Northern States Power Company
Prairie Island Nuclear Generating Plant
1717 Wakonade Dr. East
Welch, Minnesota 55089

November 26, 1996

10 CFR Part 50
Section 50.55(a)

U S Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Additional Information Related to the Request for Authorization
to Utilize ASME Boiler & Pressure Vessel Code Case N-521

The purpose of this letter is to provide additional information related to our November 1, 1995 request for NRC authorization to utilize ASME Boiler and Pressure Vessel Code Case N-521, "Alternative Rules for Deferral of Inspections of Nozzle-to-Vessel Welds, Inside Radius Sections, and Nozzle-to-Safe End Welds of a Pressurized Water Reactor (PWR) Vessel, Section XI, Division 1", during the Prairie Island Third Ten Year Inservice Inspection Interval for both Units 1 and 2. This letter also supersedes the November 1, 1995 letter and the two related letters of July 25, 1996 and August 9, 1996.

ASME Code Section XI requires that at least 25% but not more than 50% of the Nozzle-to-Vessel and Inside Radius Sections be examined in the first period of the ten year interval. However, Code Case N-521 provides for deferral of the inspection of nozzle-to-vessel welds, inside radius sections, and nozzle-to-safe end welds of a pressurized water reactor vessel to the end of the inspection interval provided specified conditions are satisfied, rather than dividing the inspections between the first and third periods of the inspection interval. This request is submitted pursuant to 10 CFR Part 50, Section 50.55a(a)(3) and Section 50.55a, Footnote 6. This request is for deferral of inspections not for deletion. We are requesting this deferral pursuant to 50.55a(a)(3)(ii):

A047

> Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

030070

Code Case N-521 was approved by the ASME Code on August 9, 1993. However, the Code Case has not yet been endorsed in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability ASME Section XI Division 1". Until the Code Case is generically endorsed by the Regulatory Guide, specific NRC authorization is required before it can be used.

Both Units of Prairie Island are currently in the 3rd 10-year ISI Interval. The interval began December 16, 1993 for Unit 1 and December 21, 1994 for Unit 2. Relief is sought for inspections of the primary system outlet nozzles during this interval. The code case addresses: nozzle-to-safe end welds, nozzle-to-vessel welds, and the inner radius sections of the nozzles. However, our need for relief is not the same for each of these groups.

1. Nozzle-to-safe end welds

   Although Code Case N-521 addresses these welds, we do not need schedular relief from the code requirements; these inspections have already been performed in the 1st period for Unit 1 and will be performed in the 1st period for Unit 2 in the refueling outage scheduled to start in January 1997. Therefore, whether or not authorization is granted to use the code case for the safe end welds will have no effect on the inspection schedule. Thus there will be no further discussion of the safe end welds.

2. Nozzle-to-vessel welds

   The nozzle-to-vessel welds have been inspected recently (see Table 1). They were inspected to the 1980 Edition, Winter 1981 Addenda of ASME Section XI requirements rather than the 1989 Edition, No Addenda (the edition of the current 10-year Interval); schedular relief to the 3rd period of the interval will alleviate the need to repeat the inspections within a short period of time. Details are discussed below. The resulting interval between inspections will be from the 3rd period of the 2nd interval to the 3rd period of the 3rd interval.

3. Inner radius sections

   The inner radii have not been inspected since 1985 on either unit (see Table 1), therefore direct application of deferment per the code case is sought. Details are discussed below.

TABLE 1

UNIT 1				
Item Inspected	Outlet Loop A		Outlet Loop B	
	NV, Item # = ISI-50, W-7	IR, Item # = ISI-50, W-7 IR	NV, Item # = ISI-50, W-10	IR, Item # = ISI-50, W-10 IR
Last Inspected	1994	1985	1994	1985
Code Edition/Add	80/W81	80/W81	80/W81	80/W81
Inspection Results	NAD	NAD	NAD	NAD

UNIT 2				
Item Inspected	Outlet Loop A		Outlet Loop B	
	NV, Item # = 2-ISI-40, N-7	IR, Item # = 2-ISI-40, N-7 IR	NV, Item # = 2-ISI-40, N-10	IR, Item #= 2-ISI-40, N-10 IR
Last Inspected	1993	1985	1993	1985
Code Edition/Add	80/W81	80/W81	80/W81	80/W81
Inspection Results	NAD	NAD	NAD	NAD

OUTLET NOZZLES NOZZLE-TO-VESSEL WELDS

During the Prairie Island Unit 2, 1997 refueling outage, which is the last refueling outage of the first inspection period, the two outlet nozzle nozzle-to-vessel welds would need to be scheduled for examination in order to meet the 25 percent minimum requirement of Section XI under the present ten-year ISI program. To do these inspections equipment is required to be placed in the reactor vessel. For Unit 1, the last refueling outage of the first inspection period has been completed. To do these inspections per the code schedule, the unit would need to be shutdown between refueling outages in order to place the inspection equipment in the reactor vessel.

We believe that the significant additional cost for each unit to perform the examinations of the reactor vessel outlet nozzle nozzle-to-vessel welds, in separate outages from the other examinations, is not a practical expenditure of resources in light of the following points:

1. All of the nozzle-to-vessel welds were last examined in 1994 (Unit 2, 1993), per ASME Section XI, 1980 Edition, Winter 1981 Addenda, with no observed indications. We consider the results of these past examinations to be representative of the present reactor vessel nozzle conditions at Prairie Island;

2. Although the scheduled welds are to be examined with automated equipment, some personnel radiation exposure will occur. As part of our overall ALARA program, any amount of personnel radiation exposure that can be saved is important. In addition, radioactive waste will be reduced;

3. It makes good sense to combine all reactor vessel examinations into one automated effort from both ALARA and cost considerations. Using the same vendor, equipment, personnel, and procedures to examine all the welds at the same time will enhance the quality of the examination results and produce a much smaller possibility of error.

4. It would provide increased safety to move the equipment into and out of the vessel only once in the interval.

Code Case N-521 allows deferral of the inspections of nozzle-to-vessel welds of a PWR vessel to the end of the inspection interval if the following conditions are met:

a) No inservice repairs or replacements by welding have ever been performed on any of the Nozzle-to-Vessel Welds, Inside Radius Sections, and Nozzle-to-Safe End Welds.
b) None of the Nozzle-to-Vessel Welds, Inside Radius Sections, and Nozzle-to-Safe End Welds contains identified flaws or relevant conditions that currently require successive inspections in accordance with IWB-2420-(b).
c) The unit is not in the first inspection interval.

Both Prairie Island units meet the provisions outlined in the Code Case. In addition, these welds were inspected twice in the second interval, once in the first period and once in the third period. Thus, deferring the third interval inspection from the first period to the third keeps with the Code philosophy for detecting age-related degradation during a specified interval of time. We therefore request authorization to use the Code Case to defer the first period inspections of the nozzle-to-vessel welds to the 3rd period of the inspection interval.

Justification for the conclusion that full compliance with the specified requirements would not result in a compensating increase in the level of quality and safety:

The reactor vessel inspections done in 1993 and 1994 were performed per the 1980 Edition, Winter 1981 Addenda as augmented by Regulatory Guide 1.150 Revision 1. To have met the requirements of the third ten year interval the inspections were required to be performed to the 1989 Edition, No Addenda of the Code as augmented by Regulatory Guide 1.150, Revision 1. Regulatory Guide 1.150 contains the recording and reporting requirements. There is no difference in extent of examination between the 1980 Edition, Winter 1981 Addenda and the 1989 Edition, No Addenda of the Code. There are differences between the two codes regarding allowable indications, but since the 1993 and 1994 inspections did not detect any recordable indications on the outlet nozzle to vessel welds, there is no need for reevaluation of any indications to the newer Code.

Another difference between the two codes is in the requirements for examiners certification. Examiner certification requirements in Appendix VII to Section V is new to the 1989 Edition. However, this edition allows the examiner to perform inspections under the examiner's current certifications until the certifications expire; at the time of expiration the examiner then requalifies under the requirements contained in Appendix VII to Section V of the 1989 Edition, No Addenda. Therefore, even if the examinations had been performed under the requirements of the 1989 Edition, No Addenda, the examiners' certifications could have been unchanged from what they were for an examination under the 1980 Edition, Winter 1981 Addenda.

Therefore, there would have been no significant differences in the 1993 and 1994 nozzle-to-vessel weld examination results had the examinations been performed to the 1989 Edition, No Addenda instead of the 1980 Edition, Winter 1981 Addenda.

The level of quality and safety will not be decreased by the use of Code Case N-521 to defer the outlet nozzle-to-vessel welds to the 3rd period of the 3rd interval since these welds were inspected in the 3rd period of the 2nd interval.

OUTLET NOZZLES INNER RADIUS SECTIONS

During the Prairie Island Unit 2, 1997 refueling outage, which is the last refueling outage of the first inspection period, the two outlet nozzles inner radius sections would need to be scheduled for examination in order to meet the 25 percent minimum requirement of Section XI under the present ten-year ISI program. To do these inspections equipment is required to be placed in the reactor vessel. For Unit 1, the last refueling outage of the first inspection period has been completed. To do these inspections per the code schedule, the unit would need to be shutdown between refueling outages in order to place the inspection equipment in the reactor vessel.

We believe that the significant additional cost for each unit to perform the examinations of the reactor vessel outlet nozzle inner radius sections, in separate outages from the other examinations, is not a practical expenditure of resources in light of the following points:

1. Four of the six inner radius sections (2 inlet, 2 outlet, and 2 safety injection nozzles) were last examined in 1994 (Unit 2, 1993), per ASME Section XI, 1980 Edition, Winter 1981 Addenda, with no observed indications. We consider the results of these past examinations to be representative of the present reactor vessel nozzle conditions at Prairie Island;

2. Although the scheduled outlet nozzles inner radius sections are to be examined with automated equipment, some personnel radiation exposure will occur. As

part of our overall ALARA program, any amount of personnel radiation exposure that can be saved is important. In addition, radioactive waste will be reduced;

3. It makes good sense to combine all reactor vessel examinations into one automated effort from both ALARA and cost considerations. Using the same vendor, equipment, personnel, and procedures to examine all the inner radius sections at the same time will enhance the quality of the examination results and produce a much smaller possibility of error.

4. It would provide increased safety to move the equipment into and out of the vessel only once in the interval.

Code Case N-521 allows deferral of the inspections of inner radius sections of a PWR vessel to the end of the inspection interval if the following conditions are met:

a) No inservice repairs or replacements by welding have ever been performed on any of the Nozzle-to-Vessel Welds, Inside Radius Sections, and Nozzle-to-Safe End Welds.
b) None of the Nozzle-to-Vessel Welds, Inside Radius Sections, and Nozzle-to-Safe End Welds contains identified flaws or relevant conditions that currently require successive inspections in accordance with IWB-2420-(b).
c) The unit is not in the first inspection interval.

Both Prairie Island units meet the provisions outlined in the Code Case. We therefore request authorization to use the Code Case to defer the first period inspections of the inner radius sections to the 3rd inspection period of the interval.

Justification for the conclusion that full compliance with the specified requirements would not result in a compensating increase in the level of quality and safety:

<u>Technical Evaluation</u>

An evaluation has been performed by Westinghouse to appraise the susceptibility of the inner radii of the outlet nozzles to flaw development and growth. A summary of this evaluation and a discussion of the methodology used is attached to this letter. The evaluation concludes that deferring the inspections of these inner radius sections does not decrease the quality or safety.

This evaluation assumes that existing flaws of significance would have been detected in the 1985 examinations, which were performed to the requirements of the 1980 Edition, Winter 1981 Addenda would have been detected. The ultrasonic examination techniques used in 1985 are essentially the same as are currently being used in our ISI program. Flaws large enough to be reportable are easily detectable with these techniques and it is highly improbable that a flaw of such size could have existed and not been detected with the techniques used.

Industry History

The inspection of reactor vessel nozzle-to-shell welds, their nozzle inner radius sections, and associated nozzle-to-safe end welds was originally distributed by Code requirements between the three inspection periods of an interval on the basis of monitoring these areas uniformly during the interval. The cost in terms of radiation exposure due to removing and replacing the core barrel caused the Code to be changed such that only 25 percent, to a maximum of 50 percent of the nozzles were to be inspected in the 1st period and inspections were dropped for the 2nd period. This resulted in the core barrel being pulled only once because the normal arrangement of the two outlet passages in the core barrel allowed direct access to the nozzles. The remaining nozzles were inspected in the 3rd period with the removal of the core barrel required. The industry believes that with the present large population of operating reactors, the examination of nozzles of a given vessel being distributed between an outage in the 1st and an outage in the 3rd period can be eliminated and concentrated in an outage in the 3rd period of the interval. The examinations of all nozzles in all vessels within a given ten-year interval are rather uniformly distributed across the industry. Each reactor is representative of the operating conditions throughout the population of the reactors for a particular nuclear steam supply system design. The number of nozzles inspected within a ten-year inspection interval will be the same, either with all nozzles of a given reactor being inspected during one outage in the 3rd period, or the inspections being distributed between outages in the 1st and 3rd periods. Therefore, we believe that inspecting all of the nozzles of a given reactor at one time will provide the necessary assurance of detecting the presence of an industry wide degrading condition.

In this letter we have made no new Nuclear Regulatory Commission commitments.

Please contact Jack Leveille (612-388-1121, Ext. 4662) if you have any questions related to this letter.

Michael D Wadley

Michael D Wadley
Plant Manager
Prairie Island Nuclear Generating Plant

c: Regional Administrator - Region III, NRC
Senior Resident Inspector, NRC
NRR Project Manager, NRC
J E Silberg

Attachment: Technical Basis for Outlet Nozzle Inner Radius Inspection Exemption

Technical Basis for Outlet Nozzle Inner Radius Inspection Exemption: Prairie Island Units 1 and 2

1.0 INTRODUCTION

The purpose of this note is to provide an assessment of the structural integrity of the Prairie Island Units 1 and 2 reactor vessel outlet nozzle inner radius sections, and the need for their inservice inspection. Two complementary approaches will be used, inspection history, and fracture assessment. Each of these approaches has led to the same conclusion, which is that inservice inspection of the nozzle inner radius does not significantly improve confidence in the structural integrity of the reactor vessel. Extending the inspection interval for the nozzle inner radius beyond 10 years will not affect structural integrity. Even if the interval were extended to 20 years or beyond, the conclusion would not change. Therefore it will be shown that extending the interval between inspections for the Prairie Island outlet nozzles will not pose any technical problems. The geometry of the outlet nozzle for the Prairie Island units is shown in Figure 1.

2.0 INSPECTION HISTORY

The outlet nozzle forgings for both Prairie Island Units 1 and 2 were examined by both UT and MT (magnetic particle testing) prior to the cladding being applied, per the requirements of the material specification. After cladding, the nozzles were required to be liquid penetrant tested to ensure the integrity of the cladding. The inspection after the hydrotest includes UT as well as radiography, for acceptance to the ASME Code, Section XI, and Section III.

For inservice inspection of the outlet nozzle inner radius region, a contact examination is carried out from the inside of the reactor vessel. The details of the previous inspections have been discussed earlier.

Underclad Cracks. The only indications of any type which have been found are small underclad cracks, which exist in all vessels manufactured by Creusot Loire and Rotterdam Dockyard. These small cracks were created by the cladding process, and have a maximum depth of about 0.12 inches. Crack growth evaluations and multiple inspections of these indications in a number of reactor vessels have shown that these flaws do not grow during service and have no impact on the integrity of the vessel.

A survey was conducted of all Westinghouse plant owners to determine if flaws have been found in the inner radius of any of the nozzles in the primary system. The results of over 300 UT examinations which have been completed on these nozzles showed that no indications have been found. The survey included nozzles in pressurizers, and steam generator primary nozzles as well, and confirmed the expected findings.

While this finding is not itself sufficient to prove there is no need for further inspection in this area, it is consistent with the other findings reported here, in that no concerns are evident with flaws in this region at the beginning of service. There are no mechanisms of damage other than fatigue for the outlet nozzle. Stress corrosion cracking has never been ovserved in either stainless or low alloy pressure vessel steel with no oxygen present. Therefore, the only scenarios of concern are for a flaw which was not found in the preservice examination to grow during service or for a flaw to initiate during service and grow. Initiation is assessed quantitatively by the fatigue analysis carried out to certify the design acceptance of the vessel. Crack growth has been assessed as part of the fracture assessment below, in Section 3.3.

3.0 FRACTURE ASSESSMENT

A fracture evaluation was carried out for the Prairie Island outlet nozzle inner radius, and documented in Reference 1, in the form of flaw evaluation charts, which show the largest allowable flaw which would meet the ASME Section XI flaw acceptance requirements of paragraph IWB 3600. Although this report was prepared in 1984, it is based on a detailed series of finite element stress analyses and fracture analysis methods which are consistent with the present state of the art. Furthermore, the requirements of Section XI IWB 3600 have not changed since that time. Therefore the conclusions of references 1 and 2 remain valid today, that is that there is a reasonable tolerance for the presence of flaws in the outlet nozzle inner radius region.

3.1 Allowable Flaw Size

This is seen in the flaw chart of Figure 2, in terms of allowable flaw depth as a percentage of the nozzle corner ligament, which is 17.3 inches for the Prairie Island units. The details of the construction of this flaw chart are described in Appendix A. In Figure 2 we see that the allowable flaw depth is presented as a function of flaw shape (a/ℓ), including the effects of fatigue crack growth. For a continuously long flaw the allowable is equal to the value from the IWB 3500 standards table, 2.5 percent of the wall thickness. For flaws with more realistic shapes, the allowable depth is much greater 0.67 inches for a 6:1 flaw ($a/\ell = 0.167$) and 1.6 inches for a semi circular flaw ($a/\ell = 0.5$).

3.2 Critical Flaw Size

Another view of the tolerance of the outlet nozzle for the presence of flaws can be obtained by calculating the critical flaw size for this region as a function of flaw shape, and these results are presented in Figure 3. The critical depth for a continuous flaw in this region was calculated to be 3.7 inches, or 21.6 percent of the ligament thickness for the dominating normal and upset transient, Loss of Power. For the governing emergency and faulted condition, the Large Steamline Break, the critical depth was 2.2 inches. The critical flaw

depth for other flaw shapes is much larger, as seen in Figure 3. Therefore it may be concluded that the structural integrity of the outlet nozzle will not be affected by the presence of any flaws smaller than those discussed above at the inner radius.

3.3 Fatigue Crack Growth

The fatigue crack growth predicted for a range of flaw sizes and shapes is shown in Table 1. It may be readily seen that the predicted crack growth is very small, even with the assumption that the surface flaw is exposed to the PWR water environment. Since service experience has shown that few (if any) flaws have even been found to break through the cladding, this is a very conservative assessment. It is clear from Table 1 that a range of flaw sizes will not grow beyond the code allowables in the 40 year design life.

4.0 CONCLUSIONS

The assessments discussed here have shown that deferring the outlet nozzle inner radius exams will not decrease the level of quality and safety for Prairie Island Units 1 and 2. Inspections which have been done have not led to the discovery of any indications in the entire service history of Westinghouse plants, or Pressurized Water Reactors regardless of manufacturer or designer. The fracture assessment showed the outlet nozzle inner radius has a large tolerance for flaws. There are no mechanisms for the development of flaws or the significant growth of existing flaws during service, so the risk of failure is not decreased by inservice inspection. The case for extending the inspection interval for these nozzles is so strong that efforts are underway to consider elimination of this inspection requirement from Section XI of the ASME Code.

Technical Basis for Outlet Nozzle Inner Radius Inspection Exemption:
Prairie Island Units 1 and 2

1.0 INTRODUCTION

The purpose of this note is to provide an assessment of the structural integrity of the Prairie Island Units 1 and 2 reactor vessel outlet nozzle inner radius sections, and the need for their inservice inspection. Two complementary approaches will be used, inspection history, and fracture assessment. These approaches lead to the conclusion that inservice inspection of the nozzle inner radius does not significantly improve confidence in the structural integrity of the reactor vessel. Extending the inspection interval for the nozzle inner radius beyond 10 years will not affect structural integrity. Even if the interval were extended to 20 years or beyond, the conclusion would not change. Therefore it will be shown that extending the interval between inspections for the Prairie Island outlet nozzle inner radius sections will not pose any technical problems. The geometry of the outlet nozzle for the Prairie Island units is shown in Figure 1.

2.0 INSPECTION HISTORY

The outlet nozzle forgings for both Prairie Island Units 1 and 2 were examined by both UT and MT (magnetic particle testing) prior to the cladding being applied, per the requirements of the material specification. After cladding, the nozzles were required to be liquid penetrant tested to ensure the integrity of the cladding. The inspection after the hydrotest includes UT as well as radiography, for acceptance to the ASME Code, Section XI, and Section III.

For inservice inspection of the outlet nozzle inner radius region, a contact examination is carried out from the inside of the reactor vessel. The details of the previous inspections are discussed in the body of the letter.

Underclad Cracks.

The only indications of any type which have been found are small underclad cracks. These small cracks were created by the cladding process and were detected by special ultrasonic testing techniques developed for this specific application. (An industry effort was developed following the 1979 discovery of these cracks in another Westinghouse reactor.) Field tests were performed during scheduled ISI outages for both Prairie Island units in 1981. Reflectors detected during these examinations were evaluated per ASME Section XI requirements and found to be acceptable. Crack growth evaluations performed to address these flaws have shown that these flaws do not grow significantly during service and have no impact on the integrity of the vessel. Corroborating these evaluations are the special inspections performed for two reactors to appraise the underclad cracking following 15 years of operation since the initial cracking was

detected. No growth was observed. The author knows of no Westinghouse PWRs worldwide that have experienced problems with underclad cracks.

A survey was conducted of all Westinghouse plant owners to determine if flaws have been found in the inner radius of the nozzles in the primary system. The results of over 300 UT examinations which have been completed on these nozzles showed that no recordable indications have been found. The survey included nozzles in pressurizers and steam generator primary nozzles, and confirmed the expected findings. In addition, this author is unaware of any recordable indications in the nozzle inner radius region of any plant of Westinghouse design worldwide.

While this finding is not itself sufficient to prove there is no need for further inspection in this area, it is consistent with the other findings reported here, in that no concerns are evident with flaws in this region at the beginning of service. There are no mechanisms of damage other than fatigue for the outlet nozzle. Stress corrosion cracking has never been observed in either stainless or low alloy pressure vessel steel with no oxygen present. Therefore, the only scenarios of concern are for a flaw which was not found in the preservice examination to grow during service or for a flaw to initiate during service and grow. Initiation is assessed quantitatively by the fatigue analysis carried out to certify the design acceptance of the vessel. Crack growth has been assessed as part of the fracture assessment below, in Section 3.3.

3.0 FRACTURE ASSESSMENT

A fracture evaluation was carried out for the Prairie Island outlet nozzle inner radius, and documented in Reference 1, in the form of flaw evaluation charts, which show the largest allowable flaw which would meet the ASME Section XI flaw acceptance requirements of paragraph IWB 3600. Although this report was prepared in 1984, it is based on a detailed series of finite element stress analyses and fracture analysis methods which are consistent with the present state of the art. Furthermore, the requirements of Section XI IWB 3600 have not changed since that time. Therefore the conclusions of references 1 and 2 remain valid today, that is that there is a reasonable tolerance for the presence of flaws in the outlet nozzle inner radius region.

3.1 Allowable Flaw Size

This is seen in the flaw chart of Figure 2, in terms of allowable flaw depth as a percentage of the nozzle corner ligament, which is 17.3 inches for the Prairie Island units. The details of the construction of this flaw chart are described in Appendix A. In Figure 2 we see that the allowable flaw depth is presented as a function of flaw shape (a/l), including the effects of fatigue crack growth. For a continuously long flaw the allowable is equal to the value from the IWB 3500 standards table, 2.5 percent of the wall thickness. For flaws with more realistic

shapes, the allowable depth is much greater. The calculated depths are 0.67 inches for a 6:1 flaw (a/l = 0.167) and 1.6 inches for a semi circular flaw (a/l = 0.5).

3.2 Critical Flaw Size

Another view of the tolerance of the outlet nozzle for the presence of flaws can be obtained by calculating the critical flaw size for this region as a function of flaw shape, and these results are presented in Figure 3. The critical depth for a continuous flaw in this region was calculated to be 3.7 inches, or 21.6 percent of the ligament thickness for the dominating normal and upset transient, Loss of Power. For the governing emergency and faulted condition, the Large Steamline Break, the critical depth was 2.2 inches. The critical flaw

depth for other flaw shapes is much larger, as seen in Figure 3. Therefore it may be concluded that the structural integrity of the outlet nozzle will not be affected by the presence of any flaws smaller than 2.2 inches at the inner radius.

3.3 Fatigue Crack Growth

The fatigue crack growth predicted for a range of flaw sizes and shapes is shown in Table 1. It may be readily seen that the predicted crack growth is very small, even with the assumption that the surface flaw is exposed to the PWR water environment. Since service experience has shown that few (if any) flaws have even been found to break through the cladding, this is a very conservative assessment. It is clear from Table 1 that a range of flaw sizes will not significantly grow in 20 years or more.

4.0 CONCLUSIONS

The assessments discussed here have shown that deferring the outlet nozzle inner radius exams will not decrease the level of quality and safety for Prairie Island Units 1 and 2. Inspections which have been done have not led to the discovery of any indications in the entire service history of Westinghouse plants, or Pressurized Water Reactors regardless of manufacturer or designer. The fracture assessment showed the outlet nozzle inner radius has a large tolerance for flaws. There are no mechanisms for the development of flaws or the significant growth of existing flaws during service, so the risk of failure is not decreased by inservice inspection. The case for extending the inspection interval for these nozzles is so strong that efforts are underway to consider elimination of this inspection requirement from Section XI of the ASME Code.

REFERENCES

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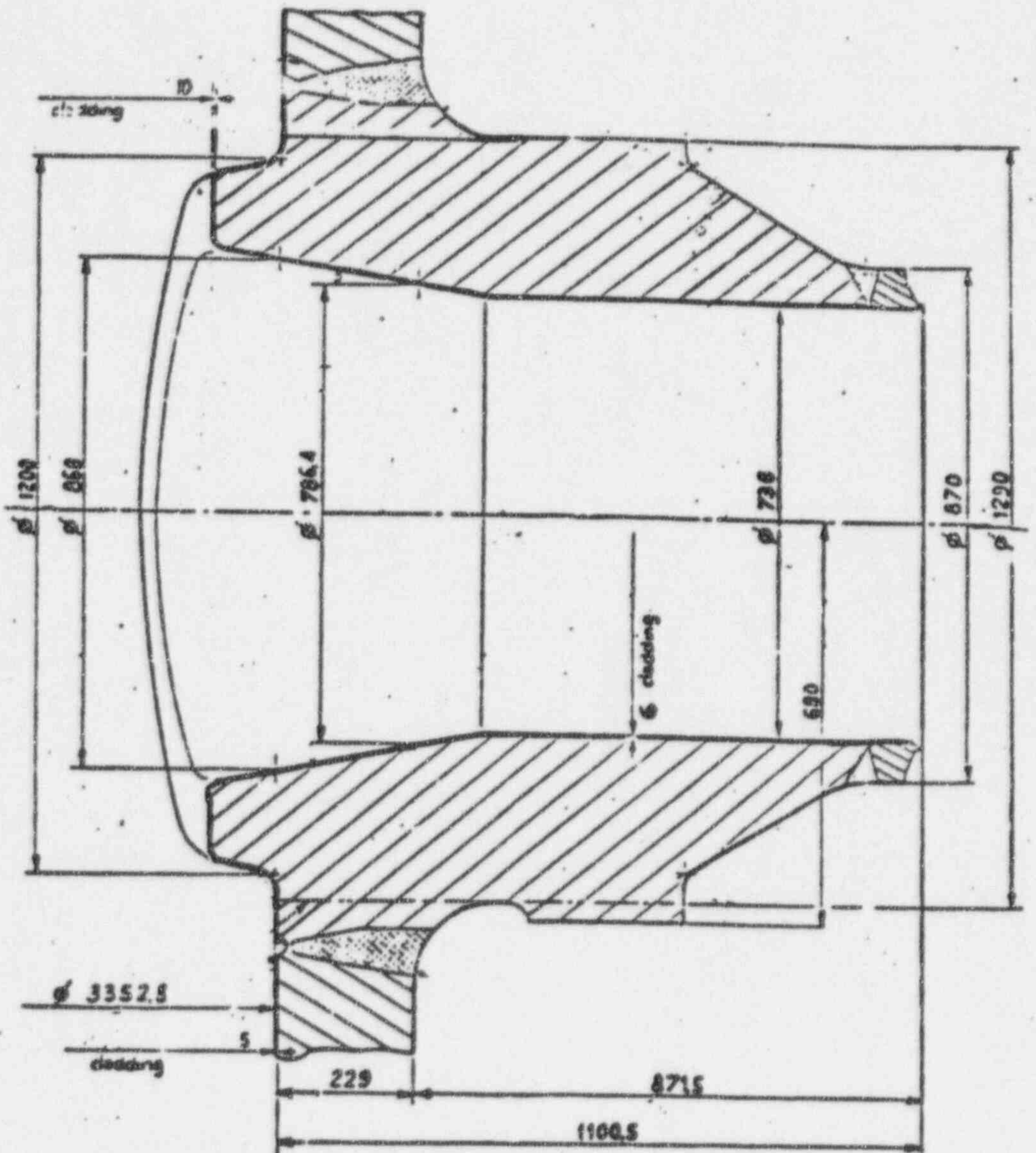


Figure 1 Geometry of the Reactor Vessel Outlet Nozzle for Prairie Island Units 1 and 2

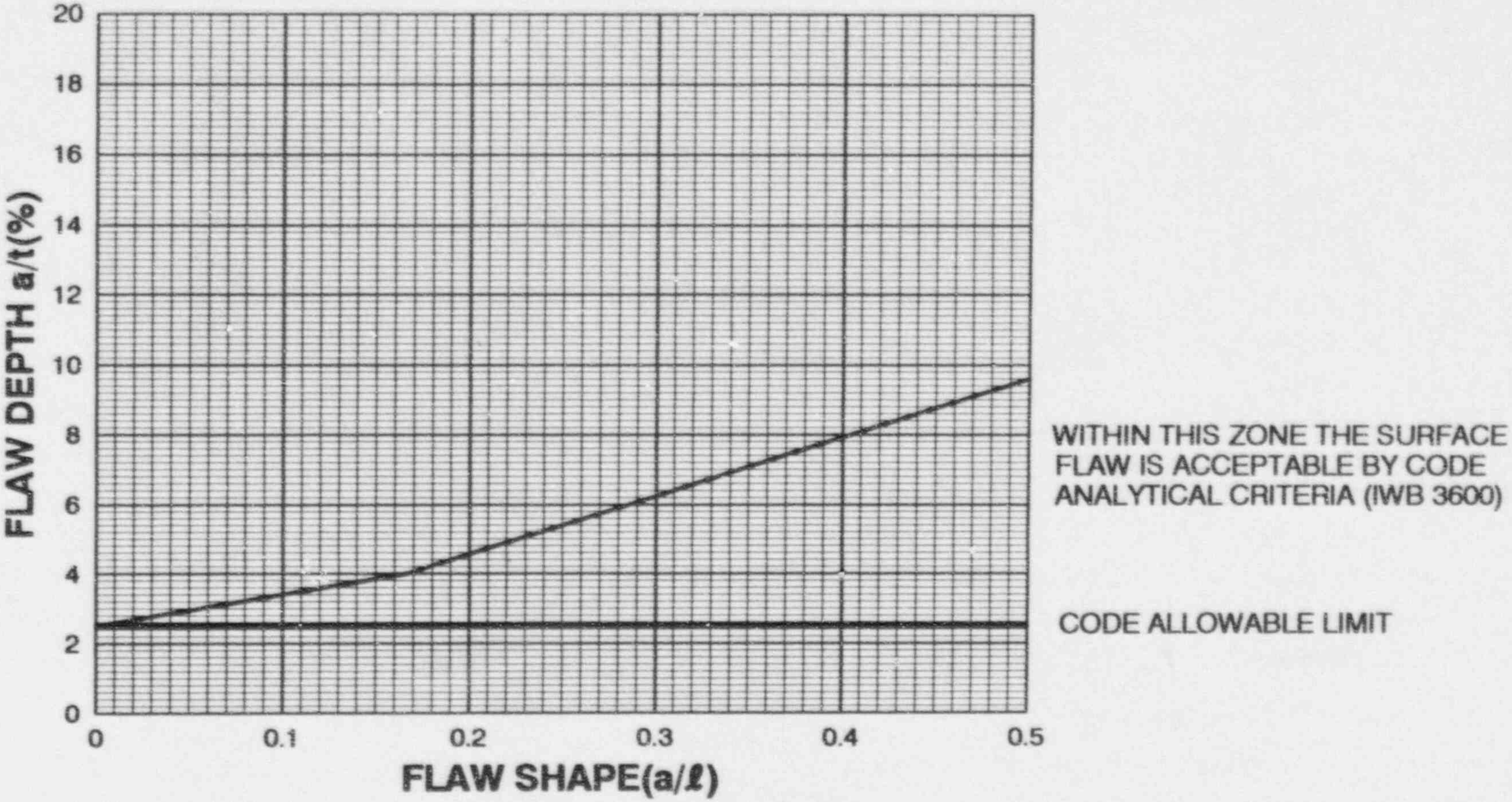


Figure 2. Surface Flaw Evaluation Chart for Code Allowable Flaws per Section XI IWB 3600 at the Outlet Nozzle Inner Radius for Prairie Island Units 1 and 2

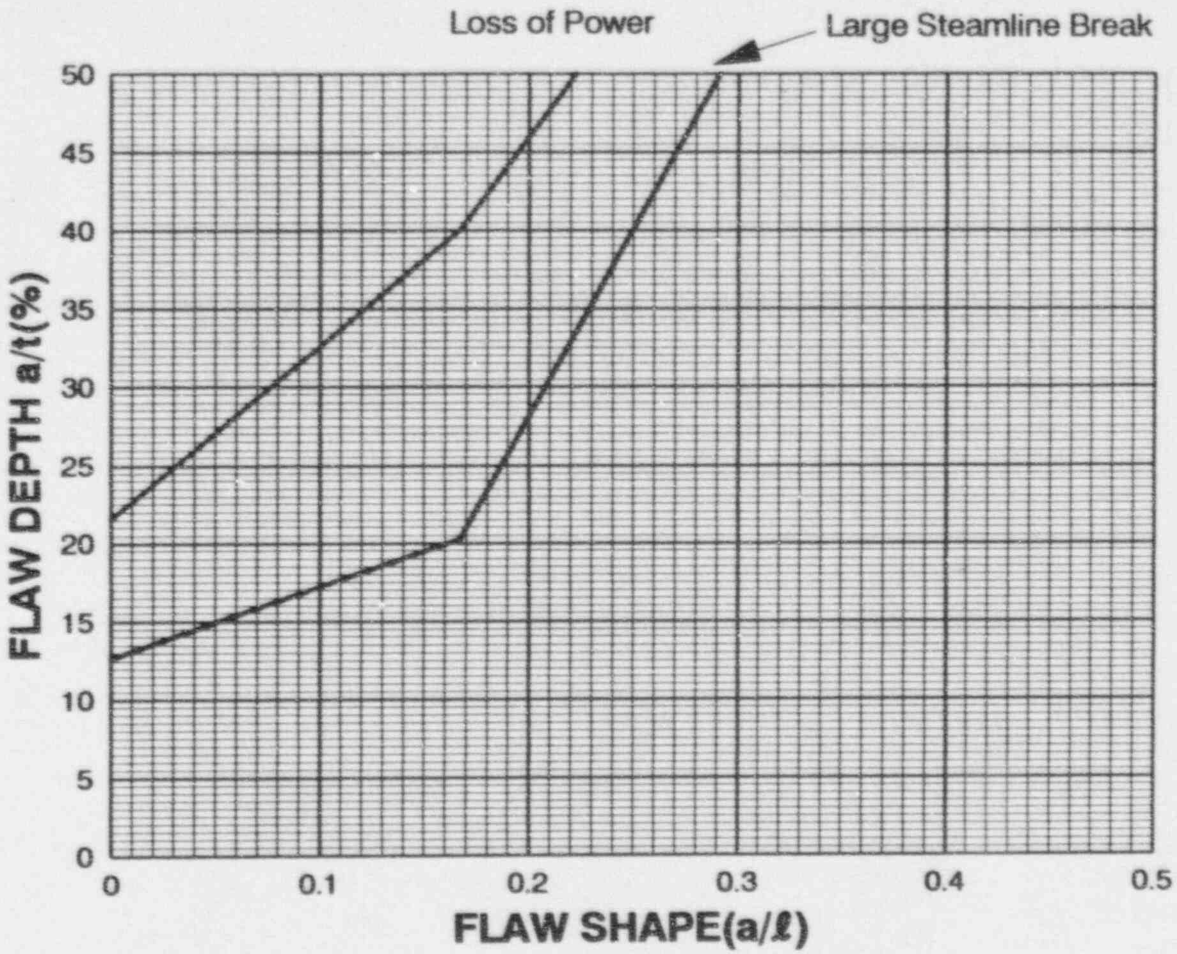


Figure 3. Surface Flaw Evaluation Chart Showing the Critical Flaw Depth at the Outlet Nozzle Inner Radius for Prairie Island Units 1 and 2

Table 1 Fatigue Crack Growth Results for the Outlet Nozzle Inner Radiue: Prairie Island Units 1 and 2(1)

Continuous Surface Flaw

Initial Depth	Depth After Year			
	10	20	30	40
0.10	0.130	0.193	0.302	0.461
0.125	0.185	0.290	0.443	0.663
0.25	0.390	0.588	0.861	1.22
0.375	0.572	0.840	1.20	1.64
0.50	0.748	1.07	1.49	1.99
0.75	1.09	1.50	2.01	2.61
1.0	1.41	1.90	2.48	3.16
1.25	1.72	2.27	2.91	3.69

Semi-elliptic Surface Flaw, a/l = 0.5

Initial Depth	Depth After Year			
	10	20	30	40
0.125	0.134	0.142	0.151	0.160
0.250	0.266	0.282	0.288	0.317
0.375	0.402	0.430	0.462	0.497
0.500	0.543	0.592	0.649	0.718
0.750	0.840	0.936	1.039	1.151
1.0	1.112	1.23	1.35	1.48
1.25	1.38	1.52	1.66	1.81

APPENDIX A - FRACTURE ASSESSMENT DETAILS

To complete the fracture assessment, all the design transients for the Prairie Island Units 1 and 2 were considered, as listed in Table 2-1 of Reference 1 and attached here for completeness as Table A-1. A detailed finite element analysis was completed for this region of the reactor vessel, and each transient was analyzed. The stress distributions through each cross section of interest were obtained from these results and the maximum and minimum hoop stresses for each transients were used in the fatigue crack growth analysis.

The material properties used in this evaluation, the reference fracture toughness curves and the fatigue crack growth law, have not changed since the analyses were completed for reference [1].

There are no other sources of stresses which would affect the outlet nozzle corner region. There are no structural welds, and no repairs have been made. Therefore there are no weld residual stresses. The piping loads are dissipated before the corner is reached, because of the reinforcement of the nozzle bore as shown in Figure 1. The other source of stress is from the cladding residual stress, but these stresses are so low as to be negligible for the temperature range of interest, above 200°F. To investigate this concern, a series of experimental measurements of clad residual stresses were made at a range of temperatures, as documented in reference (6). Tests were made on sample blocks removed from a commercial reactor vessel nozzle dropout. The hole drilling method was used to measure the residual stresses through the thickness, starting with the cladding and progressing into the base metal. The results are shown in Figures A-1 through A-3. Each figure shows the results obtained for two samples tested at the temperature noted. Twenty-five locations were measured, and the results were very consistent between the two samples for each temperature. At 70°F, the stresses in both the clad and base metal reach a maximum of 20 ksi, but near the clad-base metal interface the stresses are compressive. At 200°F, the stresses in both the clad and base metal peak at about 10 ksi, while at 400°F, the stresses are even lower, peaking at about 5-8 ksi. As the temperature increases, the clad residual stresses are continually decreasing, as we approach the stress-free temperature corresponding to the conditions when the clad was originally deposited. These residual stress measurements were made only in the direction parallel to the clad-base metal interface, but the residual stresses perpendicular to the interface are generally of the same levels, as shown in Figure A-4, from reference (6). These results show that the clad residual stresses trend toward low values as the temperature increases.

Separately, a series of flawed bend bars was tested with and without cladding, to see what the effect of the clad and residual stresses would be on the failures. The test set-up and specimen design is shown in Figure A-5, and the results are shown in Figure A-6. The results are presented in terms of the mouth opening of the specimen at failure, which is proportional to the crack driving force. The clad specimens failed at consistently lowe[illegible]k mouth opening displacements at the lower test temperatures. This is because of the residual stres[illegible]hich increase as the temperature decreases. At temperatures above 150°F, there is no effect of the cladding on the results, showing that the residual stresses have no measurable effect.

Since the lowest temperature of the transients of concern in evaluating the nozzle inner radius is 210°F, the effect of clad residual stresses can be neglected. This behavior has been studied for irradiated specimens by McCabe, who reached similar conclusions(8).

The governing normal and upset transient was found to be the Loss of Power transient, and the governing emergency and faulted condition was the Large Steamline Break. The stress distributions for each of these transients are tabulated in Table A-2. The temperature for the Loss of Power transient never drops below 500°F, and the corresponding lowest water temperature for the large steamline break is 210°F as shown in Figure A-7. The initial RT_{NDT} of the nozzle forgings is 5F for the Unit 1 outlet nozzles and -5F for the Unit 2 nozzles(1).

The effect of irradiation from the core on the outlet nozzle regions is negligible, since these nozzles are far above the core region. Therefore the fracture toughness will always be on the upper shelf for these and all other transients, so the fracture toughness was set at 200 ksi $\sqrt{in}$.

The expression for stress intensity factor was taken from the work of Raju and Newman [3] for a surface flaw in a cylinder. This expression has been shown to be conservative for a nozzle corner flaw through later work which actually modelled a nozzle corner flaw in a Westinghouse PWR inlet nozzle. The results for the PWR nozzle corner flaw were used in the flaw evaluation charts of reference 1, but these expressions are not published, so the results shown here were obtained using the Raju-Newman approach, so they can be easily verified. The nozzle crotch dimension is used as the thickness of the cylinder.

The fracture assessment reported here is a direct application of the rules of Section XI Appendix A. No exceptions were taken to any of the guidelines contained therein. The acceptance criteria of Section XI have not changed since its inception in 1974, and the material properties have not changed since the fatigue crack growth law for water environments was last revised in 1980.

Table A-1 Summary of Reactor Vessel Transients from Table 2-1 of (1)

NUMBER	TRANSIENT IDENTIFICATION	NUMBER OF OCCURRENCES	
		SPECIFIED	USED IN THE ANALYSIS
	<u>Normal Conditions</u>		
1	Heatup and Cooldown at 100°F/hr (pressurizer cooldown 200°F/hr)	200	200
2	Load Follow Cycles (Unit loading and unloading at 5% of full power/min)	18300	18300*
3	Step load increase and decrease of 10% of full power	2000	2000
4	Large step load decrease, with steam dump	200	200
5	Steady state fluctuations	Infinite	10^6
	<u>Upset Conditions</u>		
6	Loss of load, without immediate turbine or reactor trip	80	80
7	Loss of power (blackout with natural circulation in the Reactor Coolant System)	40	40
8	Loss of flow (partial loss of flow, one pump only)	80	80
9	Reactor trip from full power	400	400
	<u>Faulted Conditions</u>		
10	Large Loss of Coolant Accident (LOCA)	1	1
11	Large Steam Line Break (LSB) (other transients described in Section 4)	1	1
	<u>Test Conditions</u>		
12	Turbine roll test	10	10
13	Primary Side Hydrostatic test conditions	50	50
14	Cold Hydrostatic test	5	10

*6000 of these transients were used in the analysis of the outlet nozzle region, based on plant records, as documented in reference [3].

Table A-2 Stress Distributions for the Governing Transients: Outlet Nozzle Inner Radius for Prairie Island Units 1 and 2

Loss of Power		Large Steamline Break	
Location(x/t)	Stress	Location(x/t)	Stress
0	45.35	0	67.2
0.59	44.9	0.59	63.1
.166	37.0	.166	58.8
.332	37.2	.332	55.0
.399	37.2	.399	50.8
.474	36.9	.474	43.9
.542	36.3	.452	35.6
.613	35.8	.613	27.2
.682	35.4	.682	19.7
.751	35.2	.751	13.8
.824	35.2	.824	9.8
.896	35.7	.896	8.2
.965	36.8	.965	9.4
1.0	36.8	1.0	9.8

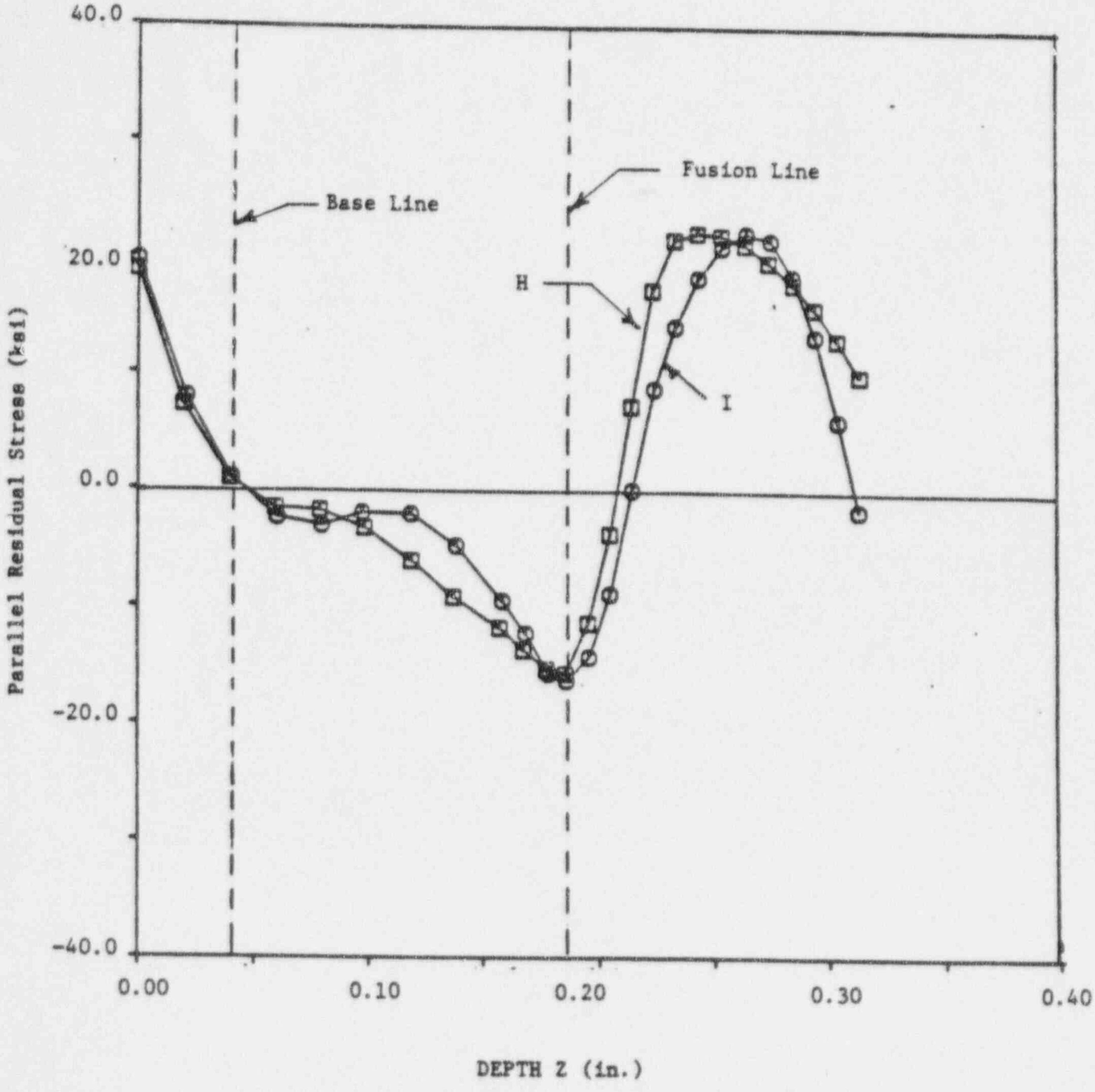


Figure A-1 Residual Stresses Parallel to the Clad-Base Metal Interface as Measured at 70°F on Specimens H and I

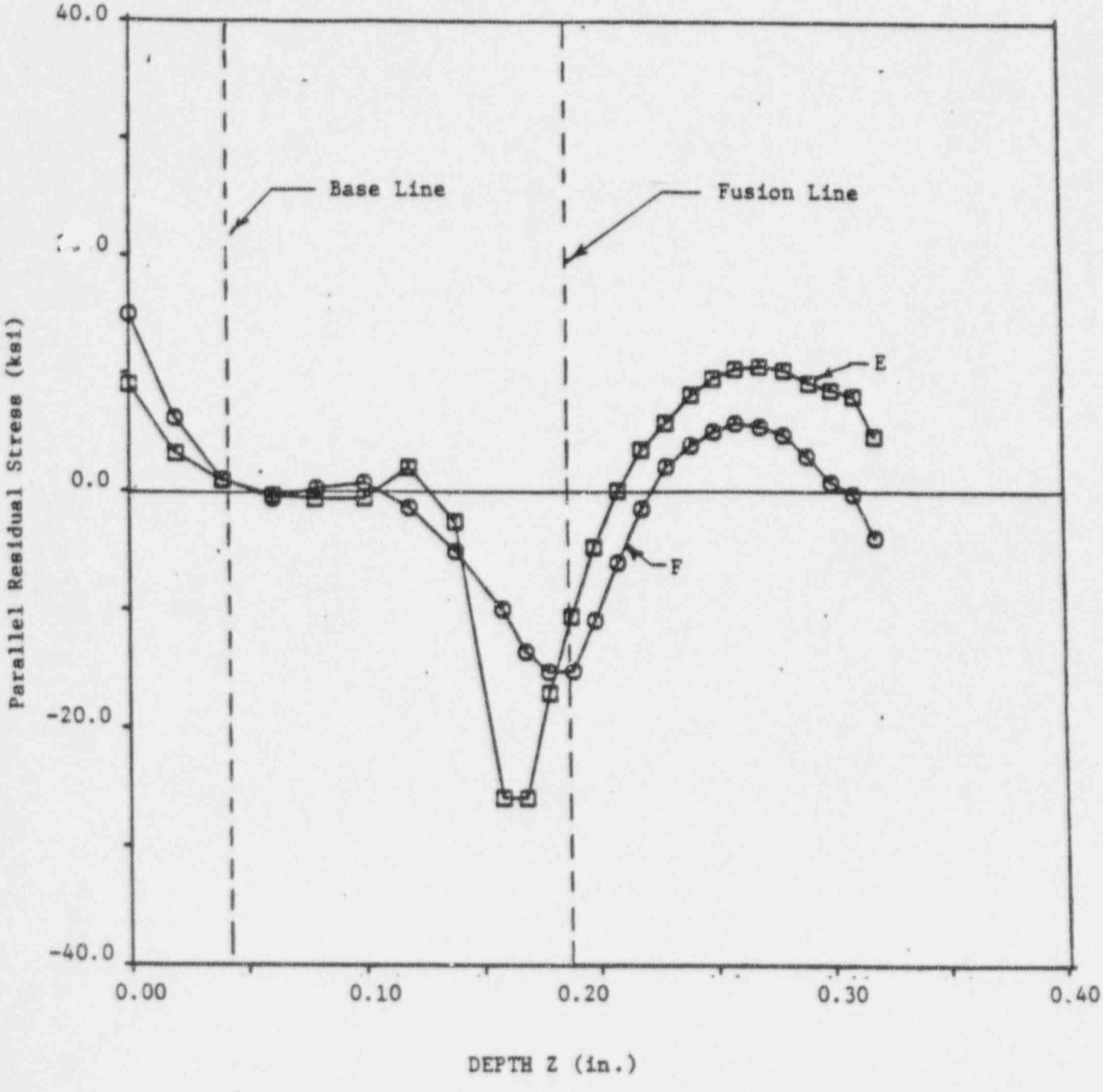


Figure A-2 Residual Stresses Parallel to the Clad-Base Metal Interface as Measured at 200°F on Specimens E and F

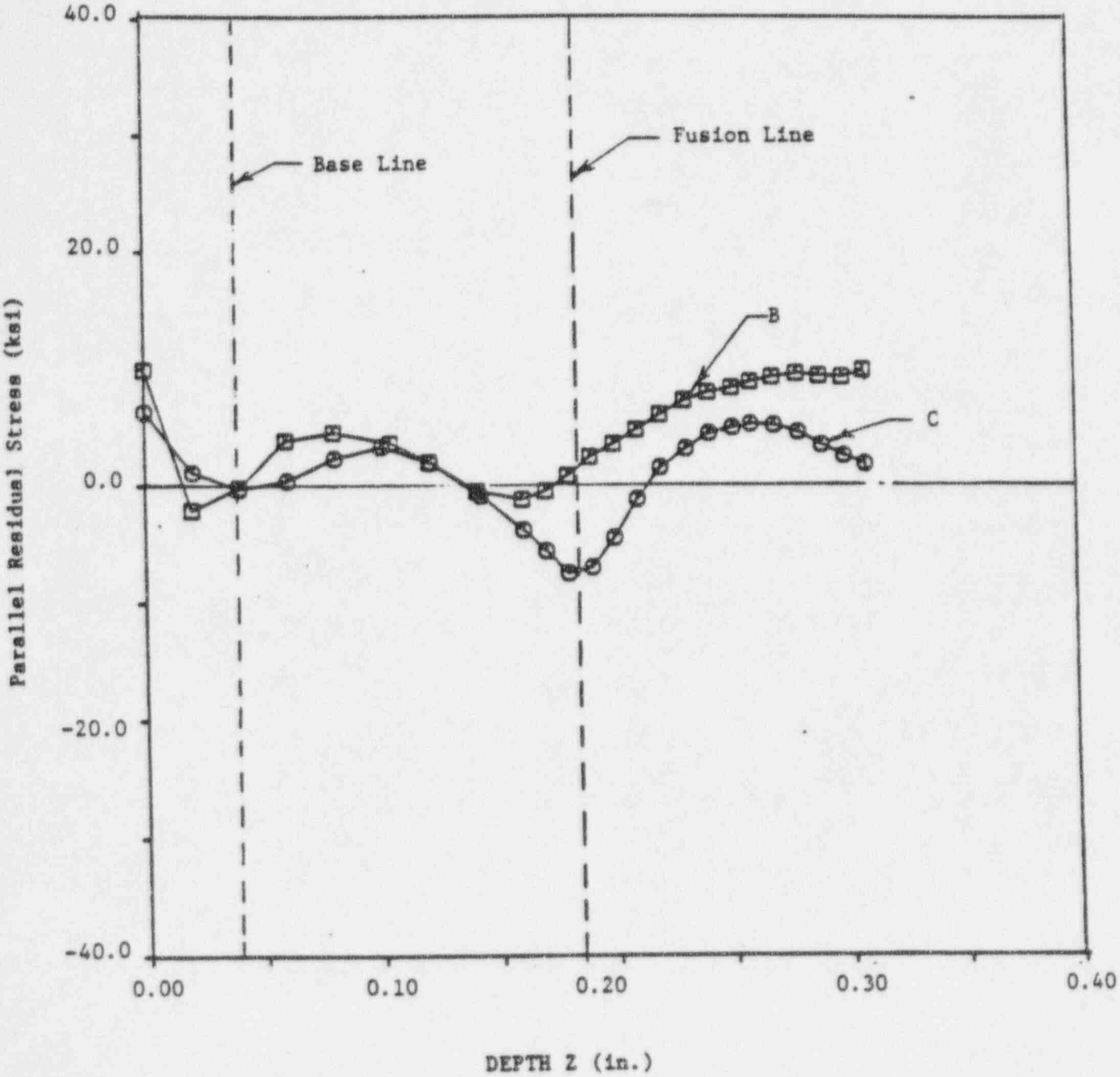


Figure A-3 Residual Stresses Parallel to the Clad-Base Metal Interface as Measured at 400°F on Specimens B and C

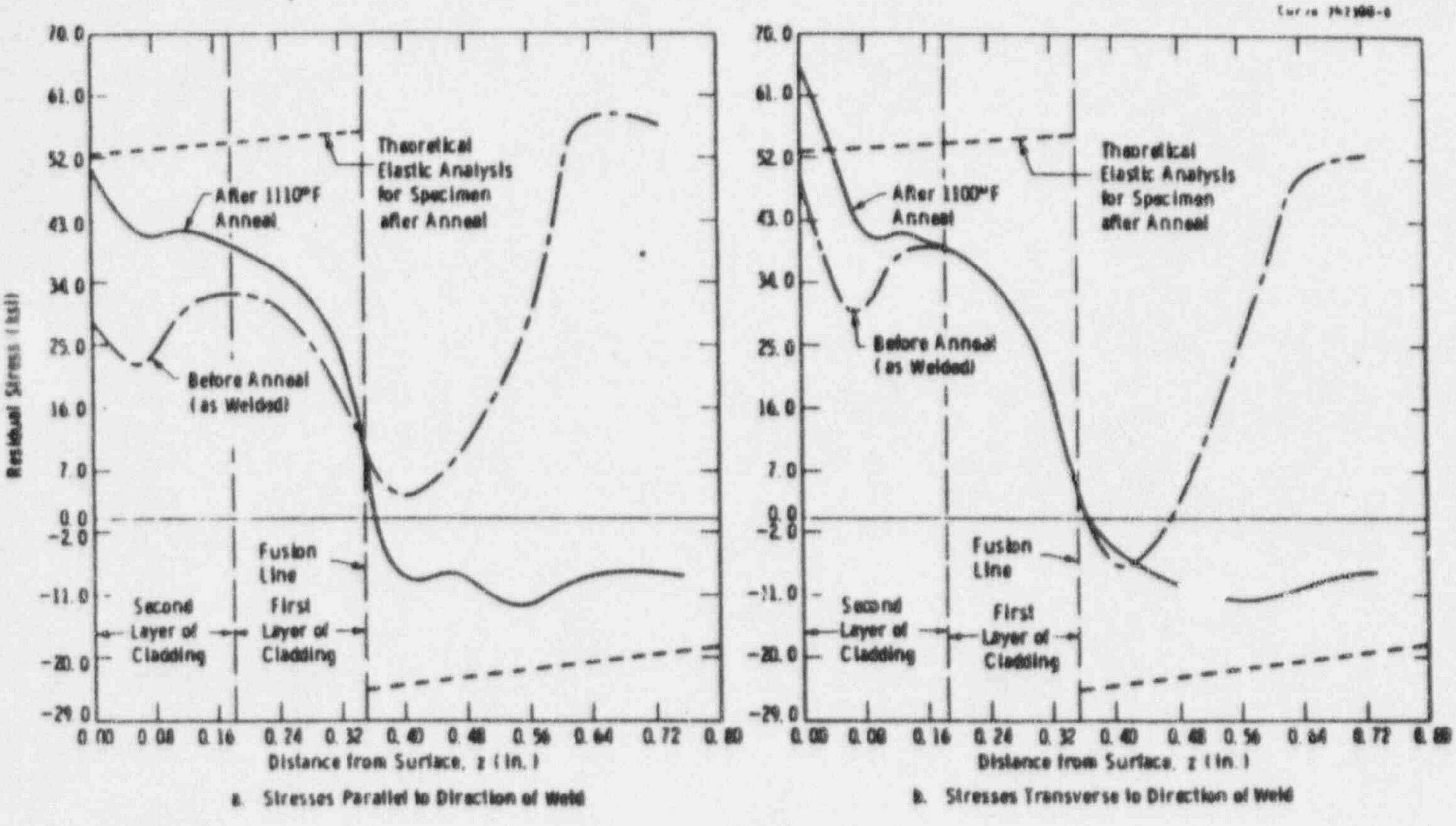
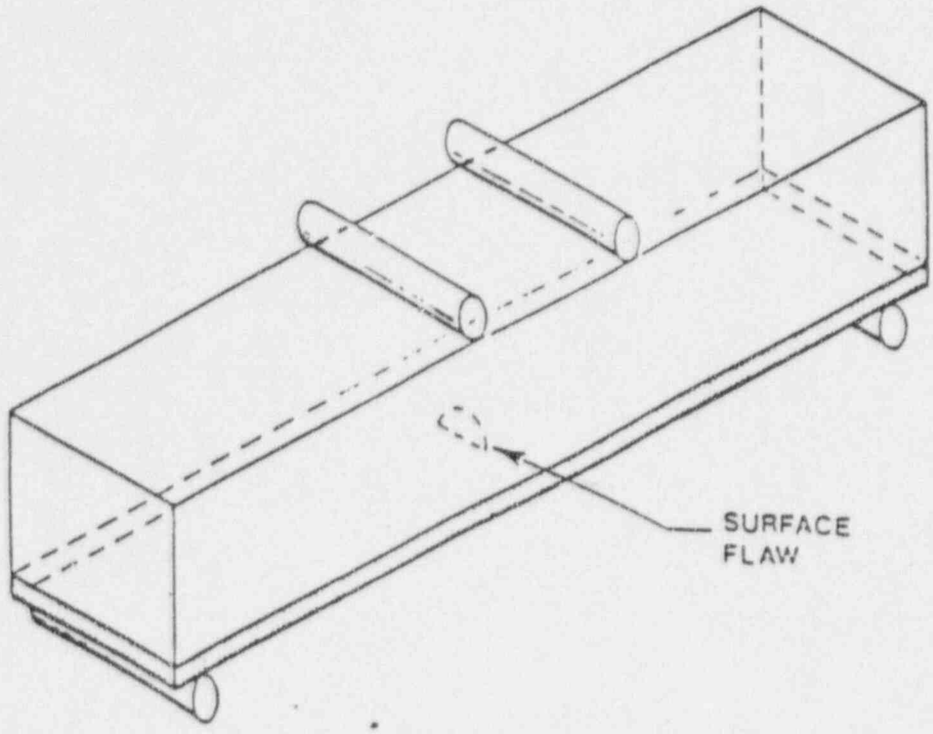


Figure A-4 Residual Stresses in a Stainless Steel Clad Block, Before and After 1150°F Anneal (Measurements are the Average of Two Specimens)(7)



FOUR POINT LOADED BEND BAR

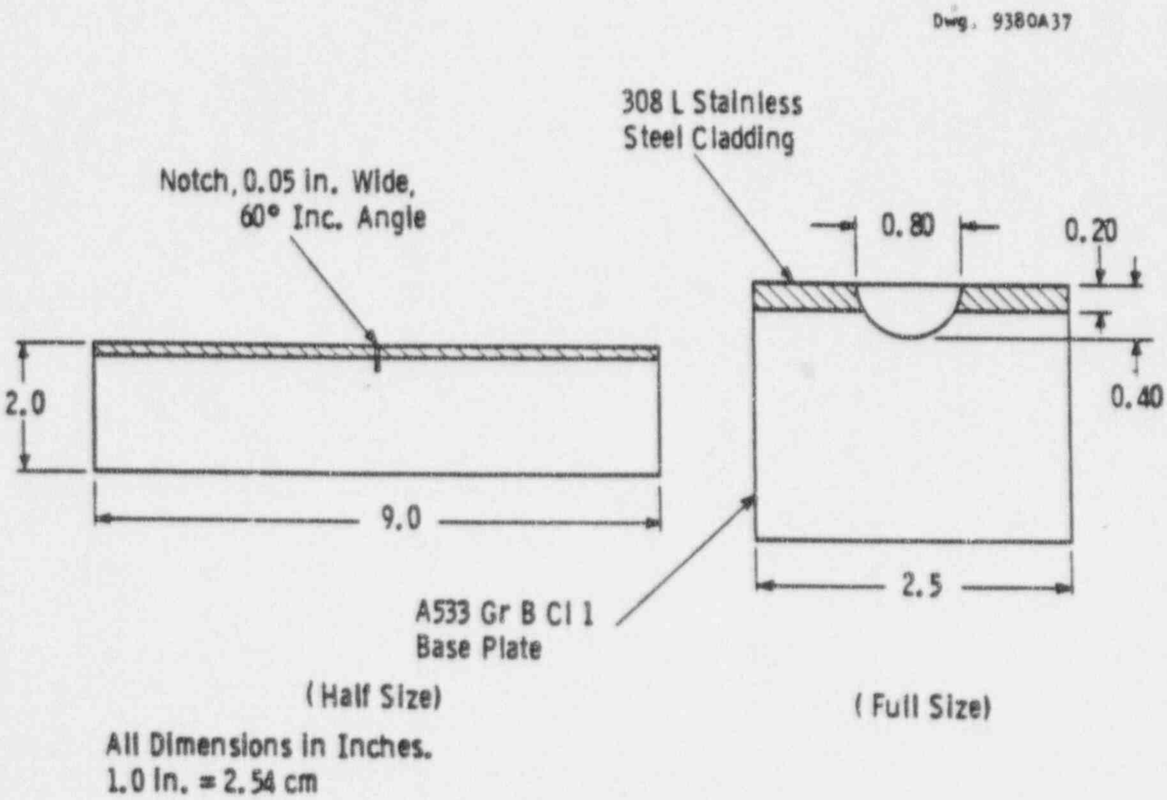


Figure A-5 Test Set-up and Specimen Geometry for Bend Bar Tests to Measure Clad Effects vs. Temperature

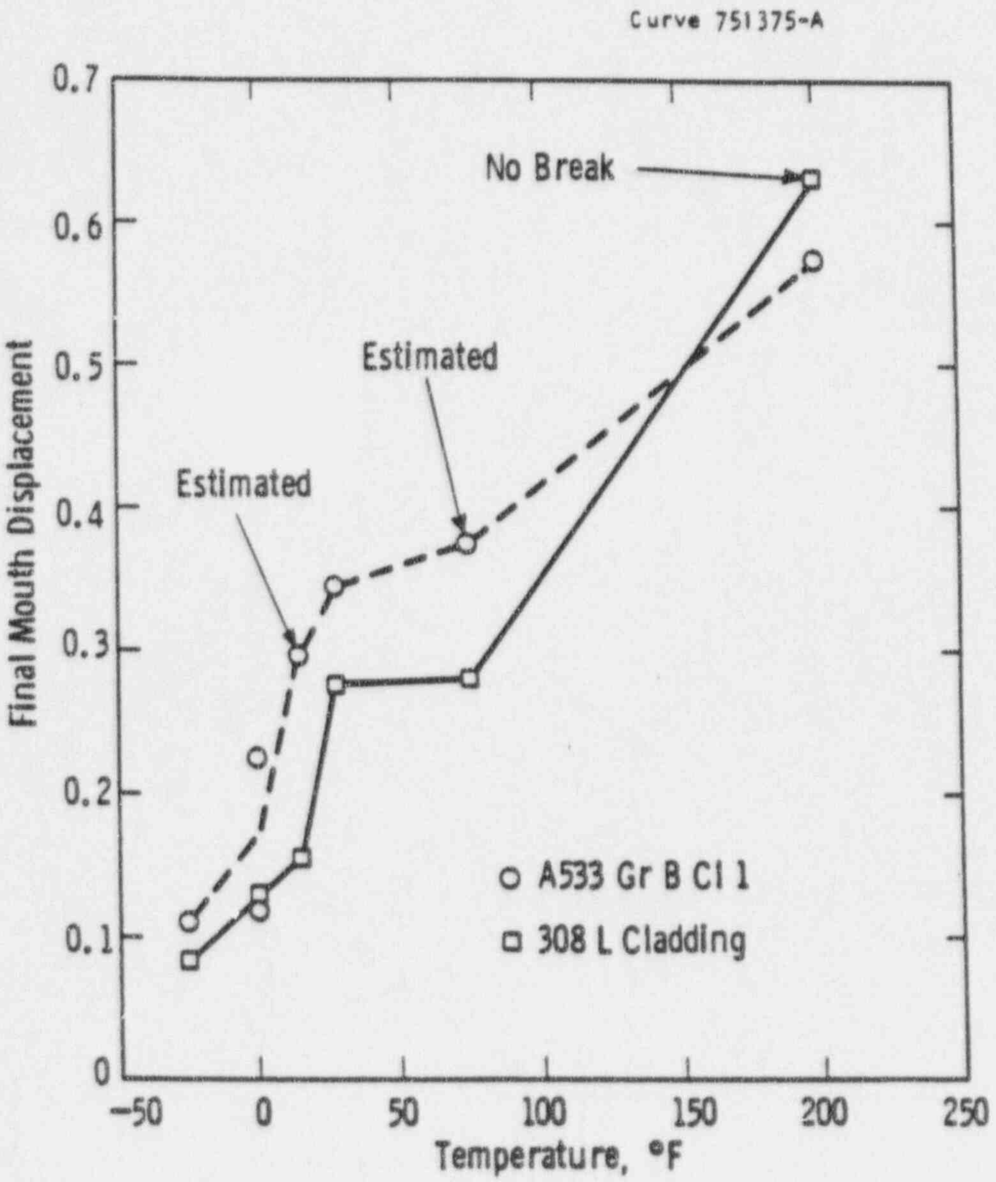


Figure A-6 Final Mouth Opening Displacement Versus Temperature for Clad and Unclad Bend Bars

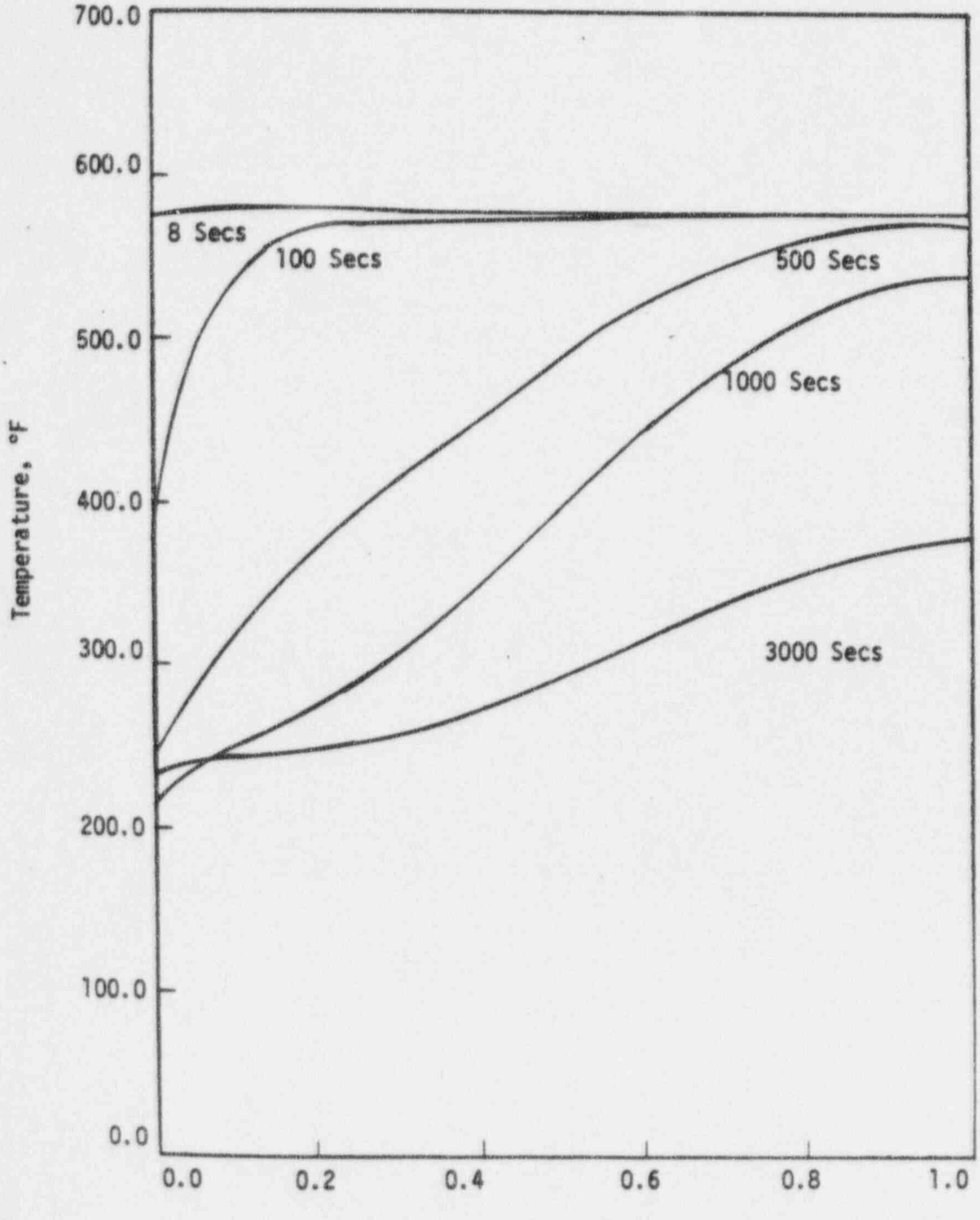


Figure A-7 Temperature History due to Large Steamline Break-Clad Model-Nozzle Corner, from Ref. 5, Figure 4-20