

The state-of-the-art of reactor vessel examination has improved over the past several years. Enhanced near-surface detection capabilities and tip-diffraction sizing methods are examples. Continued emphasis on NDE technique development promises to provide further improvements and more quantitative data concerning detection and sizing accuracies.

4) NRC Request

For each capsule in Table 4.4-5 of the Farley 2 Technical Specifications, provide the predicted neutron fluence (E>1MEV) to be received by the capsule at the time of its withdrawal.

Alabama Power Company Response

The predicted neutron fluence (E>1MEV) to be received by each capsule at the scheduled time of withdrawal, submitted in the February 10, 1984 Proposed Technical Specification Change, are listed below:

Capsule	Lead Factor	Removal Time [a]	Estimated Fluence $n/cm^2 \times 10^{19}$
U	3.12	Removed (1.1)	.56
W	2.70	4	2.18
X	3.12	6	3.78[b]
Z	2.70	12	6.54[c]
V	3.12	18	11.34
Y	2.70	Standby	-

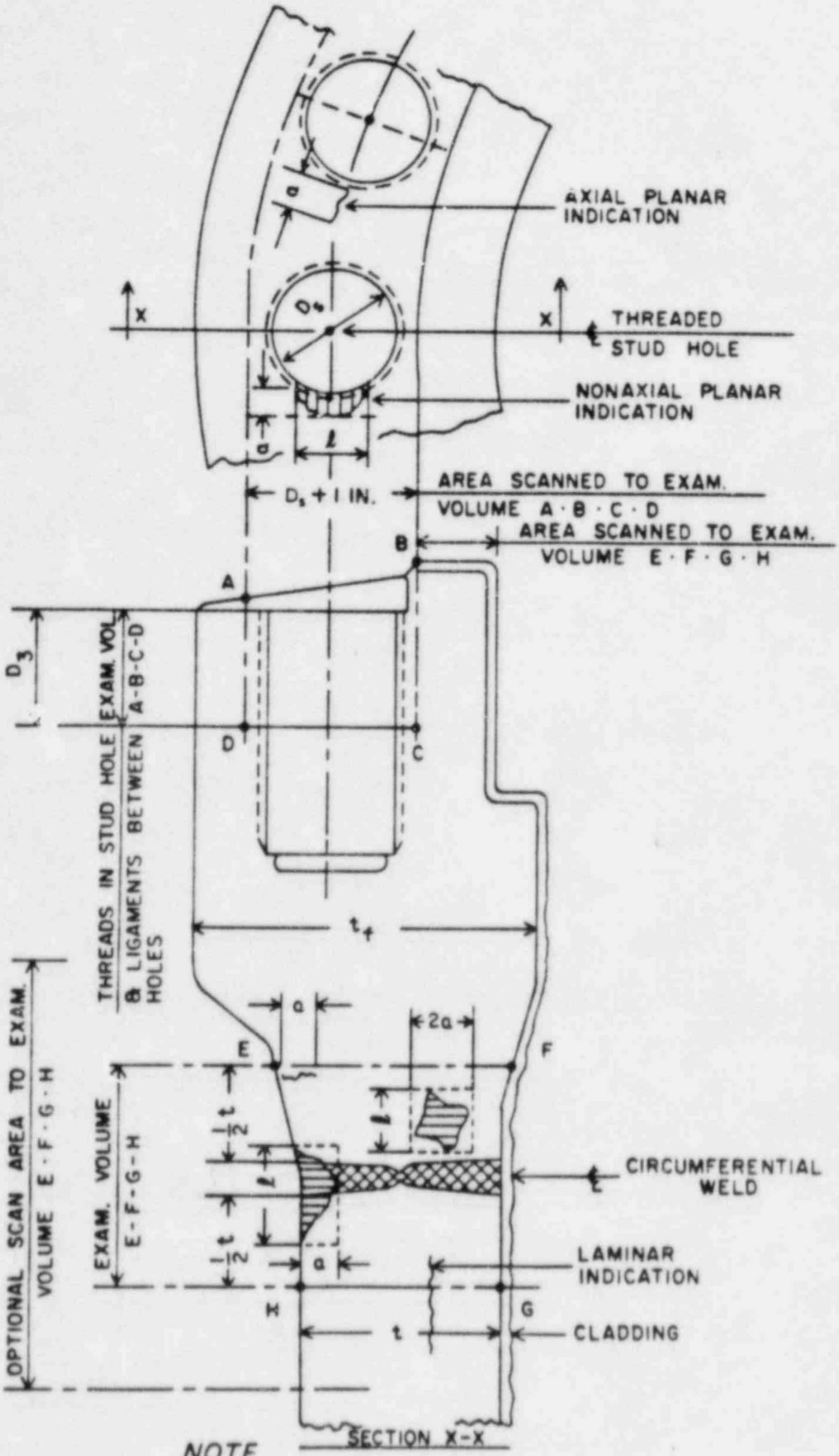
[a] Effective full power years from plant startup
[b] Approximates vessel end of life 1/4 thickness wall location fluence
[c] Approximates vessel end of life inner wall location fluence

5) <u>NRC Request (Per telephone conversation)</u>

Provide a better quality copy of the proposed heat-up/cooldown curves.

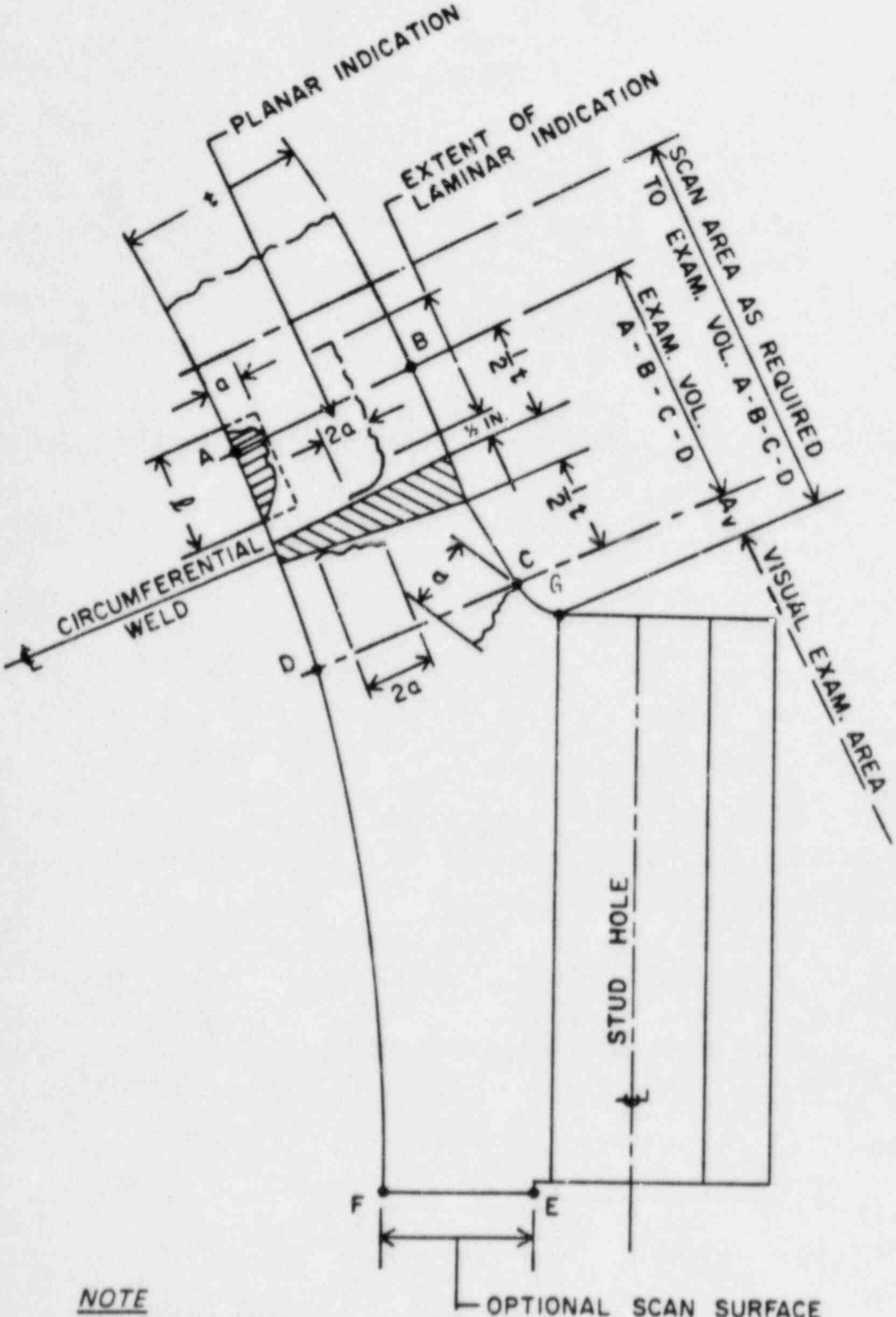
<u>Alabama Power Company Response</u>

Attachment 3 contains the requested curves.



<u>NOTE</u>
ALL FLAWS ARE EXAGGERATED IN SIZE AND SCALE

FIGURE 1 SHELL-TO-FLANGE WELD JOINT



NOTE
ALL FLAWS ARE EXAGGERATED
IN SIZE AND SCALE

FIGURE 2 HEAD-TO-FLANGE WELD JOINT