



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 21, 1994

Docket Nos. 50-244, -50-250, 50-251,
50-266, 50-269, -50-270,
- 50-280, 50-281, 50-287,
50-289, -50-295, 50-301,
50-302, 50-304, -50-313,
and 50-346

See pt.

MEMORANDUM FOR: John N. Hannon, Director
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

FROM: Jon Hopkins, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

SUBJECT: SUMMARY OF MEETING HELD ON MAY 19, 1994, WITH BABCOCK AND
WILCOX REGARDING REACTOR VESSEL INTEGRITY

On May 19, 1994, NRC staff members met in Rockville, Maryland, with Babcock and Wilcox (B&W). A list of attendees is included as Enclosure 1. The handout used at the meeting is included as Enclosure 2.

As shown in Enclosure 2, the items discussed during the meeting included reactor vessel integrity program, fracture mechanics methodology, bounding embrittlement trends, microstructural studies, and NRC comments. By letter dated May 23, 1994, B&W confirmed that the information contained in Enclosure 2 may be considered non-proprietary.

Babcock and Wilcox said that it would provide a common response to Generic Letter 92-01. After the meeting, B&W agreed to provide this response 30 days subsequent (circa June 30, 1994) to the last response letter from the NRC to licensees (circa May 30, 1994). The NRC staff said it would issue a NUREG related to Generic Letter 92-01 data in July 1994.

B&W proposed a new method to determine the RT_{RTS} value of Linde 80 welds relative to the pressurized thermal shock (PTS) screening criteria in the PTS rule. The proposed method is a proactive initiative to address plant life extensions. The new method would apply to all Linde 80 welds fabricated by B&W. In the proposed method, all B&W fabricated Linde 80 welds would have an unirradiated reference temperature of -10°F and a standard deviation of the unirradiated reference temperature of 0°F . The standard deviation of the adjusted reference temperature would be 14°F when credible surveillance data exists and 25°F when no surveillance data exists.

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PDR ADDCK 05000244
P PDR

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For memo

Mr. John N. Hannon

- 2 -

Data previously provided by the B&WOG reported that B&W fabricated Linde 80 welds have an unirradiated reference temperature of -5 °F and a standard deviation of unirradiated reference temperature of 17 °F. RG 1.99, Revision 2 reports that the standard deviation for the adjusted reference temperature is 14 °F when credible surveillance data exists and 28 °F when no surveillance data exists. The standard deviation in the unirradiated reference temperature and the adjusted reference temperature are needed to cover uncertainties in the values of the unirradiated reference temperature, copper and nickel contents, neutron fluences and calculational procedures. The NRC staff indicated that the B&WOG must provide fracture toughness data to address the uncertainties and to demonstrate that the proposed new methodology is applicable to all B&W fabricated Linde 80 welds. At the end of the meeting, the NRC staff and Babcock and Wilcox agreed to continue to communicate on this issue.

Original Signed By:
J. B. Hopkins

Jon B. Hopkins, Sr. Project Manager
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Enclosures:

1. List of Attendees
2. Meeting Handouts

cc: See next page

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6/10/94	6/17/94	6/17/94	6/17/94	6/14/94	6/21/94
YES/NO	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO

Name: G:\BWOOG.MTG



1957

1958

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1960

1961

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and Babcock and Wilcox

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E. Hackett	NRC
S. Collard	Florida Power & Light
K. Yoon	BWNT
K. Cozers	NEI
L. Connor	STS
G. West	NRC
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S. Sheng	NRC
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K. Moore	BWNT
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Docket File
NRC & Local PDRs
PDIII-3 R/F
W. Russell/F. Miraglia 012G18
L. Reyes 012G18
J. Roe 013E4
J. Zwolinski 013H24
M. Rushbrook 013E21
D. Pickett 013E21
J. Hannon 013E21
OGC 015B18
E. Jordan 04D18
ACRS (10) P315
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EDO Region II Plants 017G21
EDO Region IV Plants 017G21
G. Greenman, Region III
E. Merschoff, Region II
R. Cooper, Region I
B. Beach, Region IV
A. Johnson 014D1
R. Crotean 014H22
A. Hansen 013E21
L. Wiens 014H25
B. Buckley 014H22
R. Hernan 014C7
C. Shiraki 013D9
L. Raghavan 014H15
- G. Kalman 013H6
- R. Stransky 013E21

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**THE
B&W****OWNERS GROUP****MATERIALS COMMITTEE**

Meeting with NRC Staff

Rockville, Maryland

May 19, 1994

**B&W NUCLEAR
TECHNOLOGIES**

REACTOR VESSEL INTEGRITY PROGRAM
OVERVIEW FOR NRC

by

THE B&W OWNERS GROUP
REACTOR VESSEL WORKING GROUP

MAY 19, 1994

AGENDA

			<u>Section</u>
1:00 pm	Introduction - Meeting Purpose	G. L. Lehmann/ K. R. Wichman	
1:15	Reactor Vessel Integrity Program	K., E. Moore	1
1:30	Fracture Mechanics Methodology	K. K. Yoon	2
2:00	Bounding Embrittlement Trends	G. L. Lehmann	3
2:30	Break		
2:45	Microstructural Studies	W. A. Pavinich	4
3:15	NRC Comments	NRC Staff	
	- Program Critique		
	- NRC-Sponsored R&D		
	- Unresolved Issues		
4:00	Adjourn		

REACTOR VESSEL INTEGRITY PROGRAM

- Objectives
- Status of Integrated Surveillance Program
- Current tasks
- Plan for future work

RVIP (Cont.)

RVWG Plan and Commitment

- Obtain sufficient actual data and verified analytical methodology to demonstrate the safety of their reactor vessels
- Minimize the effect of reactor vessel integrity issues on plant operation

Duke Power Co Entergy Operations Florida Power Corp GPU Nuclear Corp Toledo Edison Co	Oconee-1, 2, 3 ANO-1 Crystal River-3 TMI-1 Davis-Besse
Commonwealth Ed Florida P&L Co Rochester G&E Virginia Power Wisconsin EP Co	Zion-1, 2 Turkey Point-1, 2 R. E. Ginna Surry-1, 2 Point Beach-1, 2

RVIP (Cont.)

Master Integrated Reactor Vessel Surveillance Program (MIRVP) Review and Update

- BAW-1543, Rev. 4
- ~100 plant specific RVSP capsules
 - 8 B&W 177-FA plants
 - 9 W plants with B&W fabricated reactor vessels
- Test reactor irradiations
 - ORNL/HSST program
 - NRC/NRL program



RVIP (Cont.)

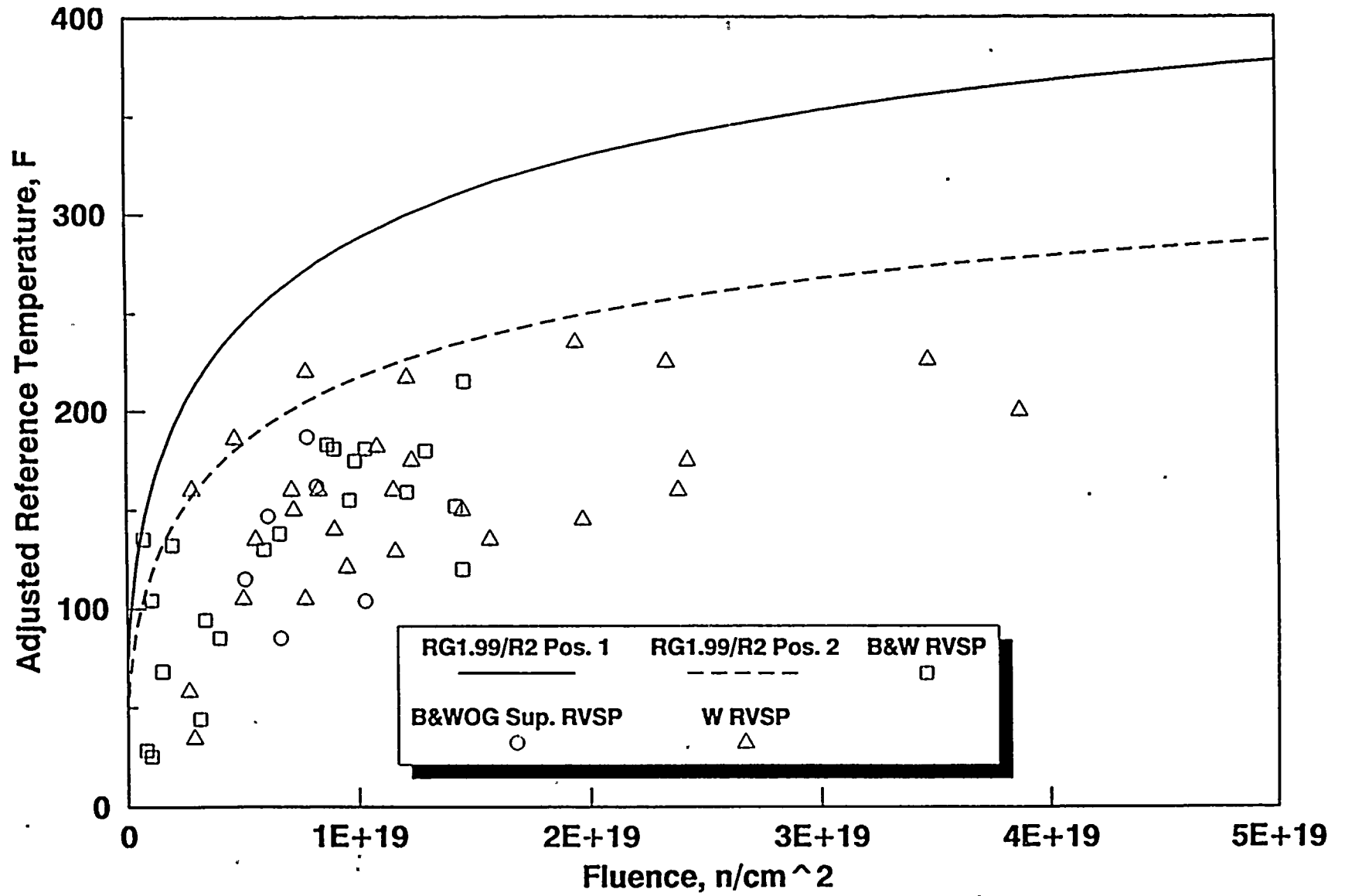
MIRVP Review and Update (Cont.)

- 14 RVWG capsules in 3 power reactors
 - 13 pertinent Linde 80 weld metals
 - Fluences from 0.6E19 to 3E19
 - 2 capsules to be used for annealing study including reirradiation
 - Specimens include 1T compact tension

- 2 RVWG capsules were tested
- 2 RVWG capsules to be tested in 1994
- Schedule extends through 2008

- 1994 RVWG capsules include one at 1.6E19 n/cm²
 - 177-FA IS 48 EFPY
 - W T/4 32 EFPY

Linde 80 Weld Metals



RVIP (Cont.)

Five-Year Plan: List of Tasks

- I. PRESSURIZED THERMAL SHOCK (PTS)
 - (1) Fracture Mechanics Analytical Methods
 - (2) Microstructure and Material Properties
 - (3) Regulatory Response

- II. PRESSURE/TEMPERATURE OPERATING LIMITS
 - (1) ASME Appendix G Methodology

- III. LOW UPPER-SHELF ENERGY ISSUE (LUSE)
 - (1) LUSE Regulatory Issues

- IV. FRACTURE TOUGHNESS TEST METHODS
 - (1) Charpy-Size Specimens

RVIP (Cont.)

Five-Year Plan: List of Tasks (Cont.)

V. COMMUNICATION

- (1) Industry and Code Review
- (2) Communication with the NRC
- (3) Status/Information Exchange

VI. INFORMATION BASE

- (1) Fluence Tracking System
- (2) Beltline Material Data Base

VII. MASTER INTEGRATED RV MATERIAL SURVEILLANCE PROGRAM (MIRVP)

- (1) MIRVP Capsule Testing and Evaluation



RVIP (Cont.)

Current RVWG-sponsored efforts in analytical technology includes the following:

- Evaluation of and familiarization with the FAVOR code; which may become the basis of future PTS reassessment
- Engineering application of current and advanced fracture mechanics methods
- Alternative method for radiation embrittlement indexing

RVIP (Cont.)

RVWG is developing bounding embrittlement trend curves for the Linde 80 class of welds.

RVWG-sponsored microstructural studies aim to establish a meaningful basis for radiation-induced embrittlement.

RVIP (Cont.)

Summary

- Irradiated materials data is being provided by the MIRVP
 - Fracture toughness information
 - Complete range of reactor vessel neutron exposure
 - To investigate annealing and reirradiation
 - To investigate irradiation temperature effects

- Optimized methods for fracture mechanics test and analysis are being developed
 - Application to reactor vessel integrity assessment

**B&W OWNERS GROUP ACTIVITIES IN
FRACTURE MECHANICS METHDOLOGY**

**K. K. Yoon
B&W Nuclear Technologies**

at

**B&W Owners Group
Reactor Vessel Working Group
Meeting with NRC**

**Rockville Maryland
May 19, 1994**

CURRENT METHODOLOGY UPGRADE

1. Appendix G Methodology Update

Code Activities - Section XI

Residual Stress Analysis for B&W Fab. Vessels

Cladding Operation and PWHT

Cladding Model

Flaw Size Reduction

Thermal KI Model

2. PTS Analysis Methods

Review of FAVOR Code

New Influence Function

New Appendix A Method

3. Fracture Toughness Update

Cognizance of ORNL Activities

ADVANCED FRACTURE MECHANICS TOPICS

A. Cognizance of NRC Sponsored Advanced FM Methodolgy Developments

1. Two Parameter J Theory
Technology Follow-up
Application Efforts
2. Shallow Crack/Biaxial Loading Tests
3. ORNL Weibul Modeling of Fracture Toughness Curve
Draft 5, Proposed ASTM Test Practice for Fracture Toughness
in the Transition Range

B. Development of Application Methodologies - B&WOG

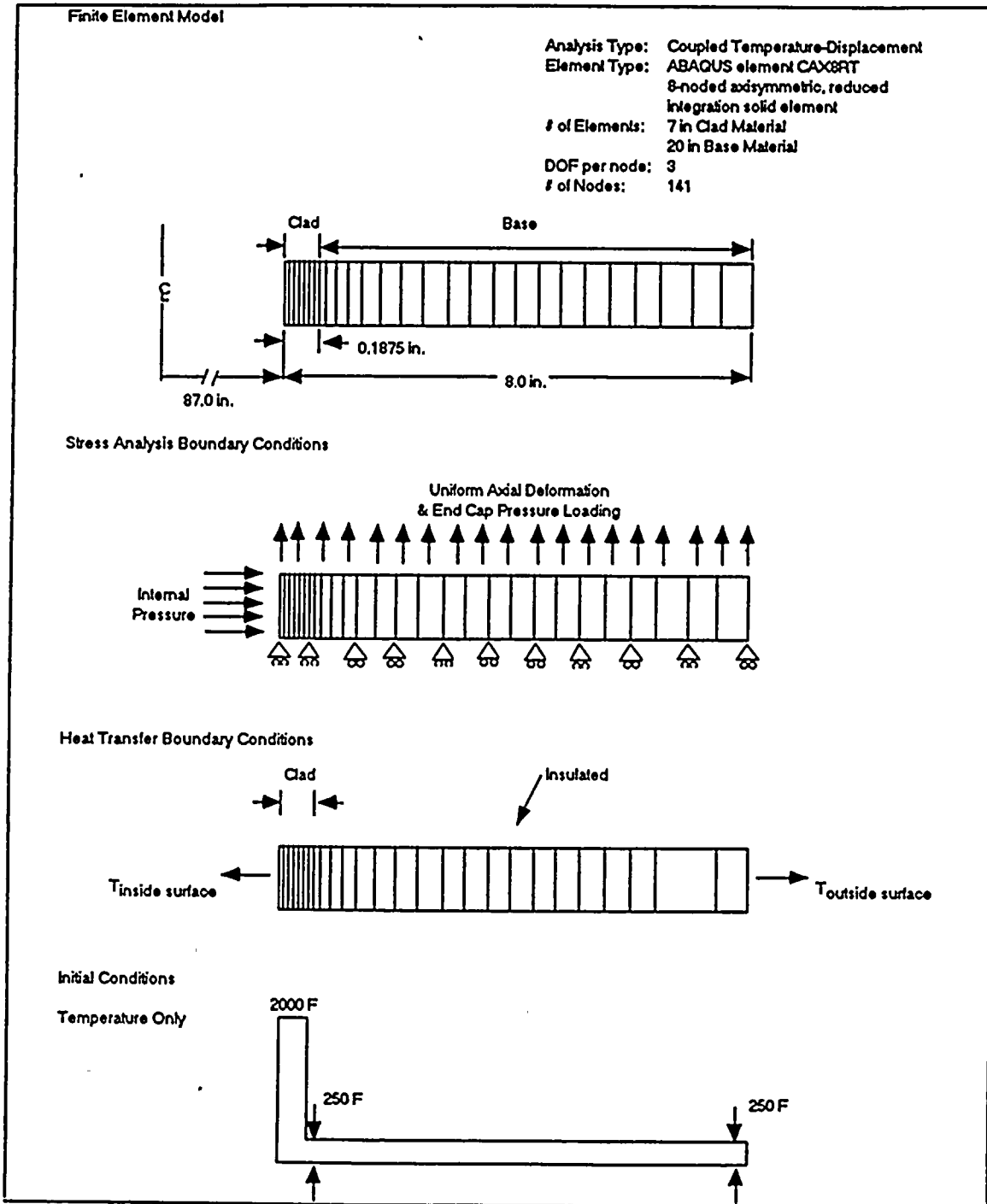
1. B&WOG Linde 80 Weld Metal Shallow Crack Testing
2. Dodds-Anderson Constraint Correction
3. Modified Boundary Layer Analysis
4. J-Q Analysis Using Linde 80 Weld Metal Properties
5. Alternative Method for Determining Initial RTNDT for Linde 80
Weld Metal
6. Charpy Size Specimen Fracture Toughness Testing in Transition
Temperature Range
7. Development of Direct Indexing Method for Irradiation
Embrittlement

CURRENT FRACTURE MECHANICS METHODOLOGY
UPDATE

Computation of Residual Stresses due to Application of Cladding in Reactor Pressure Vessels

III. Procedure

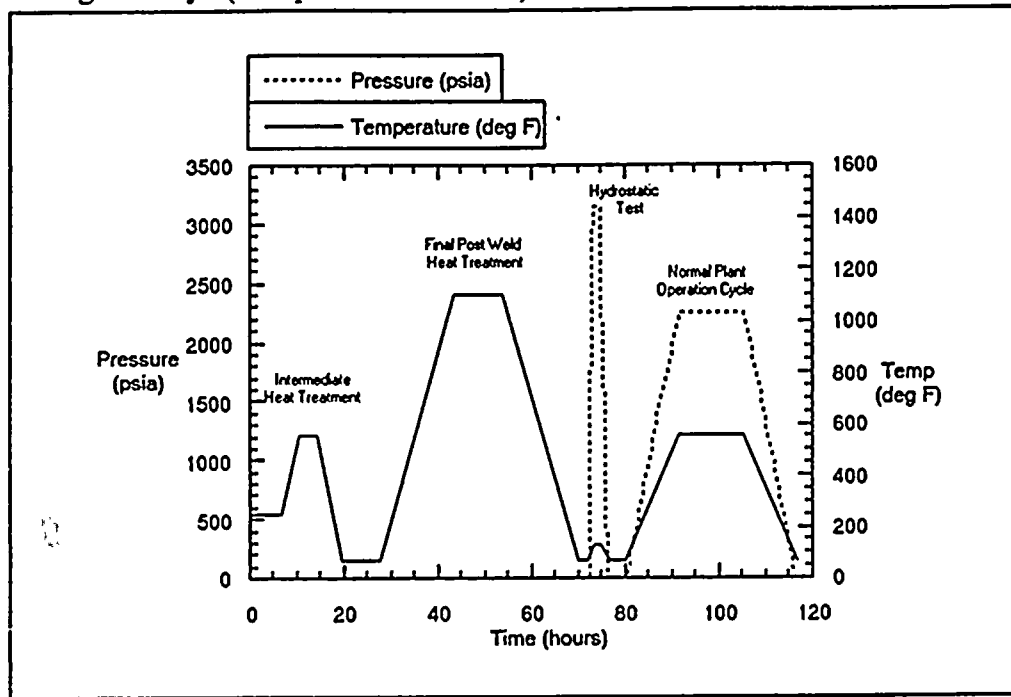
(a) 2-D Elastic-Plastic Creep Finite Element Analysis



Computation of Residual Stresses due to Application of Cladding in Reactor Pressure Vessels

III. Procedure (continued)

(b) Loading History (Temperature/Pressure)

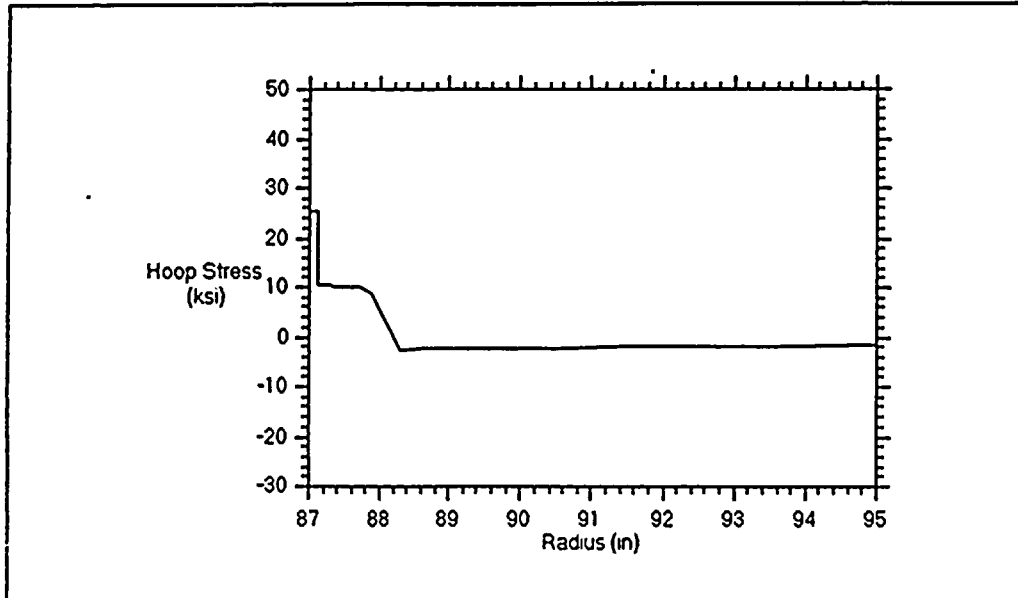


Step	Time (hrs)	Temp (deg F)	Pressure (psi)	Heat Transfer B.C.s		Procedure
				Inside Surface	Outside Surface	
	0	250	0	ambient	HS	
1	6.5	250	0	ambient	HS	Pre-Heat
2	10.5	550	0	ambient	HS	Intermediate Heat Treatment
3	14.5	550	0	ambient	HS	"
4	19.5	70	0	ambient	HS	"
5	27.5	70	0	ambient	ambient	
6	43.5	1100	0	HS	HS	Final Post Weld Heat Treatment
7	54	1100	0	HS	HS	"
8	70	70	0	HS	HS	"
9	72	70	0	ambient	ambient	
10	73.5	125	3150	HS	ambient	Shop Hydrostatic Test
11	75	125	3150	HS	ambient	"
12	76.5	70	0	HS	ambient	"
13	80	70	0	ambient	ambient	
14	91.5	550	2250	HS	insulated	Normal Operational Cycle
15	105.3	550	2250	HS	insulated	"
16	116.8	70	0	HS	insulated	"

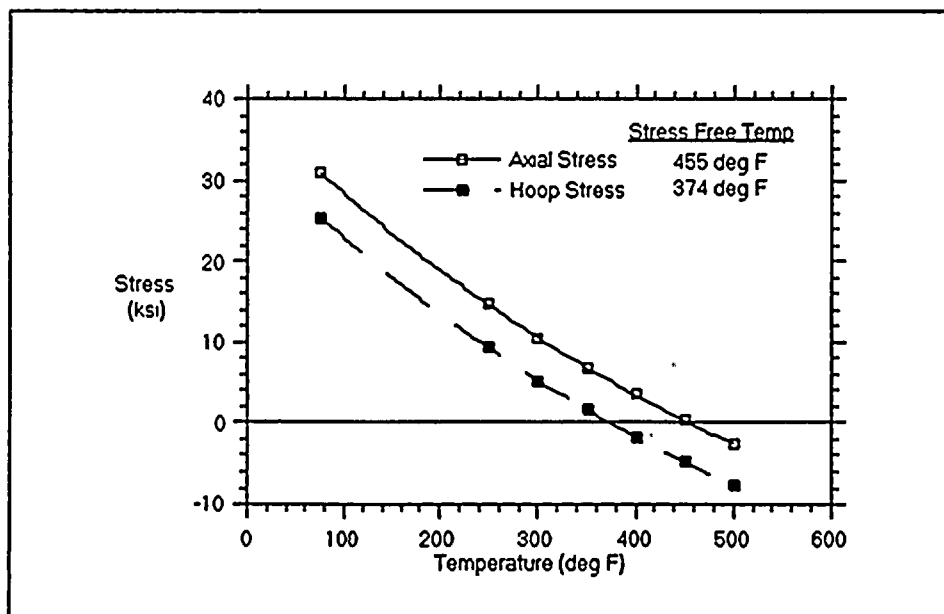
Computation of Residual Stresses due to Application of Cladding in Reactor Pressure Vessels

IV. Results

(a) General Stress Distribution (Same Trend for Both Axial and Hoop Stresses) Following Application of Cladding, Heat Treatment, and Initial Service



(b) Determination of Cladding Stress-Free Temperatures in RPV Following Application of Cladding, Heat Treatment, and Initial Service



Computation of Residual Stresses due to Application of Cladding in Reactor Pressure Vessels

V. Comparison and Conclusions

This work was based on the same general approach as given in Ref. 1. For this reason, we are able to make a direct comparison between the two approaches as shown below.

	BWNT		Ref. 1		% Difference	
	Axial	Hoop	Axial	Hoop	Axial	Hoop
Residual Stress (ksi)	30.91	25.41	28.50	24.00	-7.8	-5.5
Stress Free Temp (deg F)	455	374	~ 400			

- (1) Excellent correlation is found between BWNT and Ref. 1 for cladding stresses and the residual tensile stress field in the base material. All residual stress values were within 8% of one another.
- (2) Likewise, the predicted stress free temperatures for the cladding were equally reasonable. It is important to note that while the stress in the cladding relaxes as the temperature is increased, the residual tensile field in the base metal is unaffected.
- (3) The assumption used by most researchers that the reactor pressure vessel is stress free at the final post weld heat temperature temperature is non-conservative for shallow flaws as no residual tensile stress field exists in the base metal. This is primarily due to the fact that these references do not account for application of the cladding which is where the residual base metal stress field is created.

Ref. 1 B.R. Ganta, et. al., *Cladding Stresses in a Pressurized Water Reactor Vessel Following Application of the Stainless Steel Cladding, Heat Treatment and Initial Service*, ASME, PVP-Vol. 213, Pressure Vessel Integrity, 1991.

A REVIEW OF FLAW SIZE FACTOR BETWEEN SECTIONS III AND XI OF ASME BOILER AND PRESSURE VESSEL CODE

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Engineering and Project Services Division
B&W Nuclear Technologies
Lynchburg, Virginia

INTRODUCTION

This paper reopens the question of the appropriate safety factor to be applied to flaw size in Section XI, ASME Boiler & Pressure Vessel Code. This is intended to promote discussion on the appropriate safety factor and is not a proposal for any changes in the flaw acceptance standards of Section XI.

When the flaw acceptance standards (IWB-3500) for Section XI of ASME Code were developed, the acceptable flaw depth was selected as one-tenth of the Appendix G reference flaw⁽¹⁾, which is a semielliptical surface flaw with a depth of one-fourth of the wall thickness and a length of six times the depth. The acceptable flaw depth for a planar flaw with an aspect ratio equal 0.15 is then one-fortieth of the vessel wall thickness (IWB-3510-1 for wall thickness of 4-12 inches). The logic behind this selection was documented in Reference 2 by R. R. Maccary and it has remained the accepted basis. The acceptable flaw depth is approximately 0.2 inch for an eight inch-thick reactor vessel; and this flaw was close to the limit of flaw-sizing capability of NDE methodology at the time.

It has been more than 22 years since the first flaw evaluation procedure for pressure vessels was developed by PVRC Committee on Toughness Requirements and was published as WRC Bulletin 175 to become the basis for Appendix G of Section III. Recently, the Working Group on Operating Plant Criteria of Section XI, ASME Code, conducted a background study of Appendix G requirements regarding reactor vessel integrity⁽²⁾. When the Appendix G reference flaw size of a quarter of the wall thickness was determined, it was based on NDE techniques circa 1967. Since then there have been very significant improvements in NDE techniques, and many in-service inspections (ISI) of nuclear pressure vessels have been performed. The current ISI methodology can effectively detect and size flaws that are much smaller than the two-inch reference flaw postulated in WRC-175. The Working Group on

Operating Plant Criteria of Section XI is revisiting the definition of the Appendix G reference flaw size. A task group was formed to consider the latest NDE capabilities with fracture mechanics analysts to formulate a set of recommendations leading to a reduction in the postulated Appendix G reference flaw size.

The first obstacle this group encountered is that if the flaw size ratio of 10 is applied to a reduced Appendix G flaw size, the acceptable flaw is proportionately reduced and becomes unreasonably small. This prompted a review of the basis document to determine the original reasoning for the establishment of the flaw size ratio of 10. As a result of this review, an oversight was found, which if rectified, greatly reduces the flaw size ratio. This is discussed below.

REVIEW OF BASIC PREMISE OF SECTION XI ACCEPTANCE STANDARDS

Principal Safety Criteria

In the development of the acceptance standings, the guiding principles and rationale were derived from the following stated principal safety criteria⁽³⁾:

- (a) "The safety margins with respect to the structural integrity of the components containing flaws within the limits of the 'allowable indication standards' should not reduce the margins applied in the design of the component as related to the material's ductile behavior under conditions of normal plant operation.
- (b) "The safety margins applied in determining the stress intensity factors of materials and welds containing flaws within the limits of the 'allowable indication standards' should be comparable to the margins of (a) above but related to the nonductile behavior and material fracture toughness requirements specified in

The 26th National Symposium on Fracture Mechanics

**ROUND COMPACT SPECIMEN TEST METHOD
FOR DETERMINING J-R CURVES
AND VALIDATION BY J-TESTS**

K. K. Yoon¹, L. B. Gross¹, C. S. Wade² and W. A. VanDerSluys²

ABSTRACT:

A fracture toughness test method using a standard round compact tension (RCT) specimen is presented. This procedure is completely analogous to ASTM E 1152-87 standard for determining J-R curves using rectangular compact tension specimens. A slightly different round compact tension specimen design (BWRCT) is used by B&W Owners Group in their Integrated Reactor Vessel Material Surveillance Program⁽¹⁾ (IRVSP). This specimen is analyzed by a finite element method to investigate whether the standard RCT compliance relationship is appropriate for use. Validation tests using both square C(T) and RCT specimens were performed and the resulting J-R curves are compared. It is concluded that using this procedure both square C(T) and BWRCT specimens yield similar J-R curves.

The ASTM standard test method for determining J-R curves was revised in 1987 and the new standard was issued as E 1152-87. This standard is for testing rectangular compact tension and bend specimens. Round compact tension (RCT) specimens have been used for determining J-R curves for many years. However, E 1152 does not include RCTs. The only reference in the ASTM standards relevant to RCT is a stress intensity factor equation found in ASTM E 399-83. Futato⁽²⁾ wrote a test procedure for RCTs in Babcock & Wilcox in 1984 based on the work of Newman⁽³⁾ and Underwood⁽⁴⁾. A validation test was conducted to

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demonstrate that RCT specimen testing produces closely comparable J-R curves to those from standard C(T) testing in 1993.

This paper presents (1) a test procedure for round compact tension specimen testing in the same format of E 1152-87, (2) a finite element analysis of a round compact specimen to determine compliance of a slightly different round compact tension specimen used in the B&W Owners Group IRVSP, and (3) the results of a validation testing to compare round C(T) with standard square C(T) specimens for identical weld metal and the data analysis by the proposed procedure and by current E-1152. The resulting J-R curves are compared.

REFERENCES

1. A. L. Lowe, Jr., K. E. Moore, and J. D. Aadland, "Integrated Reactor Vessel Materials Surveillance Program for Babcock & Wilcox 177-FA Plants," ASTM STP-870, American Society of Testing Materials, Philadelphia, PA, 1985, p.931-950.
2. R. J. Futato, "Round Compact Fracture Specimen Calibration," RDD:84:2839-01-01:01, The Babcock & Wilcox Company, Research & Development Division, Alliance, Ohio, March 13, 1984.
3. J. C. Newman, Jr., "Stress Intensity Factors and Crack Opening Displacements for Round Compact Specimens," International Journal of Fracture, Vol. 17, No. 6, December 1981, pp 567-578.
4. J. H. Underwood, "K and Displacement for Disc Shaped Specimen," Unpublished data, U.S. Army Armament RD&E Center, Watervliet, New York.

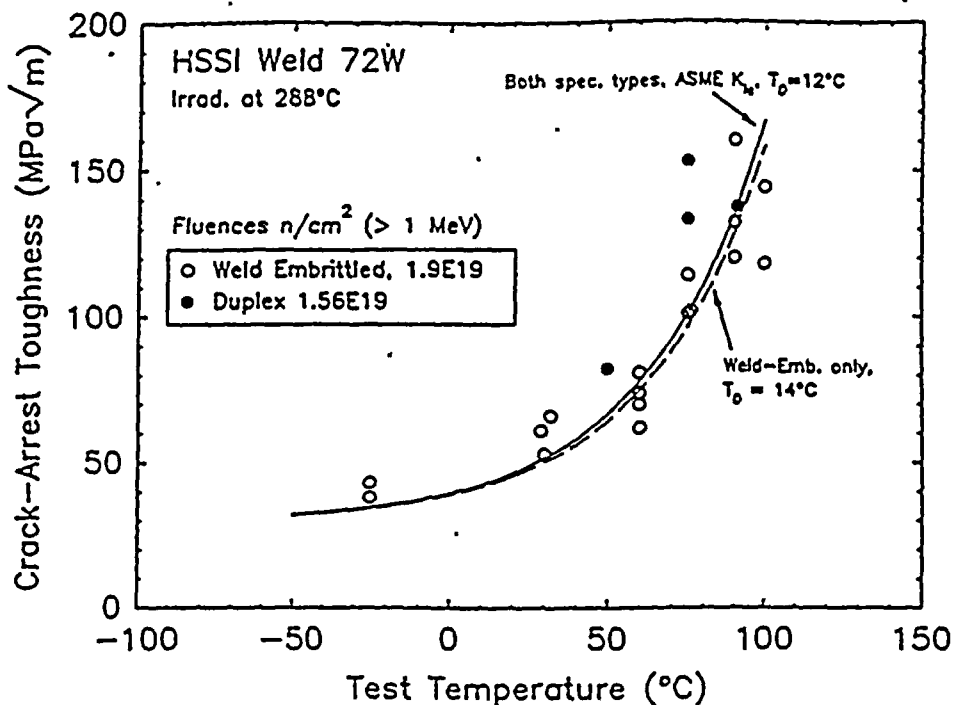


Figure 5. Crack-arrest toughness, K_{Ic} , for irradiated HSSI weld 72W showing the results of both weld-embrittled and duplex-type specimens.

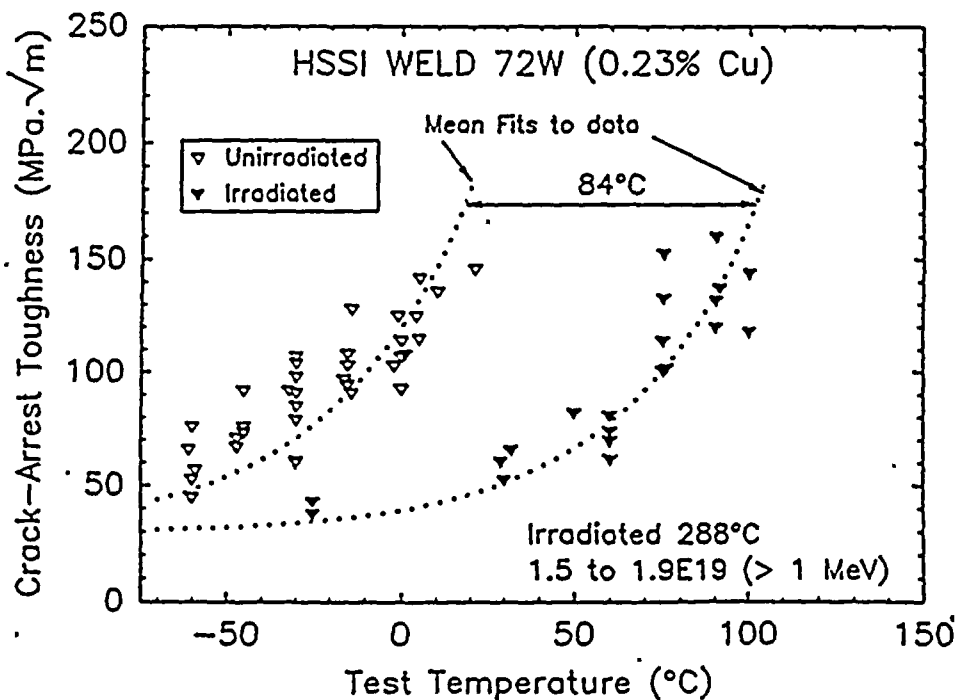


Figure 6. Crack arrest toughness values for both unirradiated and irradiated 72W weld metal and for weld-embrittled and duplex-type specimens.

HSSI WELD 72W DATA

IRRADIATED DATA RESULTS

@ Fluence Range	1.6 to 1.9E+19
Measured Shift	151F (84C)
RG199 Predicted Shift	192F to 199F
Additional Margin	41F to 48F

ADVANCED FRACTURE MECHANICS
METHODOLOGY

Biaxial

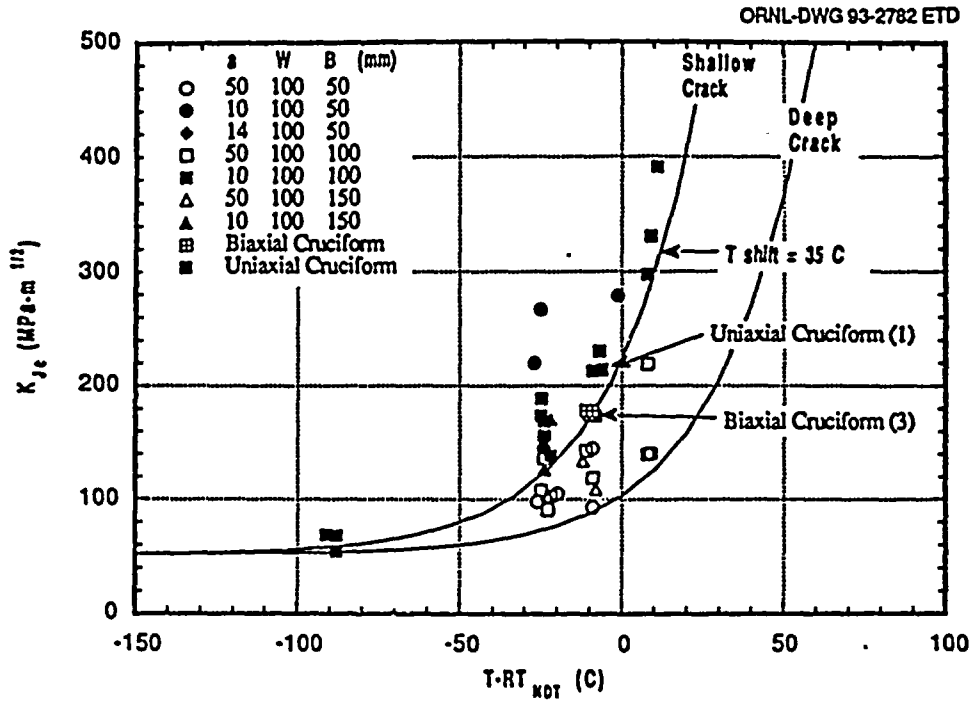


Figure 2.3 Biaxial and uniaxial shallow-crack toughness data as function of normalized temperature

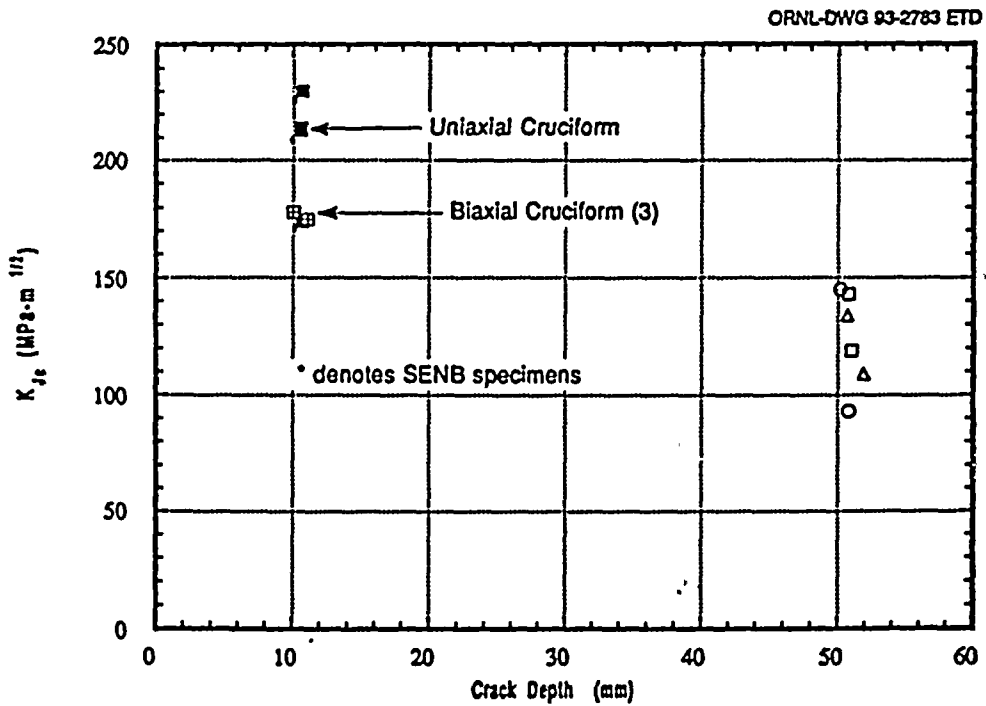


Figure 2.4 Uniaxial and biaxial toughness data as function of crack depth at $T - RT_{NDT} = -10^{\circ}C$

Constraint

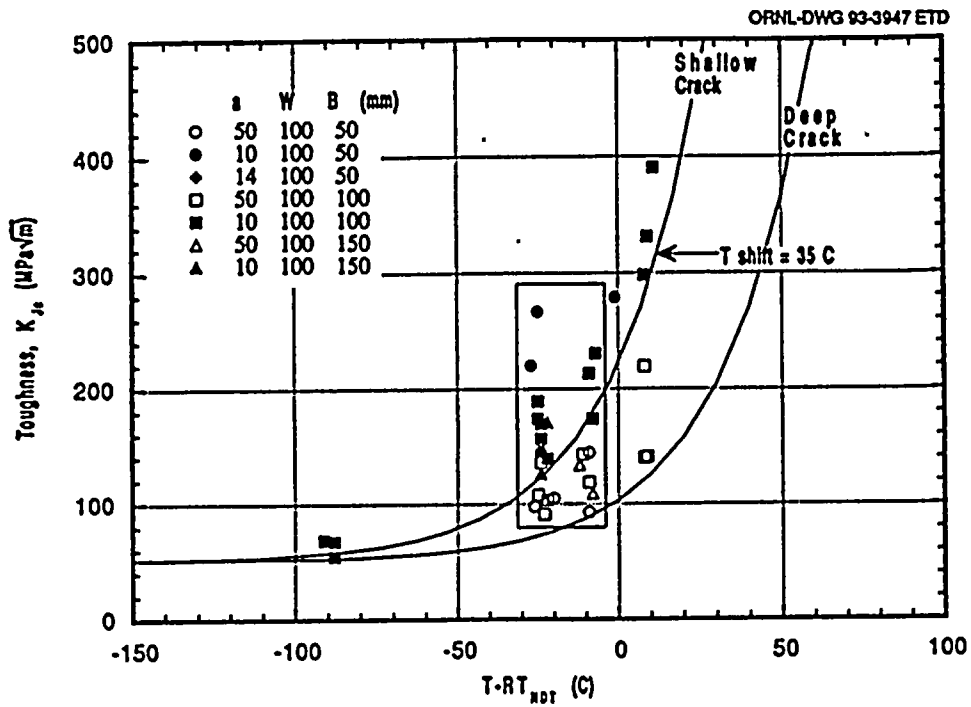


Figure 3.24 HSST shallow-crack fracture toughness results as function of normalized temperature $T - RT_{NDT}$

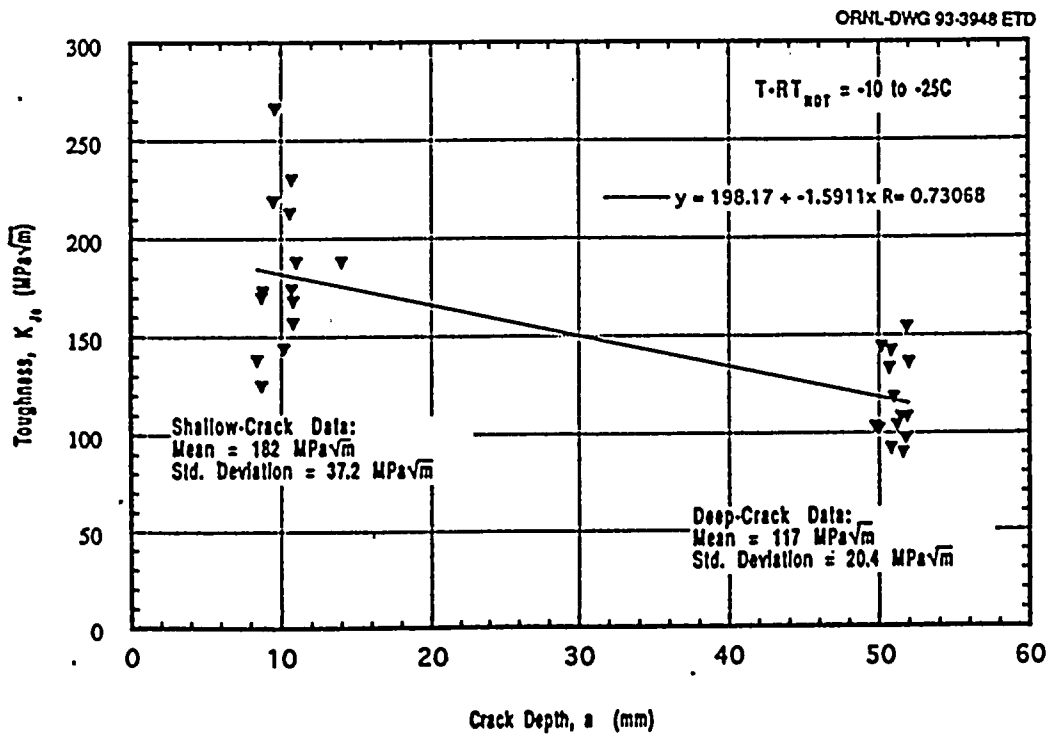


Figure 3.25 Toughness data at $T - RT_{NDT} = -25 \text{ to } -10^\circ\text{C}$ as function of crack depth

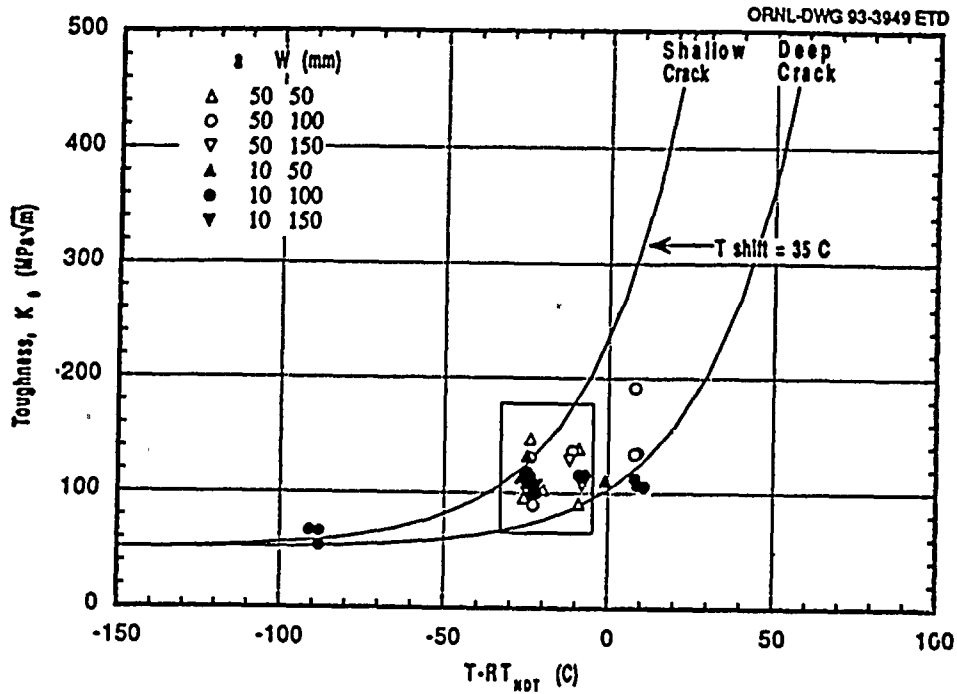


Figure 3.26 SSY toughness (K_0) results as function of normalized temperature

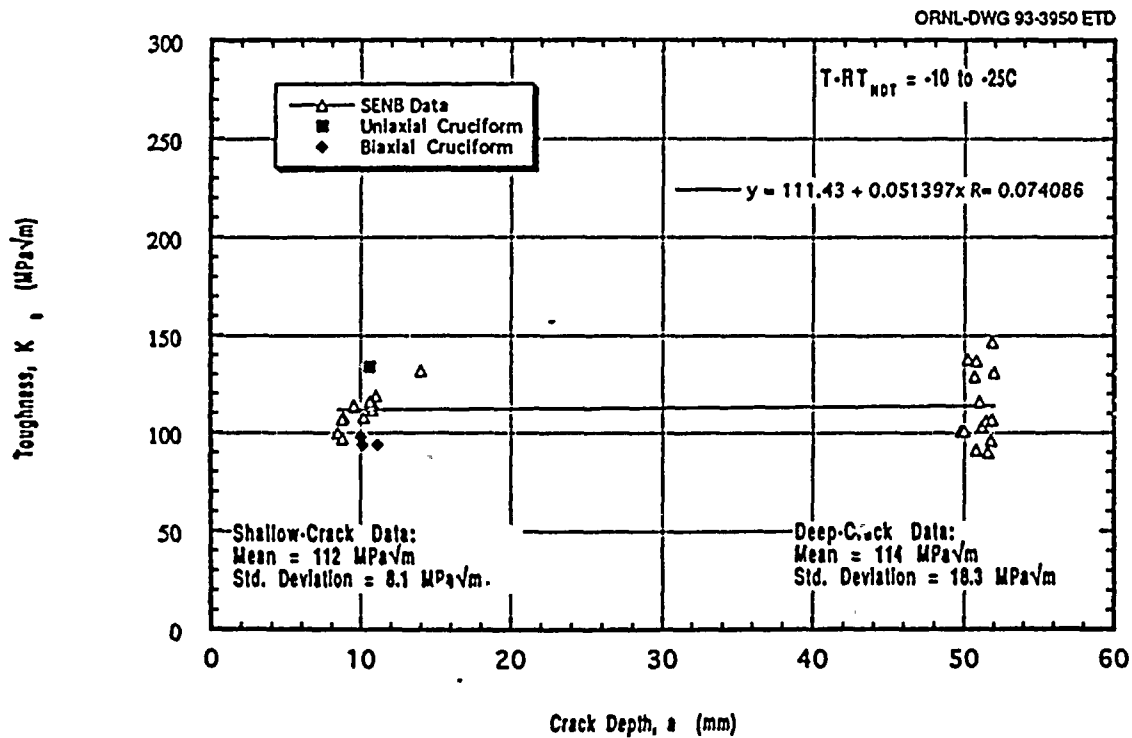


Figure 3.27 K_0 data at $T - RT_{\text{NDT}} = -25 \text{ to } -10^\circ\text{C}$ as function of crack depth

RVWG SHALLOW CRACK DATA ANALYSIS

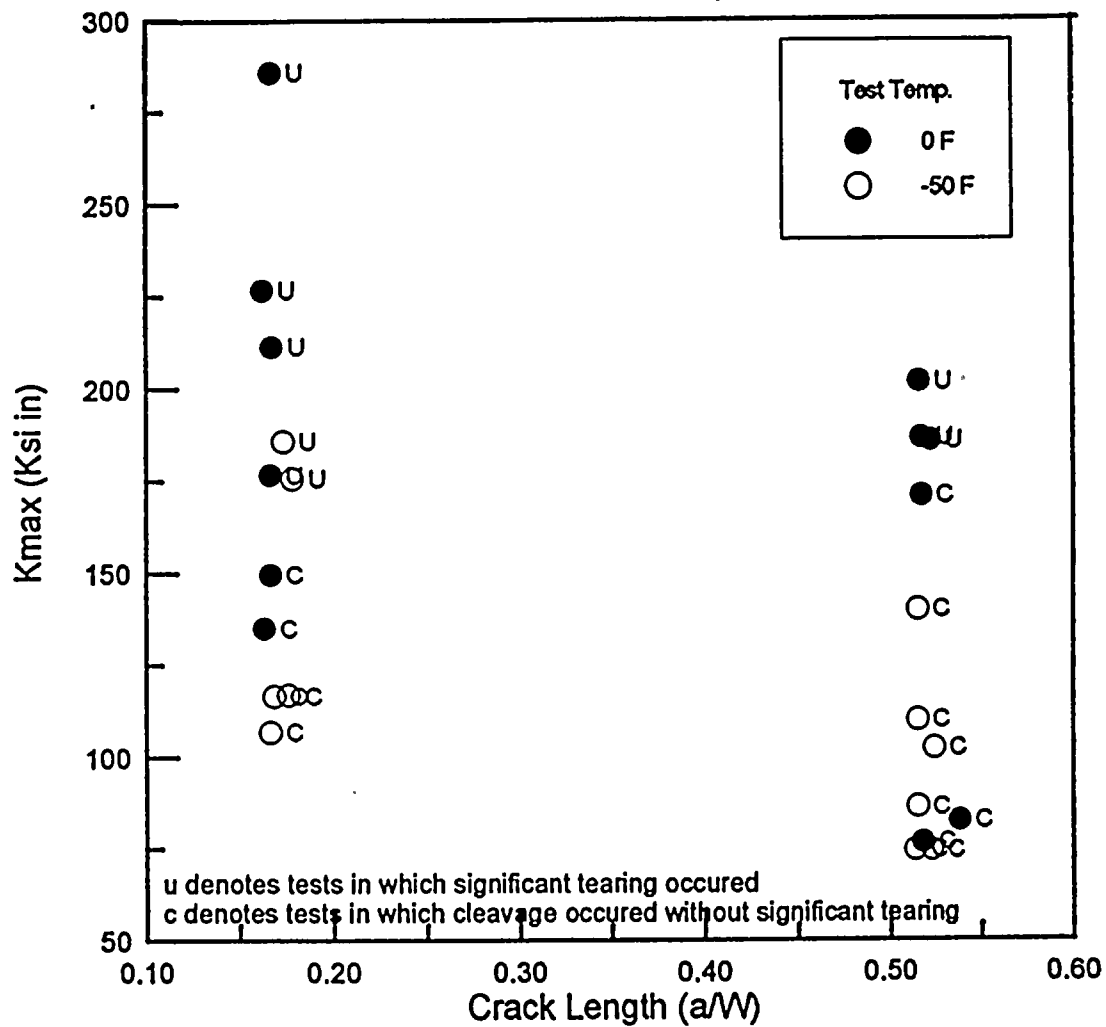
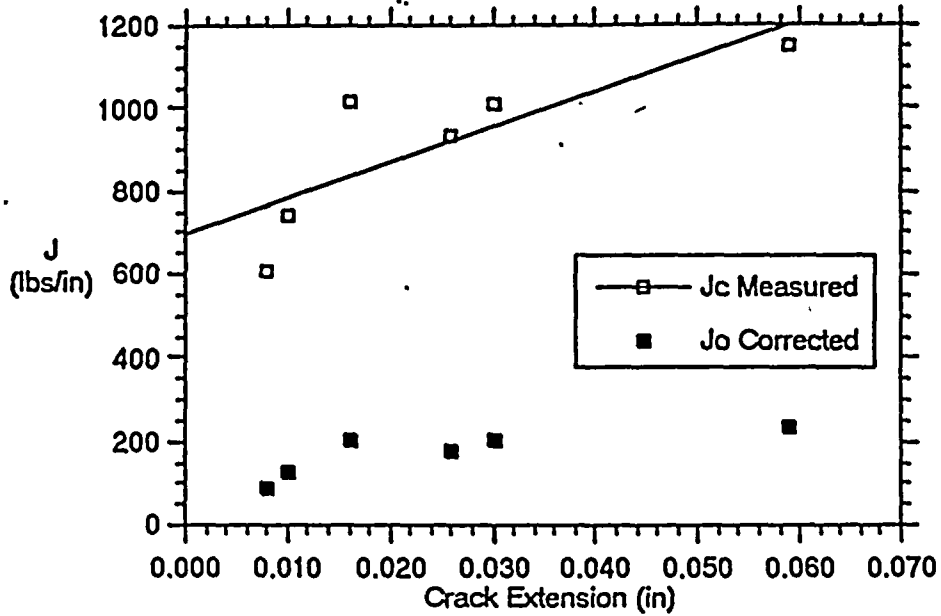
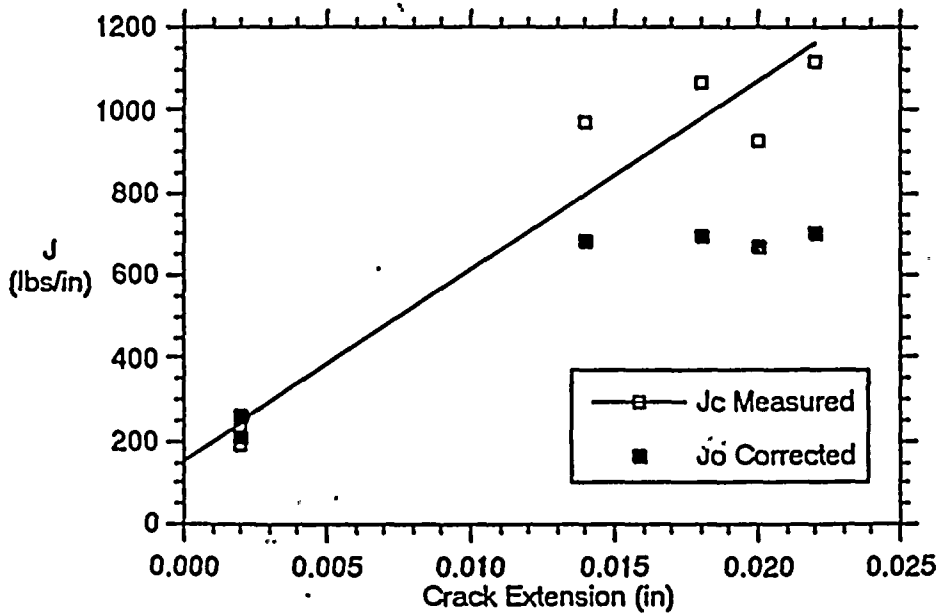


Fig 6 Values of K_{max} as a function of temperature and crack length

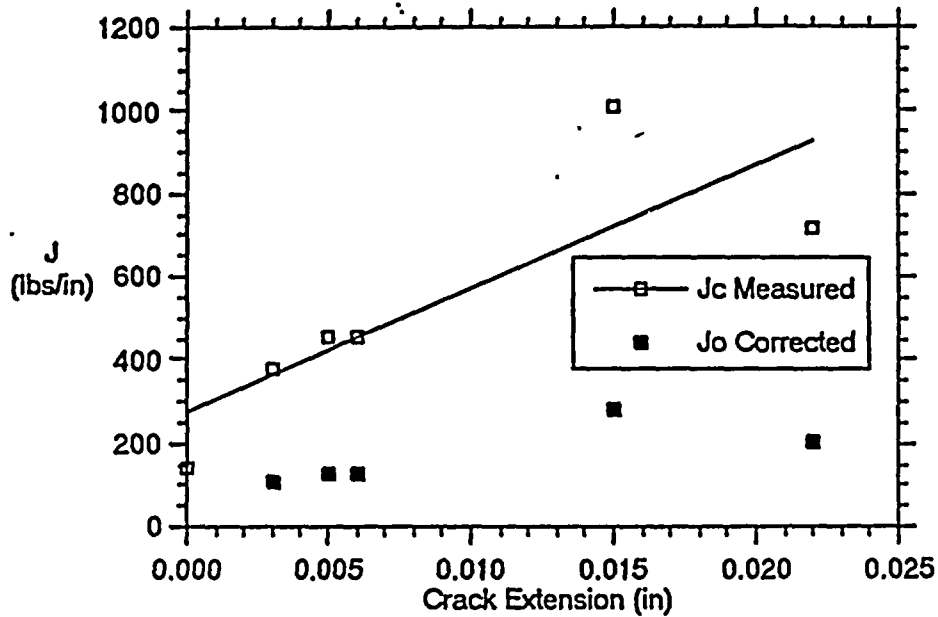
WF-292 (0 deg F), B&W R&D Division
 SE(B), a/W=0.17, W=2 in., B=1in.
 $T_i = 281$, $n \sim 10$, $J_{ic} \sim 694$ lbs/in, $E / s_{yd} = 432$



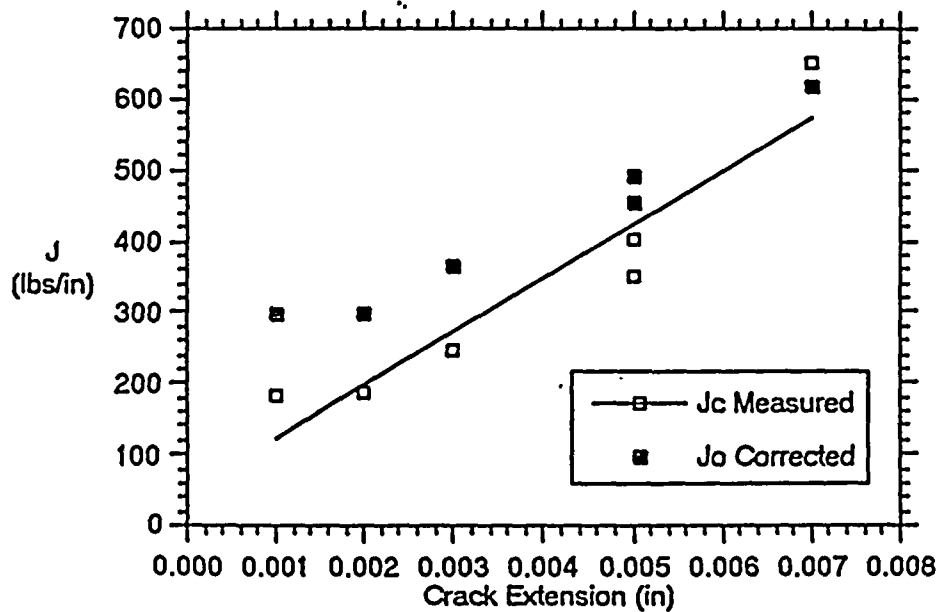
WF-292 (0 deg F), B&W R&D Division
 SE(B), a/W=0.52, W=2 in., B=1in.
 $T_i = 281$, $n \sim 10$, $J_{ic} \sim 155.93$ lbs/in, $E / s_{yd} = 432$



WF-292 (-50 deg F), B&W R&D Division
 SE(B), a/W=0.17, W=2 in., B=1in.
 $T_i = 311$, $n \sim 10$, $J_{ic} \sim 273$ lbs/in, $E / s_{yd} = 435$



WF-292 (-50 deg F), B&W R&D Division
 SE(B), a/W=0.52, W=2 in., B=1in.
 $T_i = 187$, $n \sim 10$, $J_{ic} \sim 47$ lbs/in, $E / s_{yd} = 435$



The 26th National Symposium on Fracture Mechanics

SHALLOW CRACK TESTING OF Mn-Mo-Ni/LINDE 80 WELD METALS

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Lynchburg, Virginia

W. A. VanDerSluys
C. S. Wade
Babcock & Wilcox
Alliance Research Center
Alliance, Ohio

Apparent enhancement of fracture toughness due to constraint effects in test specimens with shallow crack depths was reported in the literature on A36, A533 and other materials, compared to the traditional deep crack specimen test data. None of the materials tested is weld metal.

The constraint effect on the crack depth is primarily a geometric parameter. However, it may depend on the crack tip plastic zone shapes and stress field changes that may show sensitivity to material differences. A set of three point bend specimens was fabricated from a Mn-Mo-Ni/Linde 80 weld metal and tested at two different temperatures in transition temperature range i.e. 0 and -50 degrees F, to establish the shallow crack effect on weld metals of significance to the commercial power reactor vessel integrity issue.

In addition, current Dodds and Anderson constraint correction methods, both 2D with crack growth and 3D, were applied and the results are compared with the uncorrected data. The significance of this material data information on nuclear power plant safety evaluations under pressurized thermal shock transients is discussed.

Application of Two-Parameter (J-Q) Fracture Mechanics to Reactor Pressure Vessels Subjected to Pressurized Thermal Shocks

Objective

The long term primary objective of these analyses is to evaluate crack-tip stress fields in reactor pressure vessels (RPV) throughout a pressurized thermal shock (PTS) transient using the two-parameter J-Q fracture mechanics approach and incorporate small-specimen fracture toughness data in fracture mechanics assessments of RPVs.

Steps for incorporation of this technology into a RPV assessment approach

- o Development of boundary layer small scale yielding finite element methodology.
- o Development of full body finite element models of RPVs containing circumferentially and axially oriented cracks.
- o Perform PTS transient using the full body finite element models so that the applied Q stress can be evaluated using J-Q theory.
- o Apply the Dodds-Anderson Scaling Model to experimental fracture toughness data to determine $J_c(Q, \text{Temperature})$.

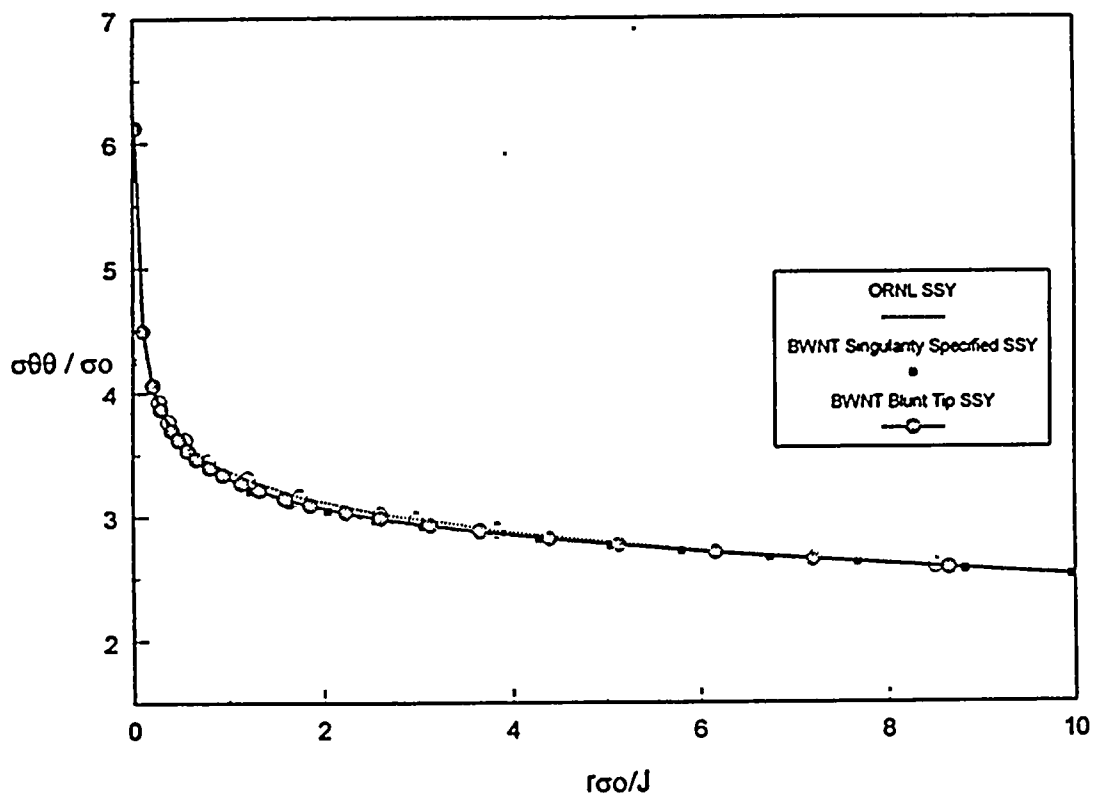
Application of Two-Parameter (J-Q) Fracture Mechanics to Reactor Pressure Vessels Subjected to Pressurized Thermal Shocks (Continued)

Step 1: Development of Boundary Layer Small Scale Yielding Finite Element Methodology

- o A boundary layer approach using the finite element code ABAQUS, assuming a rate-independent, J_2 (isotropic-hardening) incremental plasticity theory, is adopted in evaluating the reference small-strain SSY crack tip fields in this study.
- o The basis for the development of the BLM SSY approach is based on NUREG/CR-6132 which allows for direct comparison and validation of the BWNT approach.
- o Two BWNT FEA models were created.
 - BWNT Blunt Crack Tip
 - Incorporates a crack-tip region with an initial root radius at the tip of 10^{-6} times the outer radius of the mesh
 - BWNT Point Crack Tip
 - Imposes an elastic-plastic singularity at the crack tip allowing for less computational time with only small variations in stresses from the more refined mesh.
- o The plane strain reference fields for the two BWNT models was compared with the ORNL NUREG/CR-6132 model determined from the boundary layer model are shown in the figure on the next page. As can be seen, excellent correlation can be seen for all three models as all solutions are within 3% of one another. From a computational point-of-view, it is recommended that the BWNT elastic-plastic singularity specified finite element model be utilized in future investigations as the solutions are extremely accurate with the least computational time.

Application of Two-Parameter (J-Q) Fracture Mechanics to Reactor Pressure Vessels Subjected to Pressurized Thermal Shocks (Continued)

Step 1: Development of Boundary Layer Small Scale Yielding Finite Element Methodology (Continued)

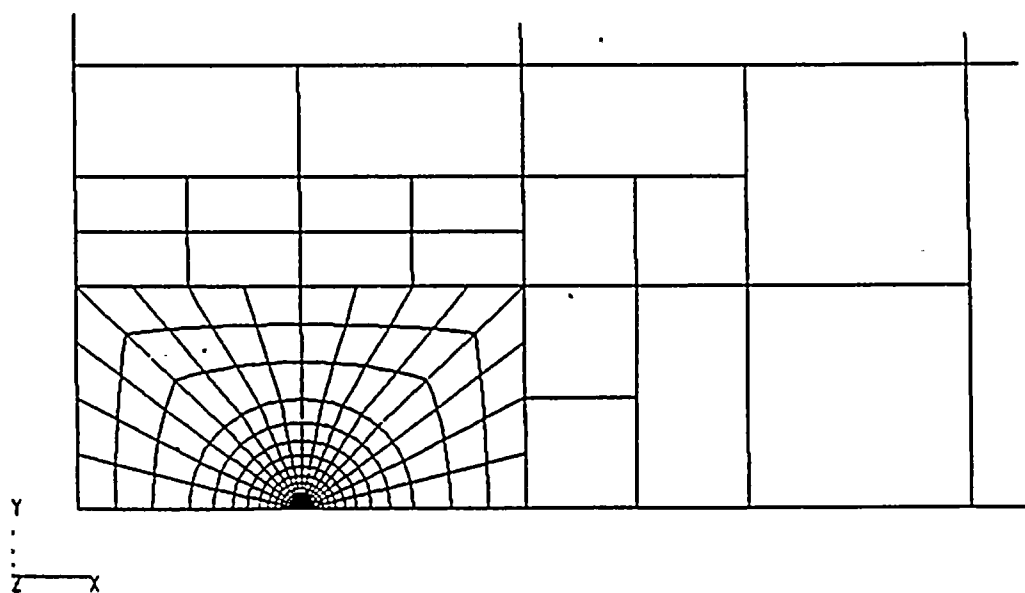
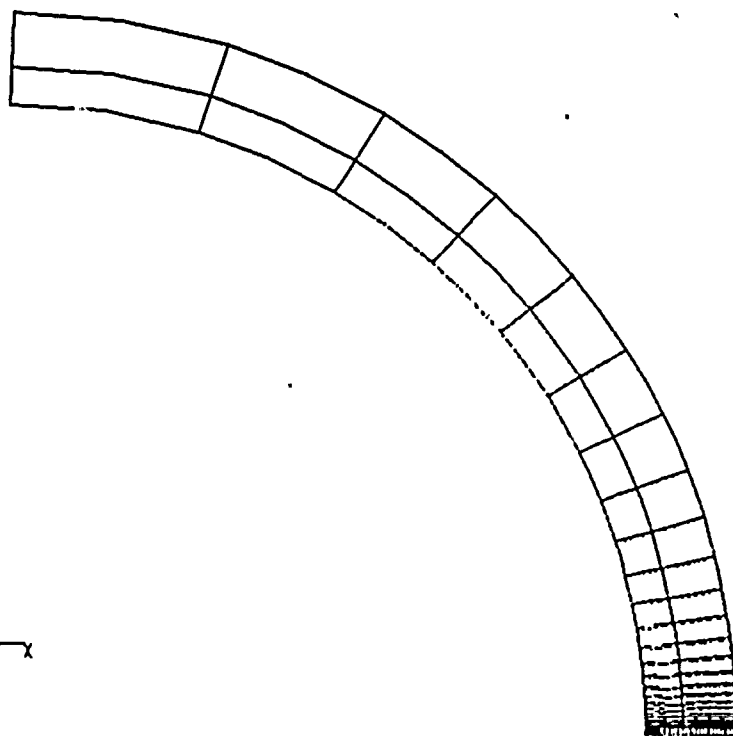


Comparison of boundary layer small scale yielding finite element analysis solutions between ORNL NUREG/CR-6132 and BWNT for A 533 B Steel at $T=46^\circ \text{C}$

Application of Two-Parameter (J-Q) Fracture Mechanics to Reactor Pressure Vessels Subjected to Pressurized Thermal Shocks (Continued)

Step 2: Development of Full Body Finite Element Models of RPVs Containing Circumferentially and Axially Oriented Cracks

FEA Mesh for Axially Oriented Flaw in RPV

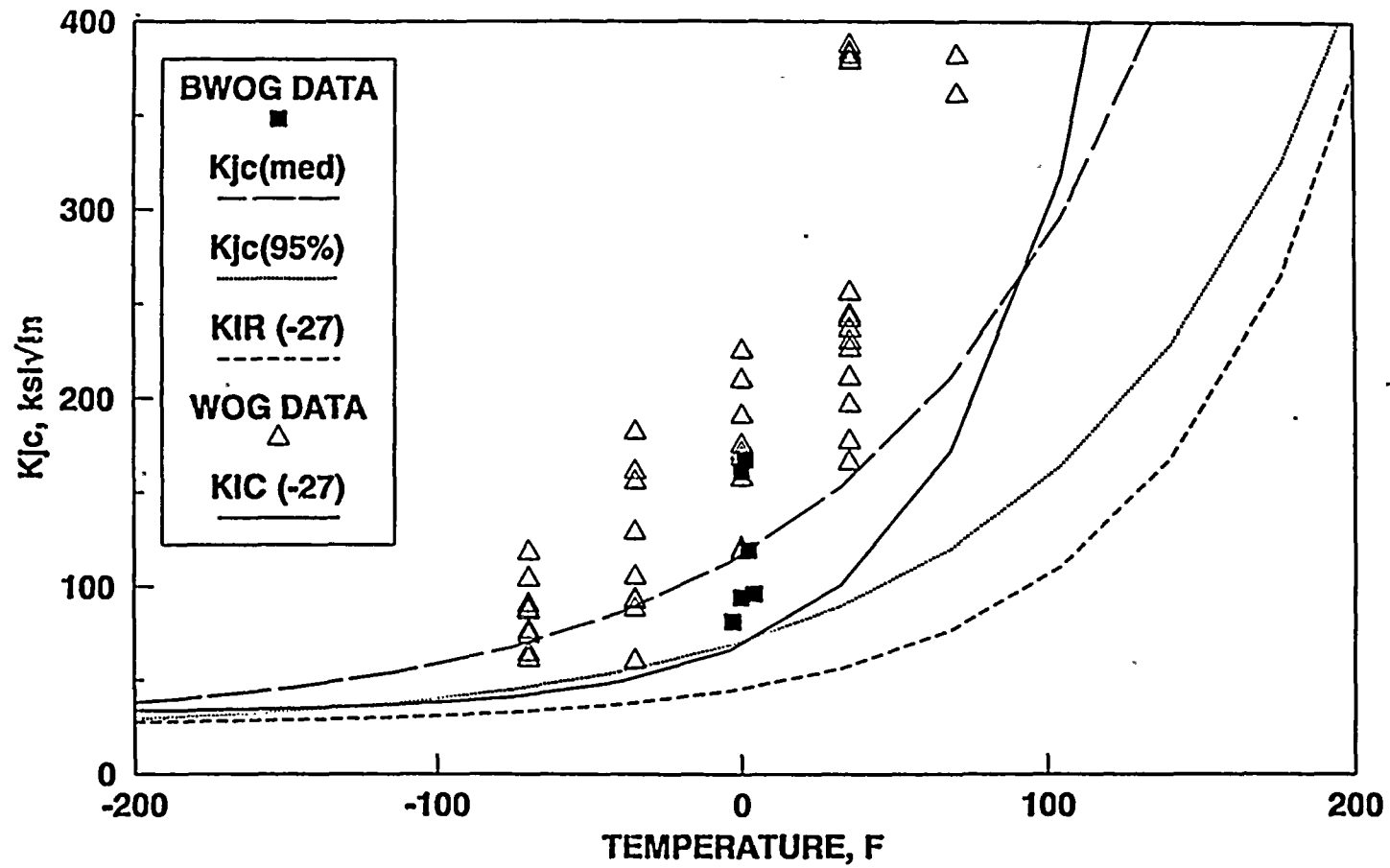


**ALTERNATIVE METHOD FOR
RADIATION EMBRITTLEMENT INDEXING**

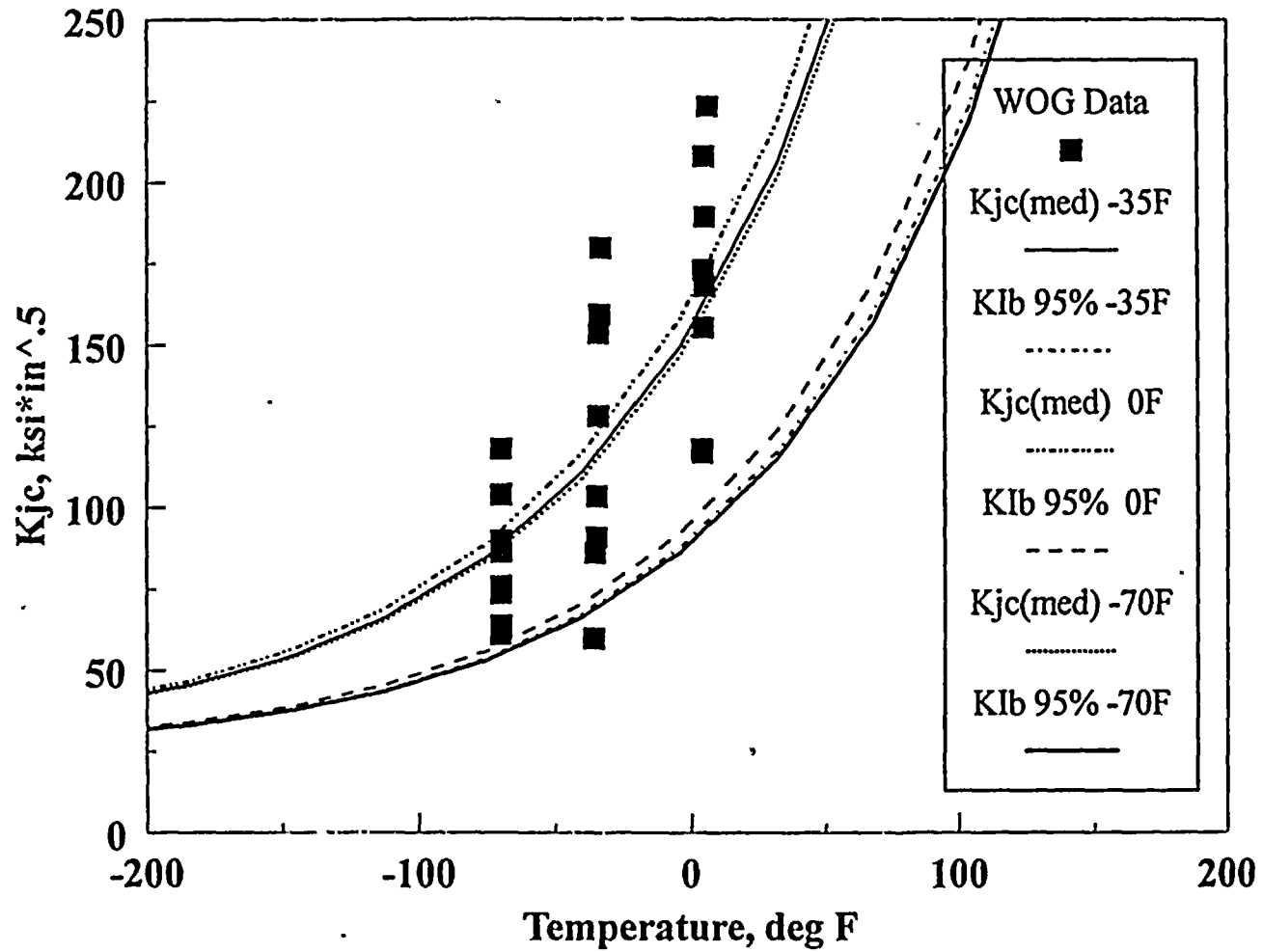
1. **BAW-2202 WF-70 Submittal - NRC Approval Received**
2. **Additional Linde 80 Weld Metal Qualification**
3. **A Topical Report for All Linde 80 Weld Metals**
4. **Development of Direct Indexing Method to Model Radiation Embrittlement**
5. **Development of Charpy Size Specimen Test Methods**
 - Dynamic versus Static Tests**
 - Effect of Test Temperature**
 - ASTM Standardization Activity**

Figure 4-7. Fracture Toughness Curves and WOG Data

FRACTURE TOUGHNESS - WF-70 STATIC DATA



WF-70 Fracture Toughness



SUMMARY

- **BWOG is actively participating in the methodology updates in Appendix G of Section XI of ASME Code and the PTS analysis methods.**
- **The NRC sponsored advanced FM methodology developments have very promising prospect of alleviating the PTS concern. BWOG is trying to develop application methodologies using these new concepts.**
- **BWOG anticipates great improvement in radiation embrittlement modeling by taking direct fracture toughness measurement approach. Testing irradiated Charpy specimens in surveillance programs may provide direct fracture toughness which in turn generates fluence specific fracture toughness curves. This approach will eliminate the indirect method of using Charpy impact energy data.**

BOUNDING EMBRITTLEMENT
TREND CURVES FOR
LINDE 80 CLASS OF WELDS

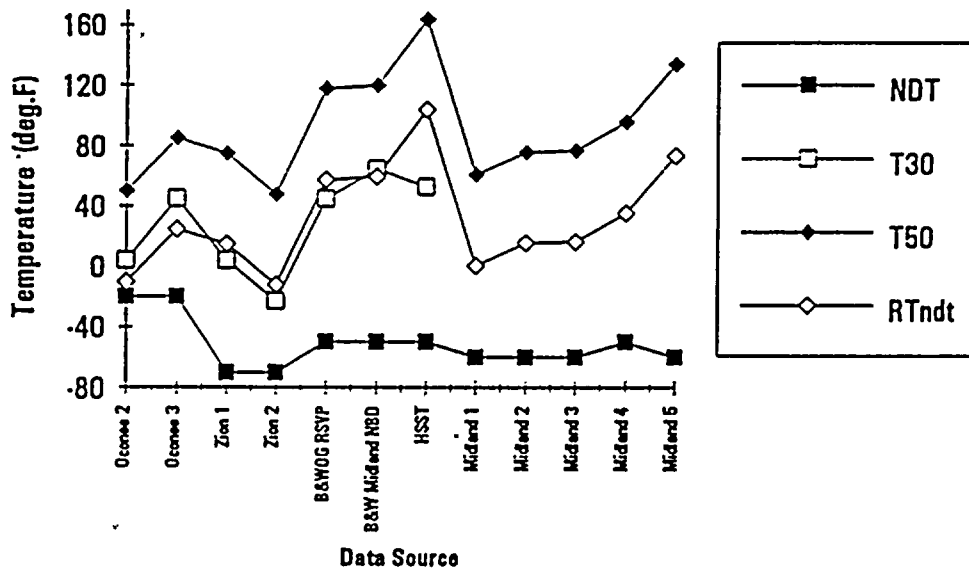
Objective:

Provide fixity for evaluating the embrittlement status RTndt & RTpts of the B&WOG reactor vessels made of Linde-80 welds.

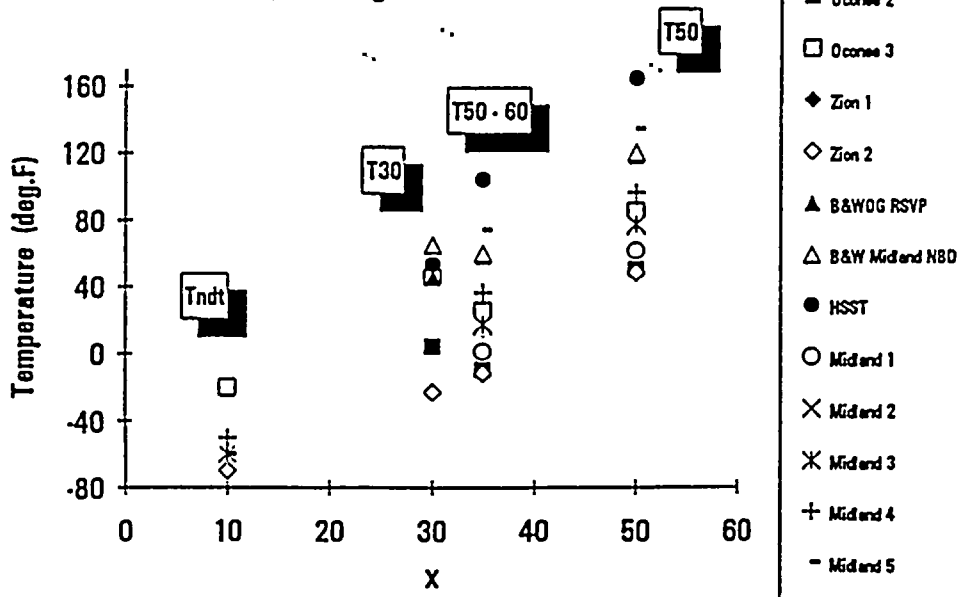
Method:

- 1 - Demonstrate the appropriateness of an Un-Irradiated RTndt of -10 deg.F with a concurrent sigma I of zero (0) for all Linde-80 welds.
- 2 - Demonstrate that use of the R.G. 1.99 R-2 chemistry factors, based on the mean of the chemistry, with a sigma shift of 14 deg.F when credible surveillance data exists and a sigma shift of 25 deg.F when no surveillance data exists, in accordance with the R.G. will yield bounding RTndt & RTpts trend curves by which embrittlement status can be assessed.

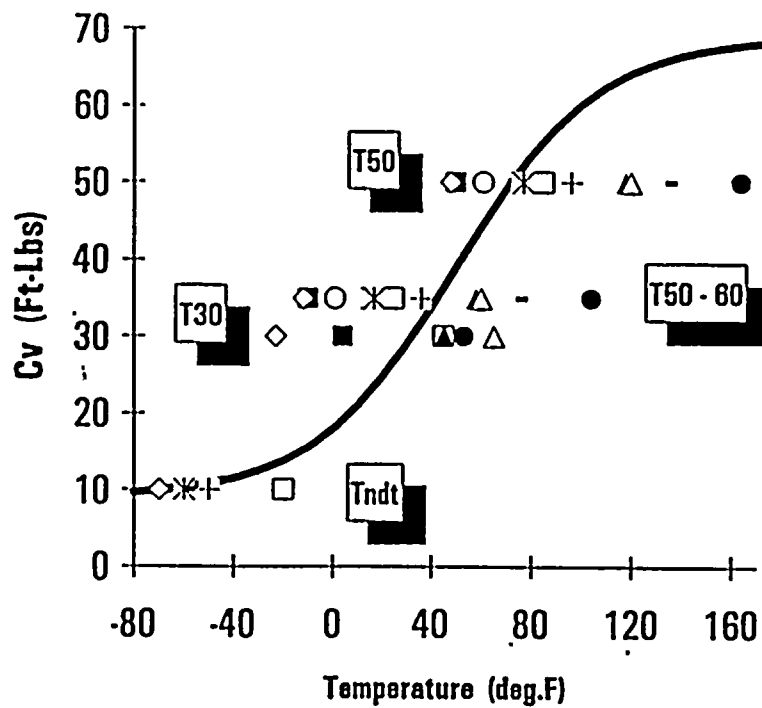
**B&WOG Data for WF-70 & WF-209 Welds
(including Midland Beltline)**



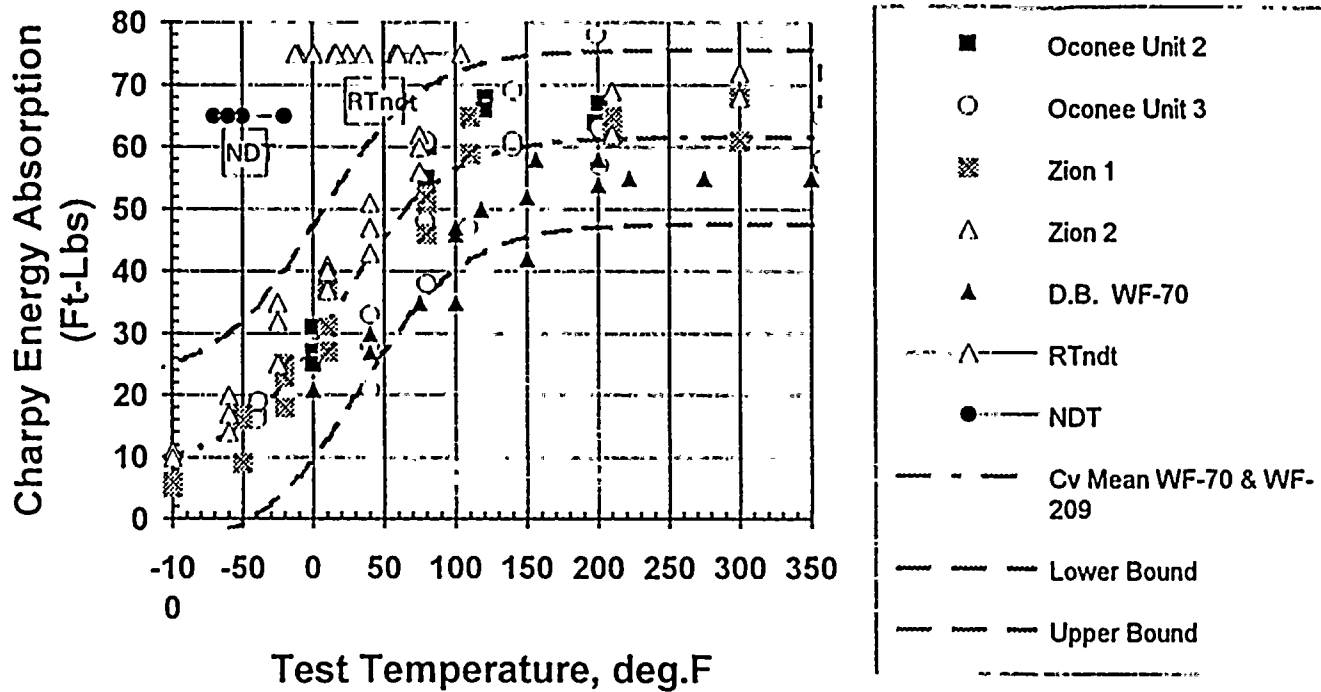
**B&WOG Data for WF-70 & WF-209 Welds
(including Midland Beltline)**



**B&WOG Data for WF-70 & WF-209 Welds
(including Midland Beltline)**

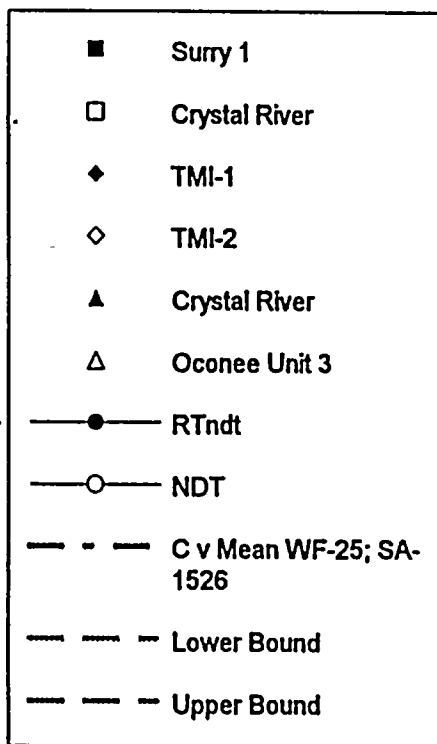
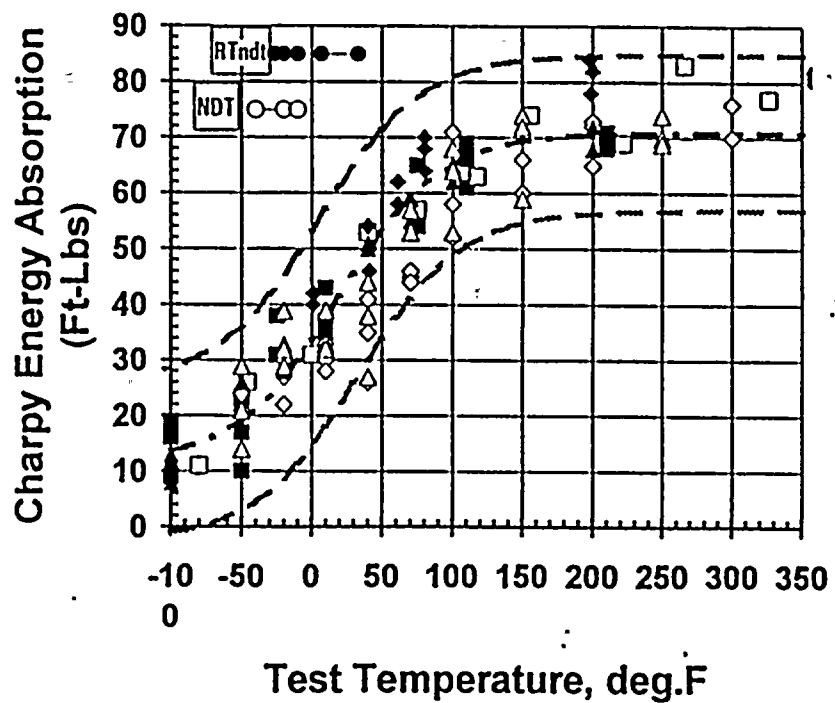


B&WOG WF-70 & WF-209 Un-Irradiated Capsules

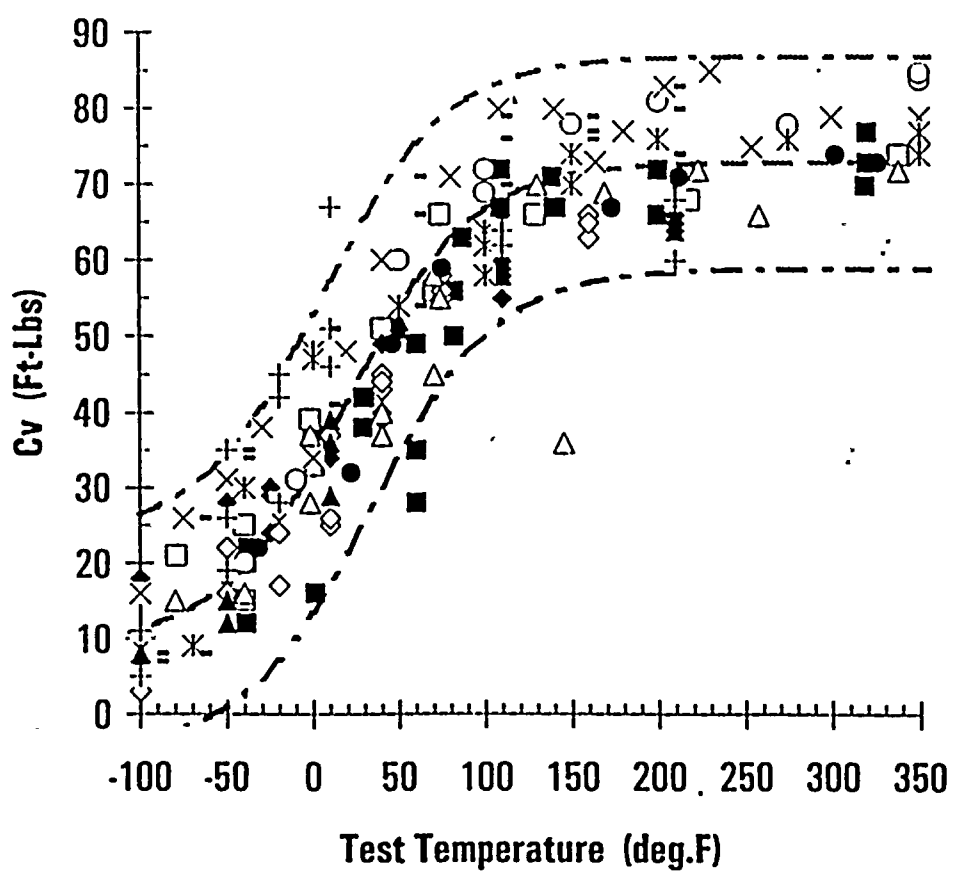




B&WOG WF-25 & SA-1526 Un-Irradiated Capsules

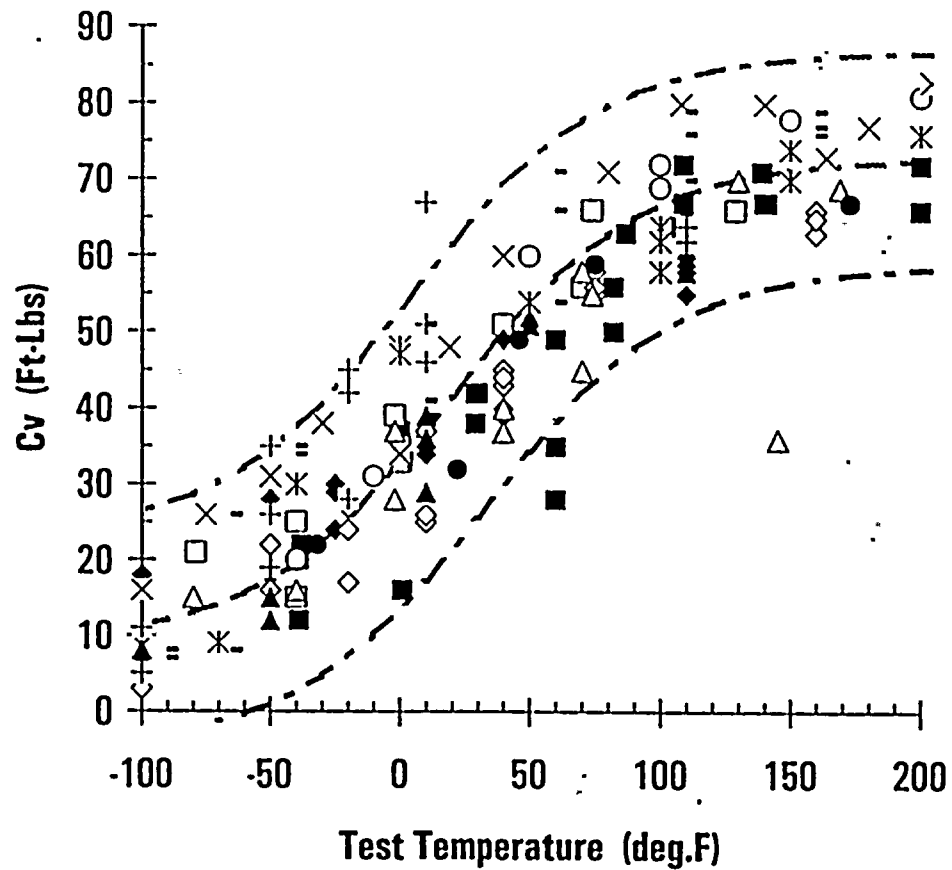


B&WOG Un-Irradiated Charpy Data
 (excluding WF-70, WF-209, WF-25 & SA1526)



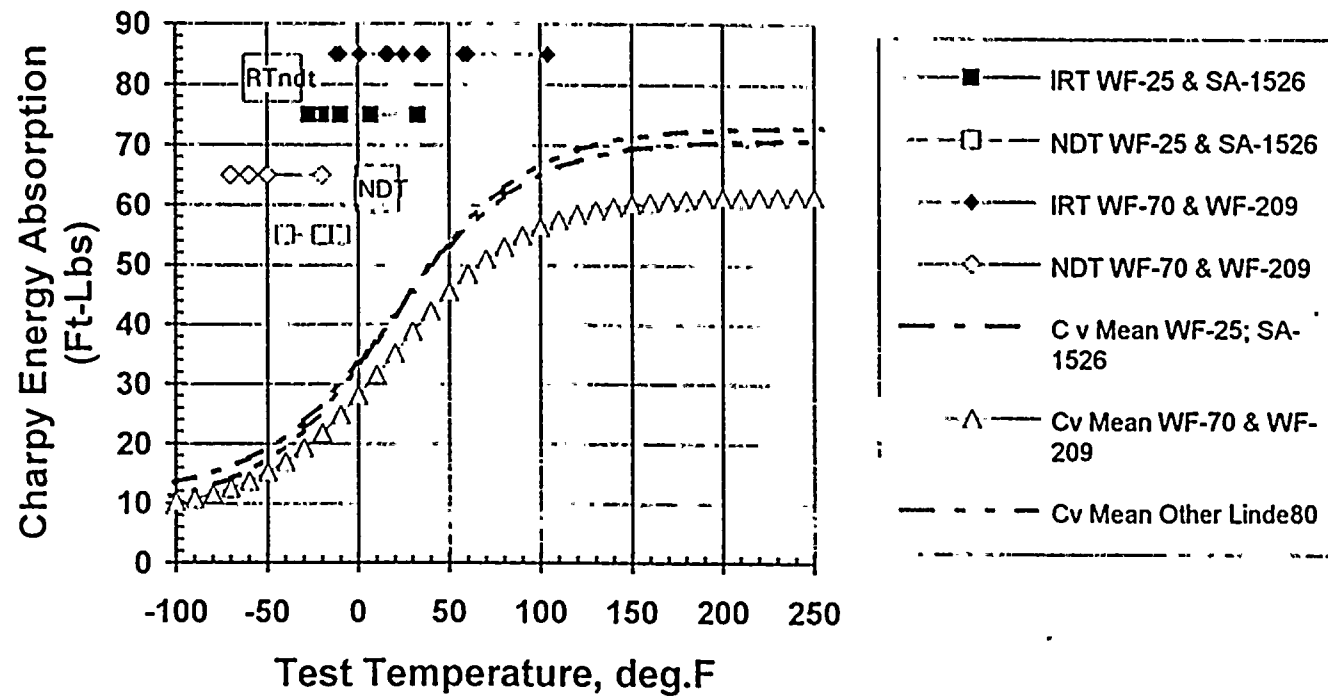
■	ANO-1 WF-193
□	R.S. WF-193
◆	PB-2 WF-193
◇	TP-3 SA-1101
▲	TP-4 SA-1084
△	D.B. WF-182
●	CR-3 WF-87
○	CR-3 SA-1585
×	D.B. WF-112
✱	D.B. SA-1135
+	PB-2 SA-1283
-	Giuga SA-1036
---	Mean All Cv
- - -	Upper Bound
- . -	Lower Bound

**B&WOG Un-Irradiated Charpy Data
(excluding WF-70, WF-209, WF-25 & SA1526)**



■	ANO-1 WF-193
□	R.S. WF-193
◆	PB-2 WF-193
◇	TP-3 SA-1101
▲	TP-4 SA-1094
△	D.B. WF-182
●	CR-3 WF-87
○	CR-3 SA-1585
×	D.B. WF-112
*	D.B. SA-1135
+	PB-2 SA-1283
-	Ginna SA-1038
— — —	Mean All Cv
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- - -	Lower Bound

B&WOG Un-Irradiated Linde-80 Welds Charpy Curve Comparisons

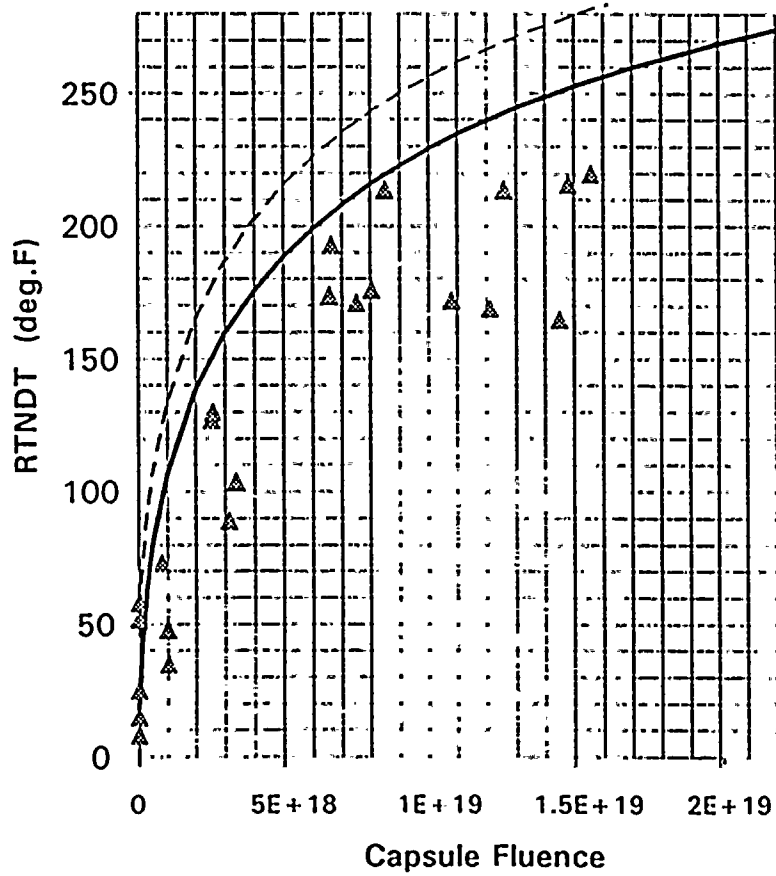


Results:

Based on a review of the Un-Irradiated Charpy and NDT data for the Linde-80 welds it is clear that the Charpy transition region for these welds will always be above -10 deg.F. Thus, use of an upper bound un-irradiated RTndt of -10 deg.F is appropriate for the Linde-80 welds.

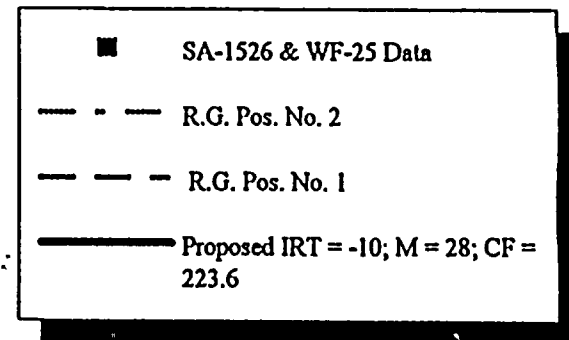
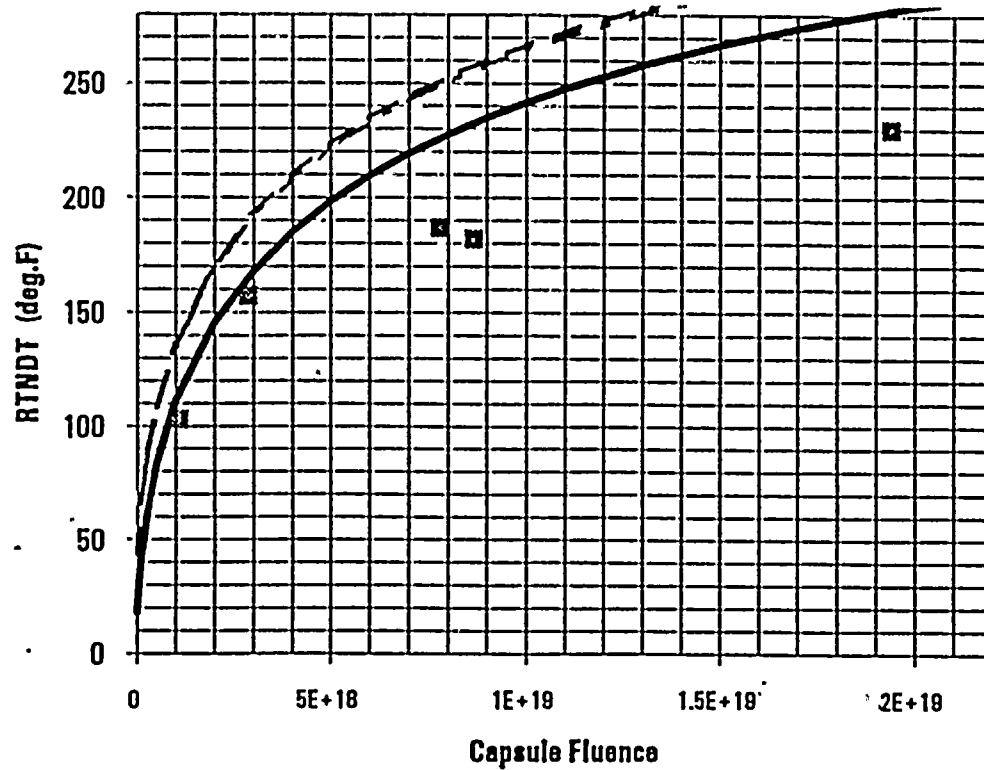


WF-70 & WF-209 Surveillance Data



- ▲ WF-70 & 209 RTndt (50Fl.bs) Data
- Proposed IRT = 10; M = 28; CF = 211
- - - R.G. Pos. No1 IRT = 5; M = 50; CF = 211

SA-1526 & WF-25 Surveillance Data



Conclusion:

Based on a review and comparison of the proposed RTndt bounding trend curves to the surveillance capsule RTndt it is apparent that the proposed method will yield conservatively bounding embrittlement (RTndt & RTpts) trends by which the status of these reactor vessels can be assessed.

**MICROSTRUCTURAL
STUDIES**

OBJECTIVES

Provide Physical Evidence To Support
Observed Mechanical Test Data Trends.

- Saturation/Stabilization
- Effective Copper
- Flux Effects
- Irradiation Temperature Effects

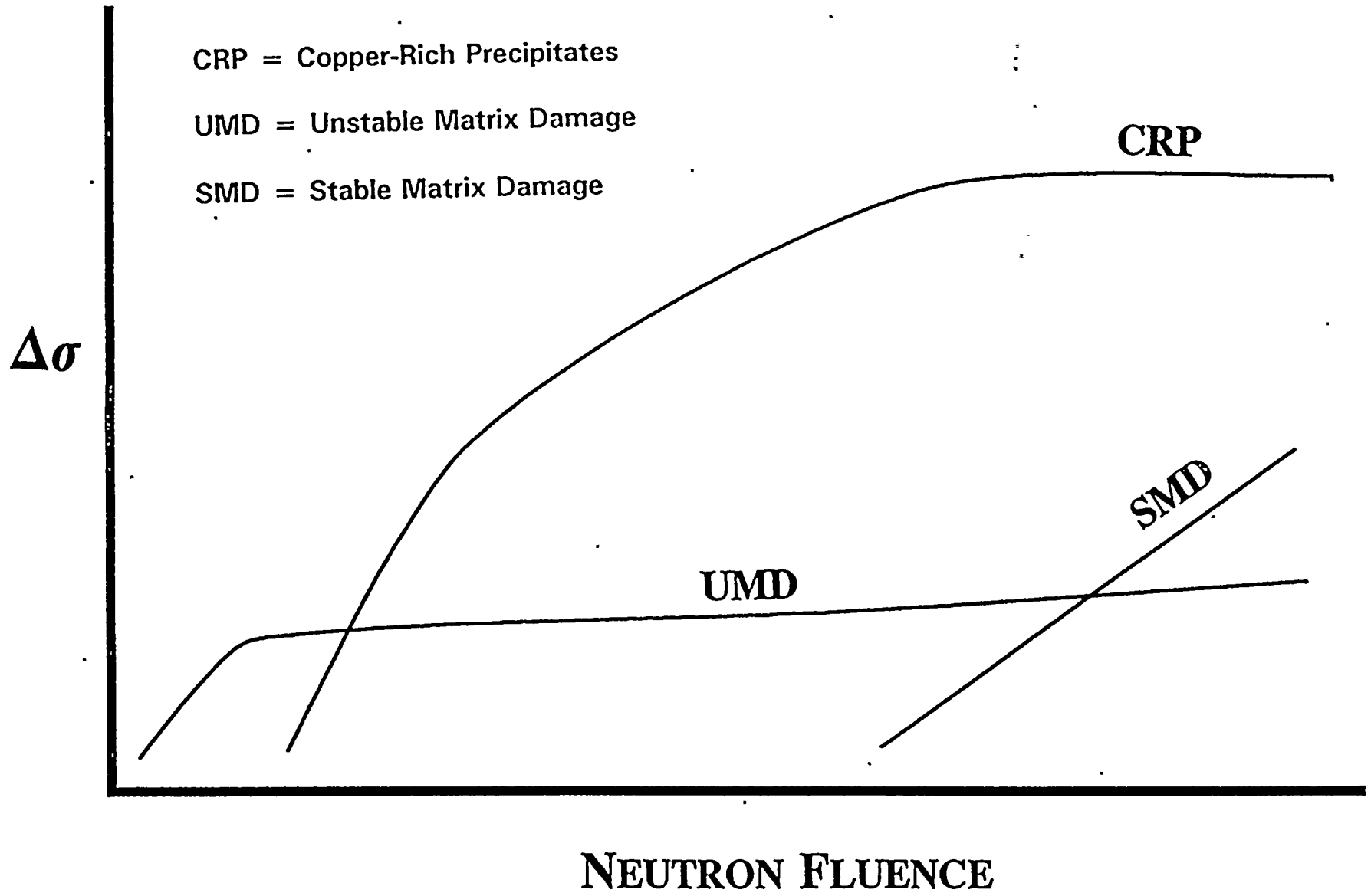
Provides Credibility To Correlations

MICROSTRUCTURAL CHARACTERIZATION

- Material Selected
- Samples Have Been Prepared
- Test Laboratories Selected
 - APFIM - Matrix Cu Content/Precipitate Composition
 - SANS - Precipitate Size and Spacing
 - Dislocation Density

MATERIAL / IRRADIATED CONDITIONS / CU

WF 209-1 Unirradiated	2.5E18	1.3E19	1.6E19	0.35
WF 182-1 Unirradiated	5.9E18	9.6E18	1.3E19	0.24
WF 447 Unirradiated	4.0E18	1.7E19		0.03

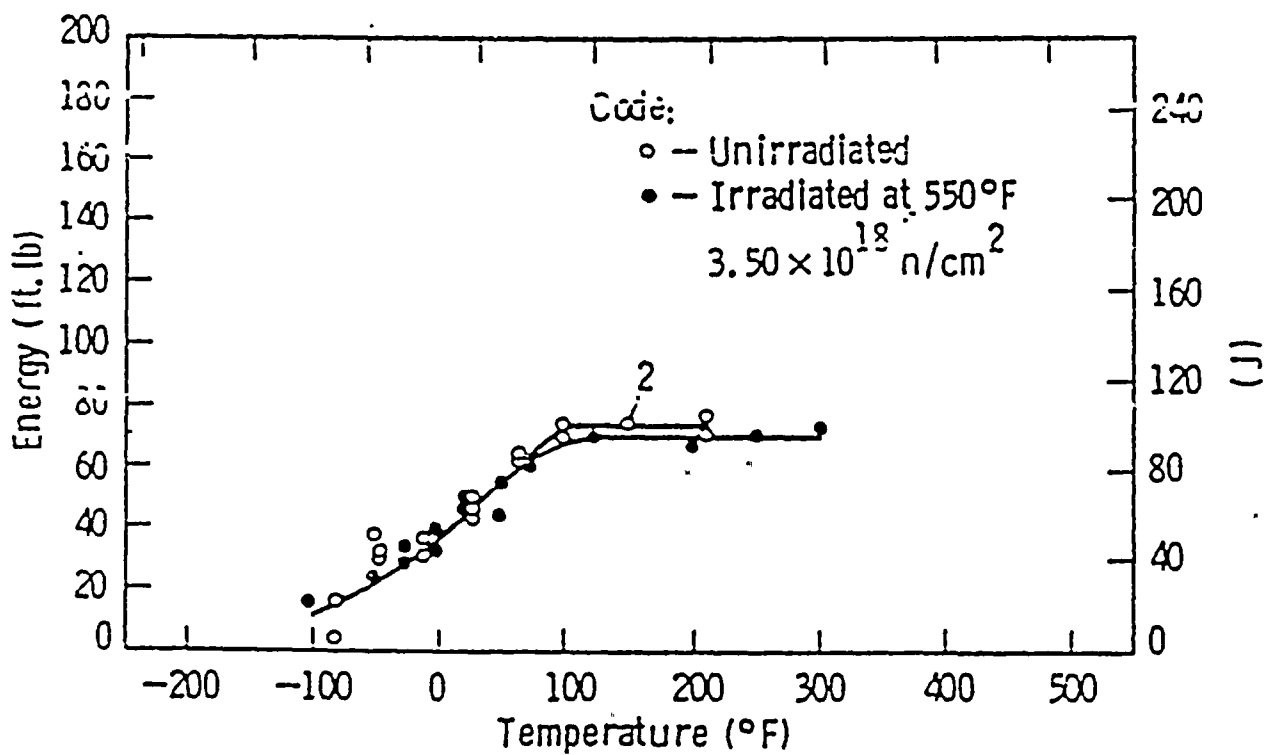
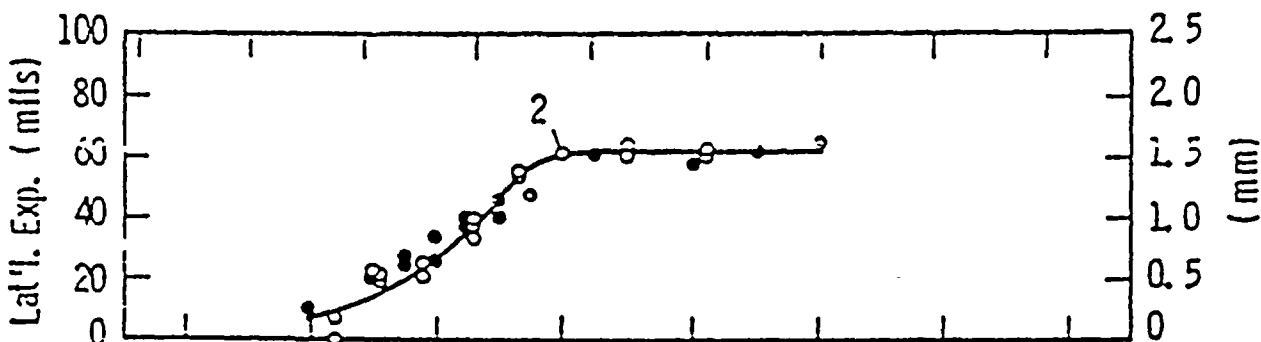
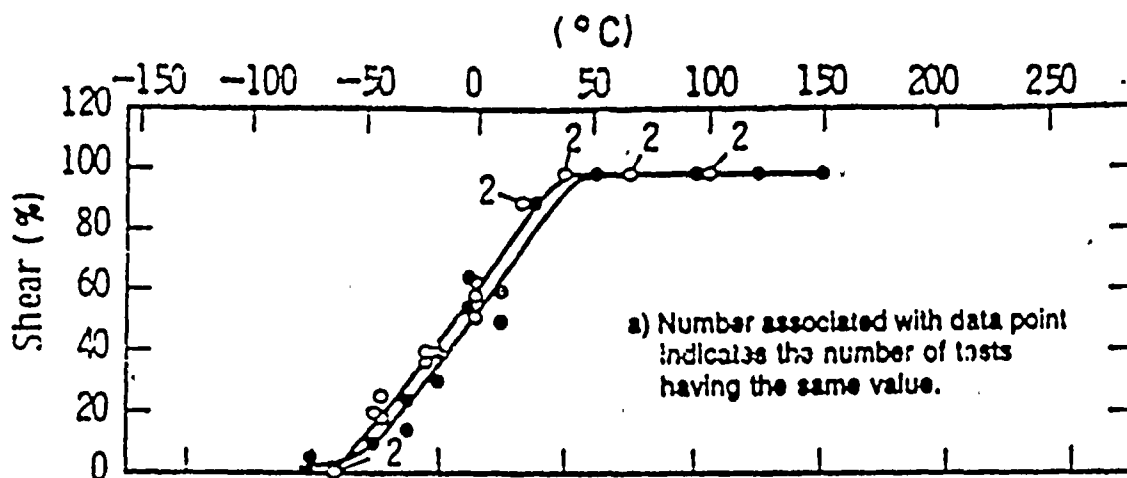


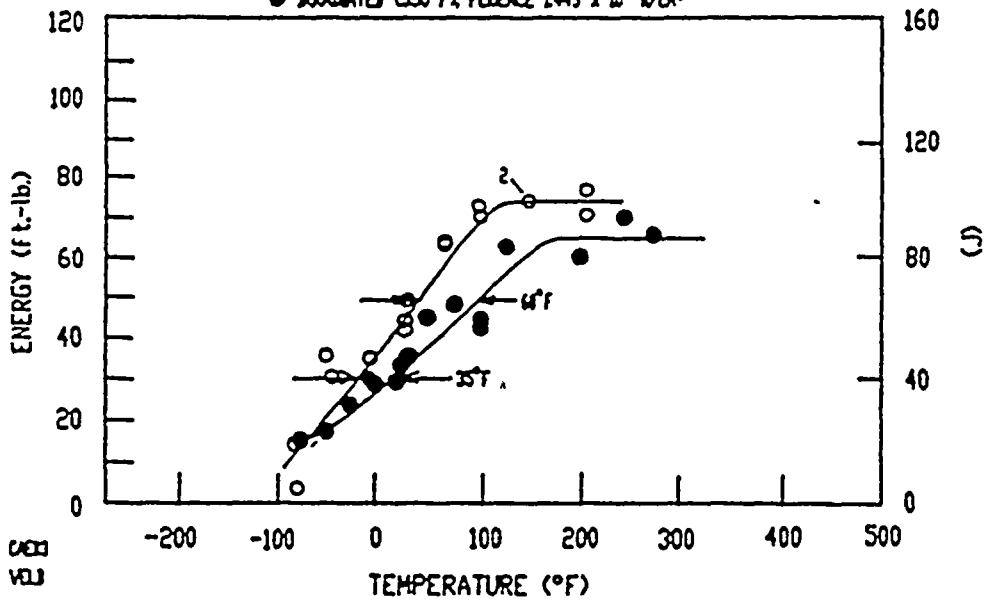
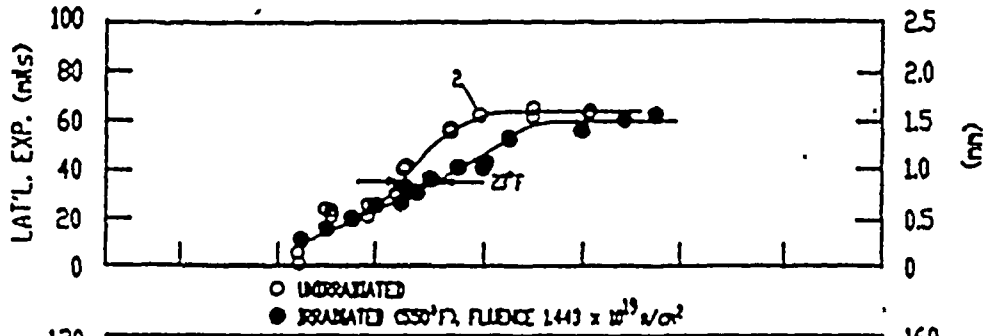
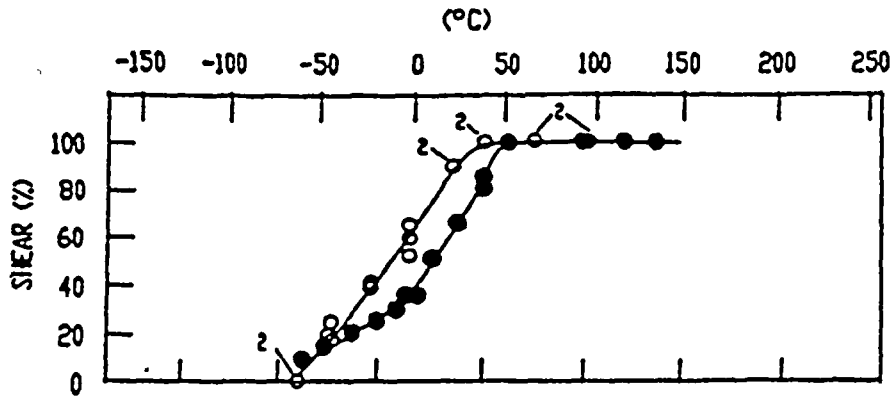
$$\Delta \sigma_Y = \Delta \sigma_D + \Delta \sigma_P$$

$\Delta \sigma_D$ = Change in yield strength due to defects

$\Delta \sigma_P$ = Change in yield strength due to precipitation

$\Delta \sigma_Y$ = Change in yield strength





○ UNIRRADIATED
 ● IRRADIATED (550°F, FLUENCE $1.43 \times 10^{19} \text{ n/cm}^2$)

- $\Delta\sigma_D$ Appears To Be Small Based On Results Of Low Cu Linde 80 Welds
- Supports Saturation Observation