ornl

OAK RIDGE NATIONAL LABORATORY

UNION CARBIDE

NUREG/CR-1351
ORNL/NUREG-64

Fatigue of Weldments in Nuclear Pressure Vessels and Piping

M. K. Booker
B.L.P. Booker
H. B. Meieran
J. Heuschkel

P-016

Prepared for the
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Under Interagency Agreement DOE 40-551-75

OPERATED BY
UNION CARBIDE CORPORATION
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

Printed in the United States of America. Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161

Available from

GPO Sales Program
Division of Technical Information and Document Control
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

NUREG/CR-1351
ORNL/NUREG-64
Distribution
Category R5

Contract No. W-7405-eng-26

METALS AND CERAMICS DIVISION

FATIGUE OF WELDMENTS IN NUCLEAR PRESSURE VESSELS AND PIPING

M. K. Booker, B.L.P. Booker, H. B. Meieran,
and J. Heuschkel

Manuscript Completed - February 26, 1980
Date Published – March 1980

NOTICE: This document contains information of preliminary nature. It is subject to revision or correction and therefore does not represent a final report.

Prepared for the
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555
Under Interagency Agreement DOE 40-551-75
NRC FIN No. B0103

Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
DEPARTMENT OF ENERGY

CONTENTS

ABSTRACT 1

INTRODUCTION 2

Method of Investigation 2

MATERIALS AND WELDING PROCESSES 7

Materials Considered 7

Welding Processes and Materials 8

OVERVIEW OF EXISTING CODES 8

Summary of Observations Regarding ASME Codes 11

GENERAL OVERVIEW OF EXISTING EXPERIMENTAL INFORMATION 13

Primary Variables 13

Weld Geometry 14

Defects 16

Cracks 18

Undercut 19

Lack of Fusion 20

Slag Inclusions 21

Arc Restarts 22

Porosity 22

Design Philosophies for Treatment of Weld Defects 24

Welding Residual Stresses 34

Microstructural Effects 35

Reheat Cracking 36

Crack Propagation 43

SUMMARY OF DESIGN AND OPERATING EXPERIENCE OF REACTOR PRESSURE VESSELS AND PIPING COMPONENTS 45

SUMMARY OF CURRENT RESEARCH 51

CONCLUSIONS AND RECOMMENDATIONS 64

ACKNOWLEDGMENTS 66

REFERENCES 66

APPENDIX A 81

APPENDIX B 87

FATIGUE OF WELDMENTS IN NUCLEAR PRESSURE VESSELS AND PIPING

M. K. Booker, B.L.P. Booker, H. B. Meieran,*
and J. Heuschkel†

ABSTRACT

The current American Society of Mechanical Engineers (ASME) Code fatigue design rules for nuclear pressure vessels and piping include no special considerations for weldments other than purely geometric factors. Some research programs aimed at nonnuclear applications have found weldments to display fatigue behavior inferior to that of pure base material. For this reason we reviewed available information on fatigue of weldments relevant to nuclear pressure vessels and piping and determined what (if any) changes in the current design rules appear to be dictated by the available information.

Our investigation consisted exclusively of a compilation and review of available information; no new data or information were generated. The information gathered consisted of a comprehensive literature survey and of extensive contacts with individuals and organizations with expertise in the areas involved. Information obtained was summarized and stored in a computerized data management system to facilitate correlation of facts and development of conclusions.

The significant areas where development of additional data would substantially increase our ability to judge the adequacy of the current ASME design rules include: (1) a better understanding of the relative importance of crack initiation and crack propagation to fatigue life; (2) additional fatigue data for prototypic commercial weldments, including cumulative damage; (3) properties of repair welds; (4) significance of reheat cracks; (5) quantitative effect of Code-allowable weld defects; and (6) the effect of variable microstructure across the weld joint. However, based on the information that is available, there is no evidence that the ASME Code fatigue design procedures need to be changed at this time.

This report reviews the current ASME design procedures, which form the general basis for fatigue evaluation both in the U.S. and abroad. Also included is a review of various factors that influence the fatigue of weldments and of service experience with nuclear systems regarding fatigue of weldments. Finally, ongoing research programs that may contribute to available information are reviewed. However, these programs are not generating most of the needed information at this time.

---

*O'Donnell and Associates, Inc., Pittsburgh.

†Julius Heuschkel and Associates, Pittsburgh.

INTRODUCTION

Pressure vessels and piping for light-water reactor nuclear power plants in the U.S. are designed and built in conformance with the requirements of Sect. III, Div. 1, Subsect. NB (Class 1 Components) of the *ASME Boiler and Pressure Vessel Code*[1] (denoted herein as ASME-III-NB). Both the pressure vessels and the piping are fabricated by welding. Figure 1 shows a schematic diagram of the components joined together to form a large pressure vessel.

The American Society of Mechanical Engineers (ASME) rules for design against fatigue failure include no special considerations for weldments (beyond some purely geometric factors that will be described later in this report). Many instances in which weldments display fatigue behavior inferior to that of base material have been documented. As a result our investigation was undertaken to review and assess available information on weldment fatigue and to make recommendations concerning the adequacy of the current ASME fatigue design rules pertaining to welds.

Our purpose was to survey and analyze existing information; no new data were generated. The survey was limited to fatigue of weldments relevant to nuclear pressure vessels and piping within the current state of the art of welding such structures. We did not attempt to assess or critique the actual procedures used in producing those welds. Furthermore, environmental phenomena, such as stress-corrosion cracking (which may be aggravated by fatigue but is not a fatigue effect), were not addressed.

Our primary objective was to investigate the adequacy of the fatigue design curves given in Appendix 1, ASME-III-NB (Fig. 2). We sought to determine if existing safety factors are adequate or if additional factors should be applied.

Method of Investigation

Our survey was two-pronged, consisting both of a review of available literature and of systematic contacts with individuals in various research

ORNL-DWG 67-1333R (M)

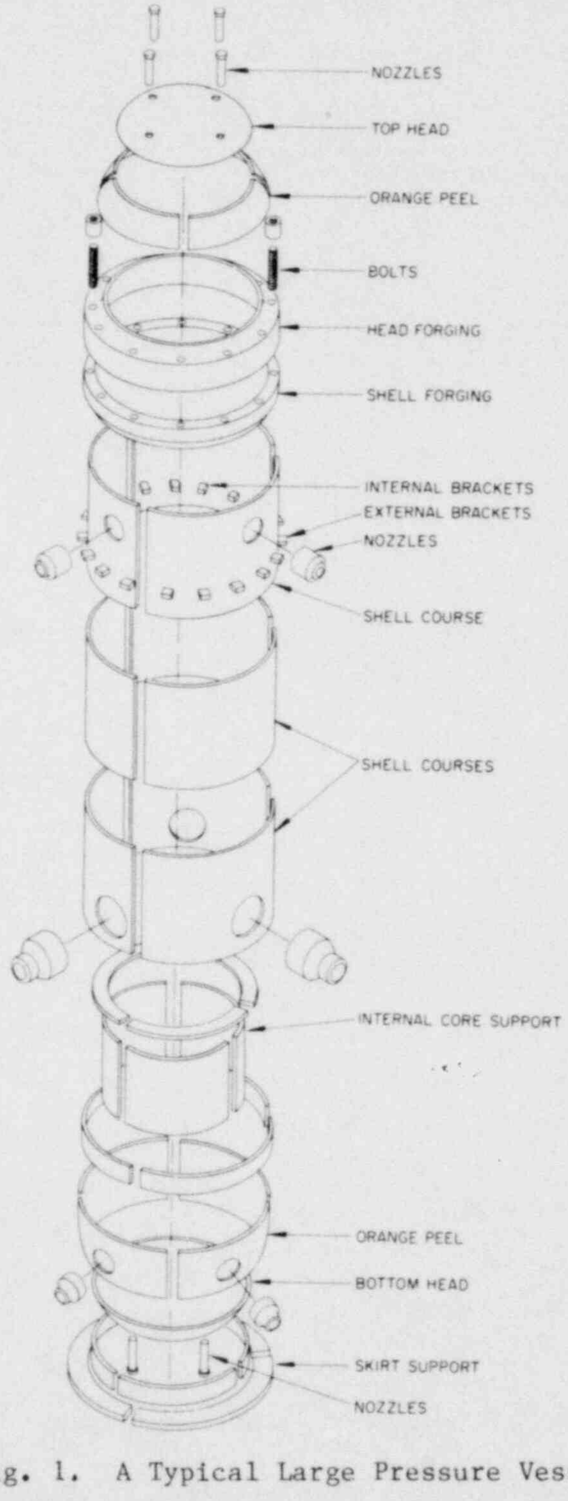

Fig. 1. A Typical Large Pressure Vessel.

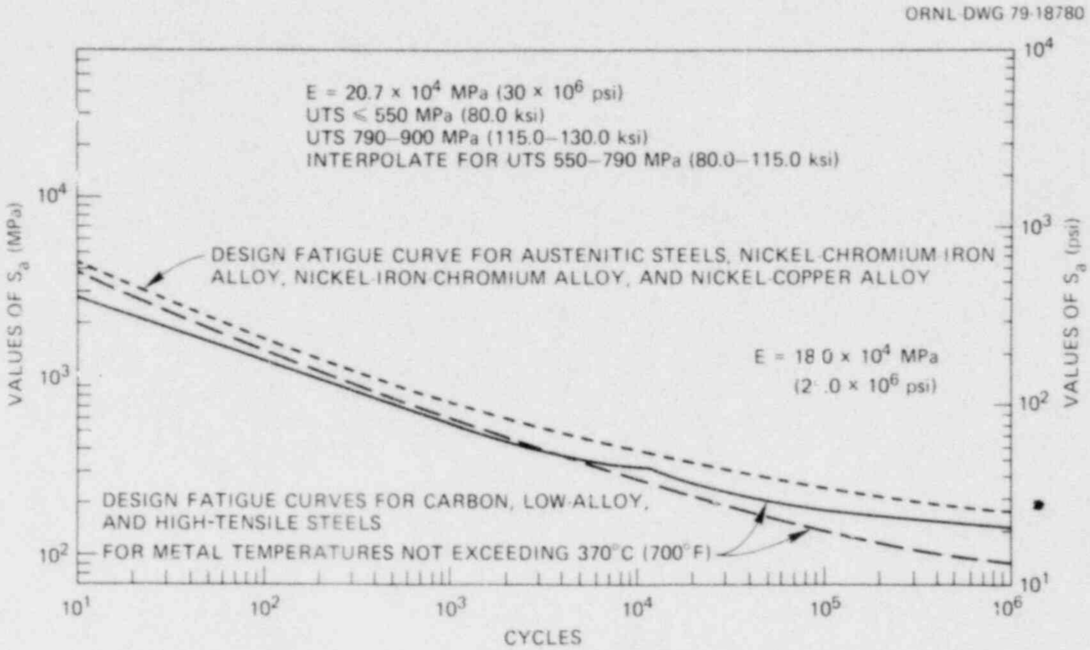

Fig. 2. Fatigue Design Curves from ASME Sect. III, Div. 1, Appendix I. Source: *ASME Boiler and Pressure Vessel Code, Sect. III, Div. 1, Appendix I*, American Society of Mechanical Engineers, New York, 1979, pp. 128–29.

and industrial organizations who were interested in the problems encompassed by our investigation. The institutions contacted were:

<u>Research Laboratories</u>

Argonne National Laboratory

Hanford Engineering Development Laboratory

Battelle Northwest Laboratory

Oak Ridge National Laboratory

Naval Research Laboratory

Bettis Atomic Power Laboratory

<u>Pressure Vessel Suppliers</u>

Babcock and Wilcox

Combustion Engineering

Chicago Bridge and Iron

Foster Wheeler

<u>Universities</u>

Lehigh University

University of Illinois

University of Kansas

Renssalear Polytechnic University

Stanford University

University of Tennessee

Iowa State University

Pennsylvania State University

University of Pennsylvania

University of Waterloo

Allied Dept. of Energy-Nuclear Regulatory Commission Activities

Battelle Memorial Institute

Southwest Research Institute

Electric Power Research Institute

International Organizations

Uddeholm (Sweden)

Canadian Atomic Energy Commission

Swedish Nuclear Power Inspectorate

Central Electricity Generating Board (UK)

United Kingdom Atomic Energy Authority

Framatome (France)

Materealprüfungsanstalt (Germany)

Miscellaneous

U.S. Dept. of Transportation

O'Donnell and Associates, Inc.

Engineering Decision Analysis (EDAC)

Reactor Manufacturers

Westinghouse

General Electric

Societies

American Society of Testing and Materials

American Society for Metals

American Welding Society

Pressure Vessel Research Committee of the Welding Research Council

American Society of Mechanical Engineers

American Society of Civil Engineers

Society of Automotive Engineers

International Welding Institute

Steel Suppliers

U.S. Steel

J&L

Bethlehem

Crucible

Allegheny Ludlum

We started surveying the literature within the framework of existing bibliographies[2-4] on the subject. The survey was then expanded to include current information as these original references led to additional ones.

Information obtained from the literature and from contacts with individuals was summarized on forms designed for the purpose, was key-punched, and was input to the ORNL Data Storage and Retrieval System (DSRS)[5] for subsequent analysis. This system was used to assimilate the information gathered and to facilitate the formation of conclusions from the information. Sample output from "LITREF," "Personal," and "Phone" computer inputs follows.

```
### 68 #####
<HEADER  >
<TY       >LITREF
<PUB      >S. J. MADDCX, A FRACTURE MECHANICS ANALYSIS OF THE FATIGUE BEHAVIOUR
OF A FILLET WELDED JOINT, WELDING INSTITUTE RESEARCH REPORT E/50/72 (JAN 1973)
<DA       >      7.3010E 03
<AU       >MADDCX SJ
<KEYWRDS >
<CONC     >SUCCESSFULLY PREDICTS FATIGUE LIFE OF WELDS BY IGNORING INIATION AND
CALCULATING CRACK PROPAGATION LIFE BY FRACTURE MECHANICS;
<ES       >FRACTURE MECHANICS; LIFE PREDICTION
<TJ       >FILLET
<INST     >WELDING INSTITUTE

### 69 #####
<HEADER  >
<TY       >PERSONAL
<DA       >      7.8100E 03
<AU       >CANONICODA
<KEYWRDS >
<CONC     >CURRENT NUCLEAR PRESSURE VESSEL TECHNOLOGY UTILIZES SA533B CLASS/PLATE
AND SA508 CLASS 2 FORGINGS CLADON INSIDE WITH STAINLESS STEEL; FERRITIC JOINTS
ARE STRESS RELIEVED PER ASME CODE (1100-1250F) FOR UP TO 28 HR; NO STRESS RELIEF
REQUIRED FOR AUSTENITICS; FATIGUE HAS NOT CAUSED SERIOUS PROBLEMS IN NUCLEAR
VESSELS ALTHOUGH THERE HAS BEEN CONCERN OVER REHEAT CRACKS; THE MAIN YANKEE
REACTOR HAD A FATIGUE PROBLEM IN STAINLESS PIPING
<ES       >STRESS RELIEF; CLADDING
<INST     >ORNL

<CONTACT >
<POS      >GROUP LEADER, PRESSURE VESSEL TECHNOLOGY GROUP
<ORIG     >M. K. BOOKER, ORNL
<CONT     >DOMENIC A. CANONICO
<LOC      >OAK RIDGE, TN.

### 70 #####
<HEADER  >
<TY       >PHONE
<DA       >      7.8100E 03
<AU       >RASKEDT
<KEYWRDS >
<TD       >FATIGUE; CRACK GROWTH
<CONC     >MOST FATIGUE AND CRACK GROWTH WORK ON WELD MATERIAL AT FAIRLY HIGH
TEMPERATURES IN SUPPORT OF LMFBR; SENT SEVERAL PUBLICATIONS DESCRIBING WORK;
RECENT WORK TO BE REPORTED IN JANUARY ORNL MECHANICAL PROPERTIES SEMIANNUAL
<EM       >304SS; 316SS
<FM       >16-8-2; 308SS; 308SSCRE
<INST     >ANL
<CONTACT >
<ORIG     >M. K. BOOKER, ORNL
<CONT     >DAVID T. RASKE

### 71 #####
<HEADER  >
<TY       >PHONE
<DA       >      7.8110E 03
```

MATERIALS AND WELDING PROCESSES

Materials Considered

Our first emphasis was the primary containment pressure vessel, with secondary emphasis on the primary piping system in light-water reactors (LWRs). The vessels are constructed almost exclusively of SA533B Class 1 steel plate[6] and/or SA508 Class 2 steel forgings.[7] In some cases SA508 Class 3 steel forgings[7] are used. The piping systems are composed primarily of AISI type 304 austenitic stainless steel[8] or SA516 Grade 70 material.[9] Table 1 shows the chemical compositions and specified tensile properties of these five materials. Filler metals and fluxes used with these materials are selected for their compatibility with the base metal. In general the weld deposit for the ferritic materials is similar in composition to the base material except that the weld metal is generally limited to a maximum carbon content of 0.12 wt %. This restriction is applied because carbon is known[10] to decrease the toughness of carbon and

Table 1. Chemical Compositions of Typical Nuclear Pressure Vessel and Piping Materials

Element	Steel Composition, Maximum and Range Values, wt %				
	SA508 Class 2	SA508 Class 3	SA533B Class 1	SA516 Gr 70	Type 304 Stainless
C	0.27	0.15–0.25	0.25	0.28[a]	0.08
Mn	0.50–0.90	1.20–1.50	1.10–1.55	0.80–1.25	2.00
P	0.025	0.025	0.035	0.035	0.45
S	0.025	0.025	0.04	0.04	0.30
Si	0.15–0.35	0.15–0.35	0.13–0.32	0.13–0.33	1.00
Cr	0.25–0.45				18.00–20.00
Ni	0.50–0.90	0.40–0.80	0.37–0.73		8.00–10.00
Mo	0.55–0.70	0.45–0.60	0.41–0.64		
V	0.05	0.05			

[a]Actual value depends on section thickness. This value is typical.

low-alloy steel welds. The type 304 stainless steel is welded by using type 308 stainless steel filler material. The composition of type 308 resembles that of type 304, with slightly increased ratios of ferrite-forming elements to austenite-forming elements. As a result welds made with type 308 contain δ-ferrite and avoid cracking problems often associated with purely austenitic weld metal.[11] In addition the ferritic materials are generally clad on the interior by a type 308 stainless steel weld deposit for protection from the possible corrosive effects of the reactor water environment.

In the course of the investigation, few data were available for the exact materials of interest, especially for the pressure vessel steels. Thus, our search for general trends was broadened to include similar materials and even the entire range of low-alloy steels.

Welding Processes and Materials

For largely economic reasons nuclear pressure vessels, attachments, and piping are most commonly welded with the submerged-arc (SA) and shielded metal-arc (SMA) welding processes. Welding is performed with multiple passes, with perhaps 100 passes needed to complete a weld between thick sections. Welds in the pressure-containing components use full penetration. Figure 3 illustrates a submerged-arc weld, showing the base metal, weld metal, and heat-affected zone. Details concerning welding procedures can be found elsewhere.[12-14]

OVERVIEW OF EXISTING CODES

Although our primary efforts were directed toward evaluation of the fatigue design procedures currently used in ASME-III, we also surveyed design codes from Britain, West Germany, Sweden, France, Japan, and Canada. In general, designers in the various countries employ the philosophy embodied in the ASME Code. The scopes for employment of the various applicable sections of this Code are given in Appendix A.

As stated above, the basic code that describes procedures governing the methodology for fatigue analysis of Class 1 components is ASME-III-NB

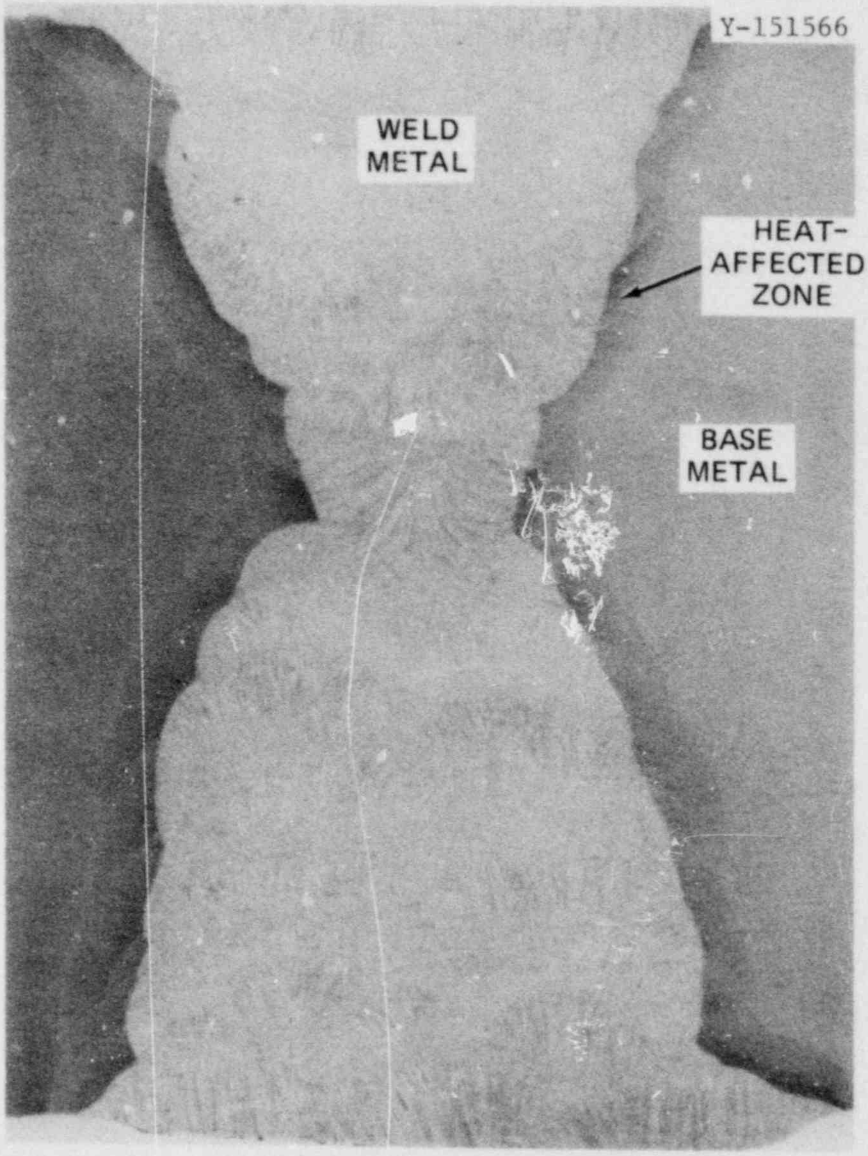

Fig. 3. A Butt Weld, Showing Base Metal, Weld Metal, and Heat-Affected Zone.

and its appendices. This code provides methods to define an alternating stress amplitude (S) with which an allowable cyclic lifetime (N) can be described. The alternating stress is a combination of the primary, secondary, and peak stresses and incorporates stress concentration factors (SCF). The SCF is defined by the geometry of a specific component. The SCF used for welds is geometric only and includes no special factors to identify a weld.

Specific mandatory fatigue design curves (S vs N) are given for various Class 1 materials in Appendix 1 to ASME-II-NB. These curves are illustrated in Fig. 2. The ASME-III states that these design curves "are obtained from uniaxial strain cycling data in which the imposed strain amplitude (half range) is multiplied by the elastic modulus to put the values in stress units. A best fit to the experimental data is obtained..."[1] The curves are adjusted where necessary to include the maximum effect of mean stress. The design stress intensity values are obtained from the best-fit curve by applying a factor of 2 on stress or a factor of 20 on cycles, whichever is the more conservative at each point. Figure 4 illustrates the construction of such a curve for type 304 stainless steel. Alternating stresses of variable amplitude are treated on the basis of Miner's life fraction rule.[15] The fatigue design curves and methods are not applicable above 371°C (700°F) for carbon, low-alloy, and high-tensile steels and above 427°C (800°F) for austenitic steels and nickel-copper alloys. Note that the material properties of weld metal and heat-affected zone material are considered the same as those for the base metal.

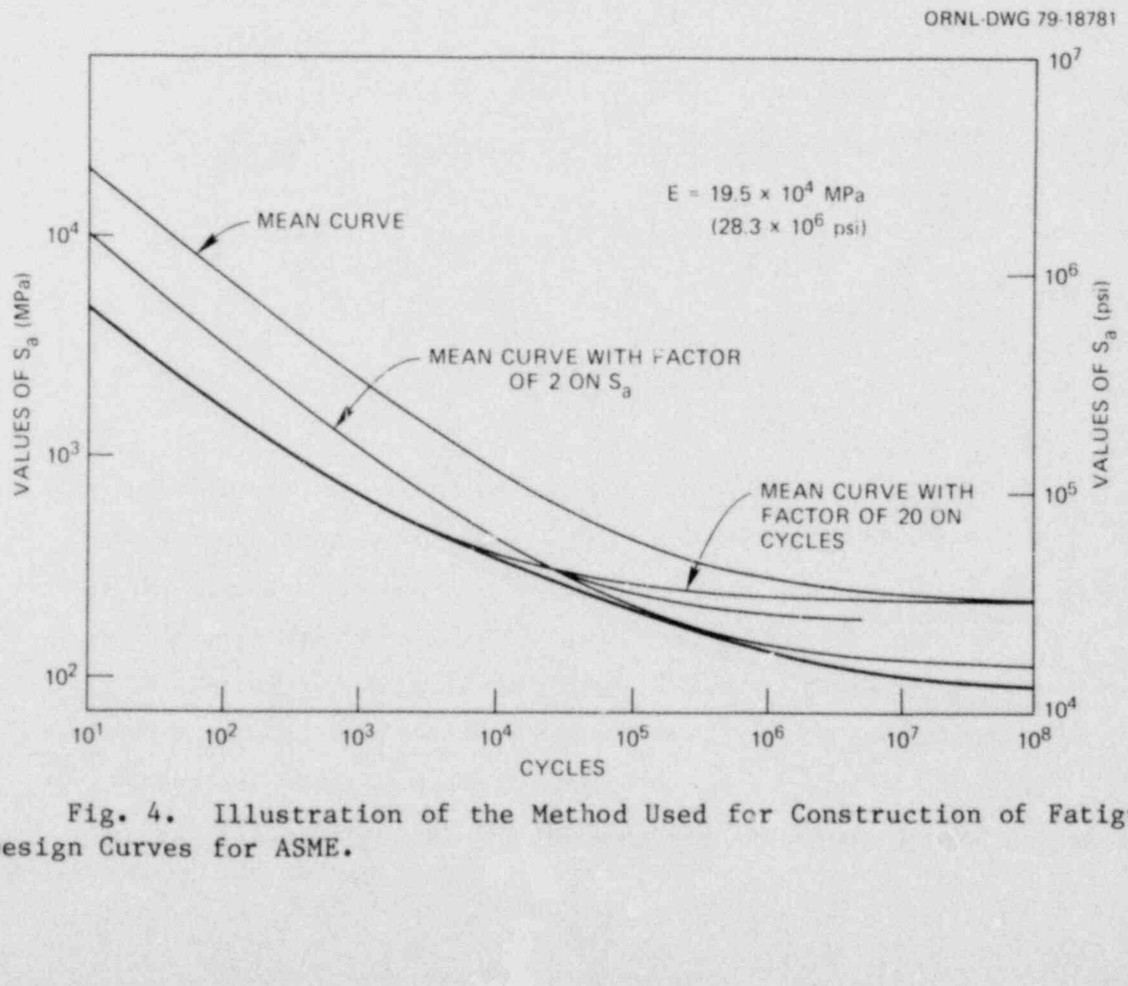

Fig. 4. Illustration of the Method Used for Construction of Fatigue Design Curves for ASME.

Summary of Observations Regarding ASME Codes

In design by analysis the alternating stress S_{alt} is multiplied by the ratio of the modulus of elasticity given in the respective design fatigue curve to the value of the modulus of elasticity used in the analysis. As there are no specifically defined modulii for weld metals in the Codes, the designer must employ specified values for the base metal. Furthermore, the designer is not required to consider the moduli at the temperature of operation as opposed to that at room temperature. An additional complicating factor is that no moduli are specified for bimetallic weldments (ferritic-to-austenitic steels). The designer is left with the option of considering the average modulus of two sides of a gross structural discontinuity (at room temperature). Furthermore, no guidance is given on the choice of which of the two types of fatigue curves in Fig. 2 to use for a bimetallic weldment.

In the analysis of pipes, the designer is correspondingly not given any guidelines on how to assess the coefficient of thermal expansion of a bimetallic weld; he/she is only guided to the coefficient of thermal expansion on one or the other side of a gross structural discontinuity (again at room temperature).

In the simplified elastic-plastic analysis of discontinuities, the fatigue strength reduction factor, K_e, is used to account for geometric effects. This factor is applied as a correction to the allowable alternating stress and is dependent upon the nature of the base metal. The value of K_e can vary from 1.0 to as much as 5.0, as illustrated in Fig. 5. Note that S_n is the primary plus secondary stress, while S_m is the allowable stress intensity.

A summary of the fatigue strength reduction factors defined for ASME-III for vessels, pumps, valves, and pipes is presented below.

Design by Analysis: $K_e \leqslant 5.0$.

Vessel:

K_e = 1 for category A full-penetration weld.

K_e = 1 for category B full-penetration.

$K_e \geqslant 2$ for category B full-penetration weld with backing strips not later removed.

K_e = 1 for category C full-penetration weld.

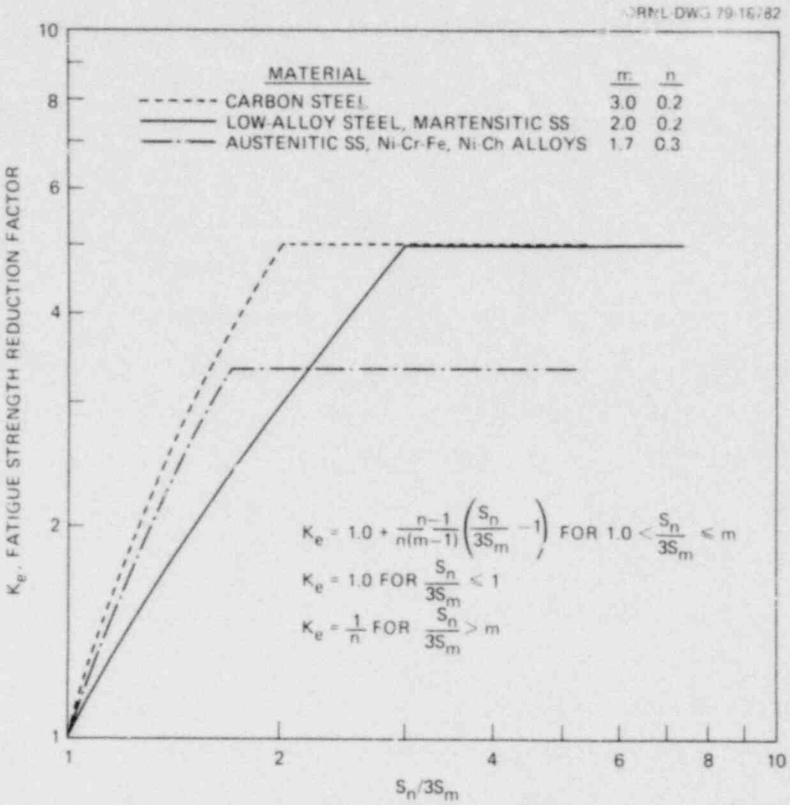

Fig. 5. Fatigue Strength Reduction Factors for Elastic-Plastic Analysis.

K_e = 1 for category D full-penetration weld.

K_e = 4 for category D full-penetration weld.

Pumps: K_e = 1 for full-penetration welds.

K_e = 4 for partial-penetration weld piping connections that are less than 51 mm (2 in.).

Valves: K_e = 2 for normal duty of external fillet at the crotch.

Pipes: K_e = 1.1–3.8 for girth butt welds.

K_e = 4.2–6.0 for girth to socket.

K_e = 1.1–3.5 for longitudinal butt.

K_e = 1.0–6.0 for butt welding tee.

Elastic-plastic (valves and pipes only): K_e = 1–5.0 for carbon steel.

K_e = 1–5.0 for low-alloy steel and margin.

K_e = 1–3.33 for austenitic stainless steel Ni-Cr-Fe and Ni-Cu alloys.

Note that the factors for pipes (up to 6.0) are in excess of the maximum listed for design by analysis.

Although specific rules in ASME-III compensate for localized stresses associated with geometric discontinuities, such as weldments, nuclear power plant component designers sometimes provide an additional "safety factor" in the design of weldments. This safety factor is essentially the placement of weld joints in areas where the operating stresses will be lower than those for the unwelded base metal. Designers of components for the Naval Nuclear Program automatically reduce allowable alternating stresses for weldments to 80% of those allowed for base metal.

GENERAL OVERVIEW OF EXISTING EXPERIMENTAL INFORMATION

Although relatively little experimental information is available on fatigue of weldments involving the materials of our primary interest, there is massive documentation concerning fatigue of weldments in general. This literature was scanned in search of general trends that might be useful in our evaluation. Much of the existing information is irrelevant for our application, and some is conflicting or confusing. Nevertheless, some trends do emerge and are summarized in this section.

Primary Variables

Our purpose was to seek reasons why weldments may display fatigue properties inferior to those of base material. It has often been found that such inferiority does exist under certain conditions. This inferiority has generally been attributed to early crack initiation caused by the presence and properties of the weld rather than to inferior crack growth characteristics. A review of the literature in general and of several excellent overview articles[16-19] reveals several common reasons for possible disparities between base metal and weldment fatigue behavior. These reasons include:

1. weld geometry,
2. defects,

3. residual stresses resulting from welding, and
4. variable microstructure and materials properties resulting from the weld.

Available information regarding these effects will be reviewed in this section, primarily from crack initiation. In addition, the problem of reheat cracking, which has caused some concern in the nuclear industry, will be separately reviewed. Finally, available information regarding fatigue crack growth behavior of the materials of interest will e reviewed. Details of testing techniques for weldments will not be discussed. Reports by Munse[18] and Reemsnyder[19] are good starting points for those wishing information about specifics of test methods.

Weld Geometry

It has generally[16-18] been noted that fatigue failures in welded structures tend to initiate in areas of local stress concentration. This result has been reinforced[20-21] for pressure vessels. Welds can introduce macrogeometric stress concentrations resulting simply from weld configuration in a variety of ways (Fig. 6). Time and again[22-35] researchers have found that such stress concentrations cause significant reductions in fatigue life. Figures 7 and 8 illustrate typical geometry effects in the classical case of a butt weld with reinforcement. Removal of the stress concentration, such as by grinding off the reinforcement, has general been found to improve fatigue strength substantially.[24-35] For example, in a compilation by Pollard and Cover,[17] the fatigue strength of mild steel reinforced butt welds was found to lie in the range of 44 to 81% of the base metal fatigue strength, depending on geometry. Removal of the reinforcement increased this range 75 to 100%, depending on the soundness of the weld.

An exception can be found in the work of Archer.[36] Fatigue tests of model pressure vessels showed that fatigue initiated at pores that had been exposed to the surface in the process of machining off the reinforcements in the seam welds of the cylindrical pressure vessels. This result led Boulton[37] to recommend leaving the reinforcement in vessel main seams but grinding the weld toe to decrease the severity of the local stress to concentration. However, general acceptance of this recommendation was not evident. Also, these conclusions applied to nonnuclear vessels.

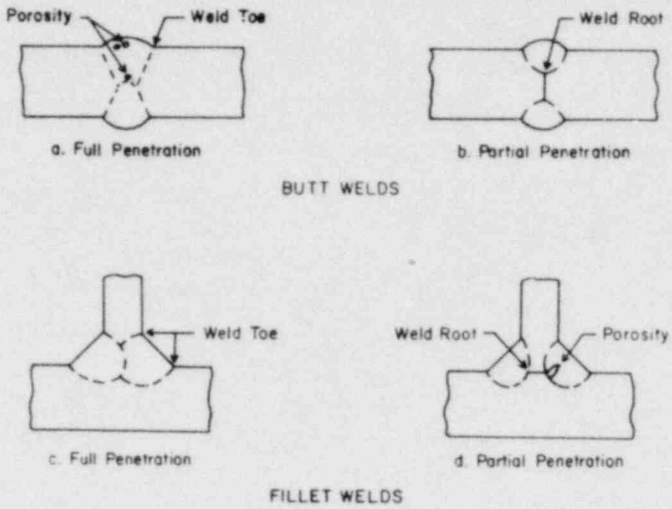

Fig. 6. Typical Stress Concentrations Occurring at Welds. The arrows indicate points of possible geometric stress concentrations. Note that partial penetration welds are not allowed in nuclear pressure vessels and primary piping. Reprinted from: H. S. Reemsnyder, "Development and Application of Fatigue Data for Structural Steel Weldments," p. 8 in *Fatigue Testing of Weldments, Am. Soc. Test. Mater. Spec. Tech. Publ.* 648, American Society for Testing and Materials, Philadelphia, 1978.

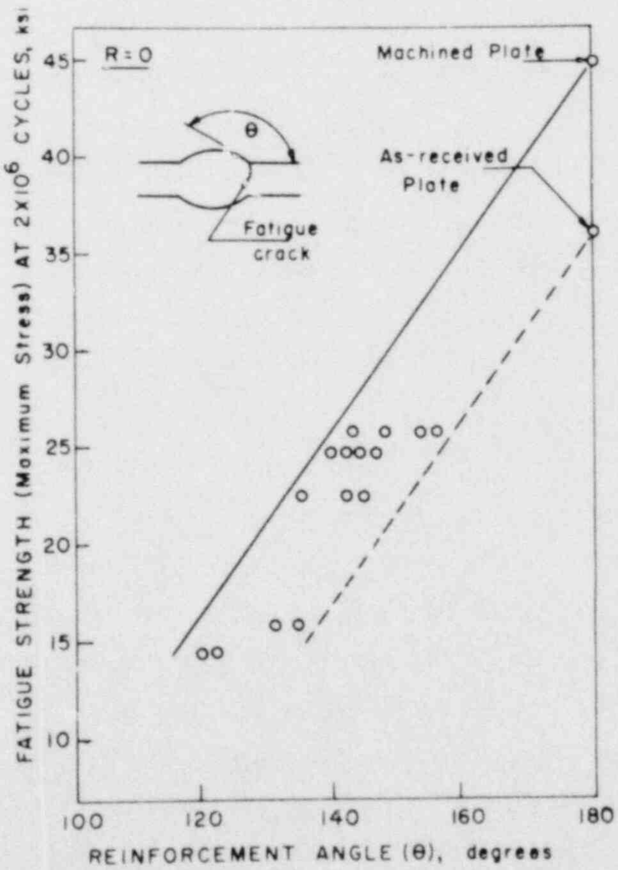

Fig. 7. Relationship Between Fatigue Strength and Butt Weld Reinforcement Height. Reprinted from: B. Pollard and R. J. Cover, "Fatigue of Steel Weldments," *Weld. J. (Miami)* 51: 546-s (November 1972). (1 ksi ≅ 6.895 MPa.)

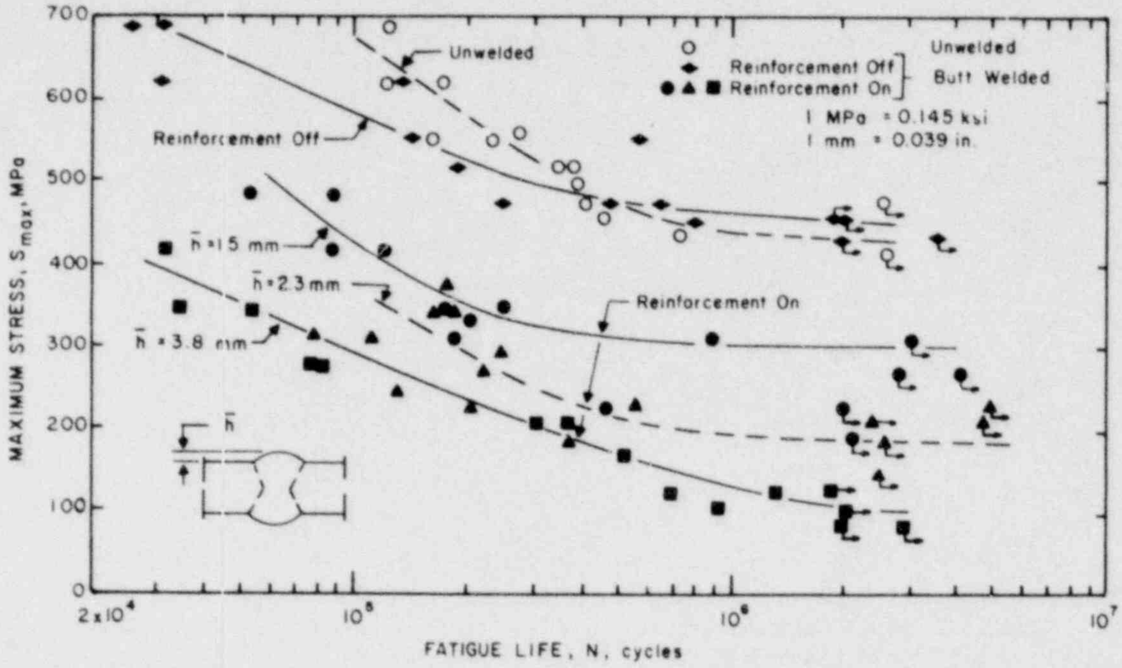

Fig. 8. Effect of Reinforcement Height on Fatigue Behavior of Butt Welds. Reprinted from: H. S. Reemsnyder, "Development and Application of Fatigue Data for Structural Steel Weldments," p. 10 in *Fatigue Testing of Weldments, Am. Soc. Test. Mater. Spec. Tech. Publ.* 648, American Society for Testing and Materials, Philadelphia, 1978.

Regardless of the general importance of geometric factors in determining the fatigue strength of weldments, they are not of major concern in this investigation. In most of the welds being studied here, there are no geometric stress concentrations. Moreover, where such concentrations do exist the existing fatigue strength reduction factors employed by the ASME Code appear[38] conservative.

Defects

Perhaps of more concern than geometry in the design of nuclear pressure vessels and piping are various types of weld defects, which include such geometric malformations as misalignment, undercut, surface irregularity, or lack of penetration. Other defects include arc strikes, slag inclusions, cracks, and pores. It is also possible for the weld metal or nearby heat-affected zone to undergo substantial detrimental microstructural alteration, such as growth of oversized grains or segregation of various constituents.

Note that the mere existence of any defects may or may not yield degradation in fatigue or any other properties. To avoid confusion the term "defect" should strictly be applied only when the effects render the product incapable of meeting acceptable standards of performance.[39] Otherwise, the term "discontinuity" should be applied to any interruption in typical weldment structure, such as those listed above. A discontinuity may or may not be a defect. Since our concern is with reductions in properties, we will primarily use the term defect in the following discussions, although such terminology may not be strictly correct in all cases.

The exact influence of defects on fatigue properties depends upon the nature and location of the defect, the type of loading, and the interactions with various other effects. As a result, published information sometimes appears contradictory and confusing. The essential trend that emerges is that defects can act as local points for initiation of fatigue cracks and thus lead to premature failure.[23,40-59] Most available data come from high-cycle (10^5 to 10^7 cycles to failure) fatigue tests, although some information is available for low-cycle fatigue. An excellent recent review of the data is available.[39]

Based primarily on high-cycle fatigue data, Matting and Neitzel[45] ranked different types of defects in order of severity of their detrimental effects on fatigue strength. Their ranking was:

1. cracks,
2. undercut,
3. lack of fusion,
4. slag inclusions,
5. arc restarts, and
6. porosity.

Any such ranking only indicates general trends, and any generic defect ranges in severity, depending upon size, shape, location, orientation, residual stresses, loading conditions, and other factors. For example, surface defects are generally more damaging than similar defects buried within the weld, as revealed in the work of Archer[36] and of Boulton[37] noted above.

The quantitative definition of the effects of various types of weld defects is also complicated by two practical considerations. First, prototypic production welds generally do not contain the systematic variations in defect type, size, and location required to allow an analytical description of effects. Conversely, welds with intentional defects to provide these systematic variations may not be representative of the behavior expected in production welds. Second, in a real structural application it may not be possible to gather all the information needed to fully characterize any existing weld defects through available nondestructive examination techniques. Thus, the exact procedures that should be used to design against the effects of weld defects are somewhat open to debate. We will briefly review available information concerning the influence of different types of weld defects on fatigue strength. Then, efforts to develop design systems based on this information will be discussed.

Cracks

The detrimental effect of cracks on fatigue strength is recognized in ASME Sect. III, Subsect. NB-5000 on examination. Any indication of the existence of cracks in a weld by preservice nondestructive examination is cause for rejection of the weld. Such cracks may occur for several reasons, either in the weld metal or in the nearby parent metal.

Cracks can commonly occur through solidification cracking in the hot weld metal or through cold cracking in hardened weld metals or heat-affected zones (HAZs). Linnert[60] gives a good general description of such possible cracking mechanisms. In addition the phenomenon of "reheat cracking" in the HAZ can produce small cracks that elude detection by nondestructive techniques. This phenomenon will be discussed later.

There has been little direct examination of the effect of cracks on fatigue strength in welds, probably because existence of detectable cracks usually precludes a weld from going into service. The work of Warren,[61] in which the existence of cracks transverse to the direction of applied stress in a mild steel weld yielded approximately a 55 to 65% reduction in the 10^7-cycle fatigue strength, is an exception. Unfortunately, the exact severity of those cracks is unclear.

The presence of microcracks in high-strength steel weldments has been indirectly blamed[51,61] for their typically not showing improved fatigue strength over weldments in lower strength steels. However, the effect of cracking was not systematically investigated. On the other hand, Munse et al.[62] found that cracks in HY-80 steel had no detrimental effect on fatigue behavior. Still, the existence of a crack and the corresponding stress concentration would be expected to serve as a likely point for initiation of a fatigue crack, and we feel it appropriate that ASME-III-NB-5000 precludes the existence of detectable cracks in preservice inspection.

Of course, detectable cracks might develop during service, even though not in existence beforehand. The ASME Sect. XI gives allowable defect indications from in-service inspections. Possible implications of these rules for fatigue will be discussed later in this report. Defects that exceed the allowable indications must be removed or repaired unless the operator can show by the fracture mechanics methodology of Appendix A to Sect. XI that the flaw will not grow to critical proportions during the service life of the component. Some investigators[43,63-68] have maintained that the fatigue strength of welds can be analyzed solely from crack growth, thus eliminating any possible unconservative results from early crack initiation. That work will be discussed in more detail later in this report.

Undercut

Undercut at the weld toe (Fig. 9) is a welding defect whose effect is caused strictly by a surface geometric stress concentration. As such, the comments in the above section on geometric effects apply equally to undercuts. However, undercutting is a "defect" in that it would never be intentionally introduced, whereas some geometric effects such as a reinforcement might be. A quantitative study[69] of undercut effects showed that a 0.9-mm-deep undercut caused a nearly 50% reduction in the fatigue endurance limit of mild steel welds in pulsating tension fatigue, while a 1.27-mm-deep undercut reduced the fatigue life in HY-80 welds by about a factor of 3. Still, undercutting is relatively easy to prevent by good welding practice and is forbidden by ASME-II-NB-5000.

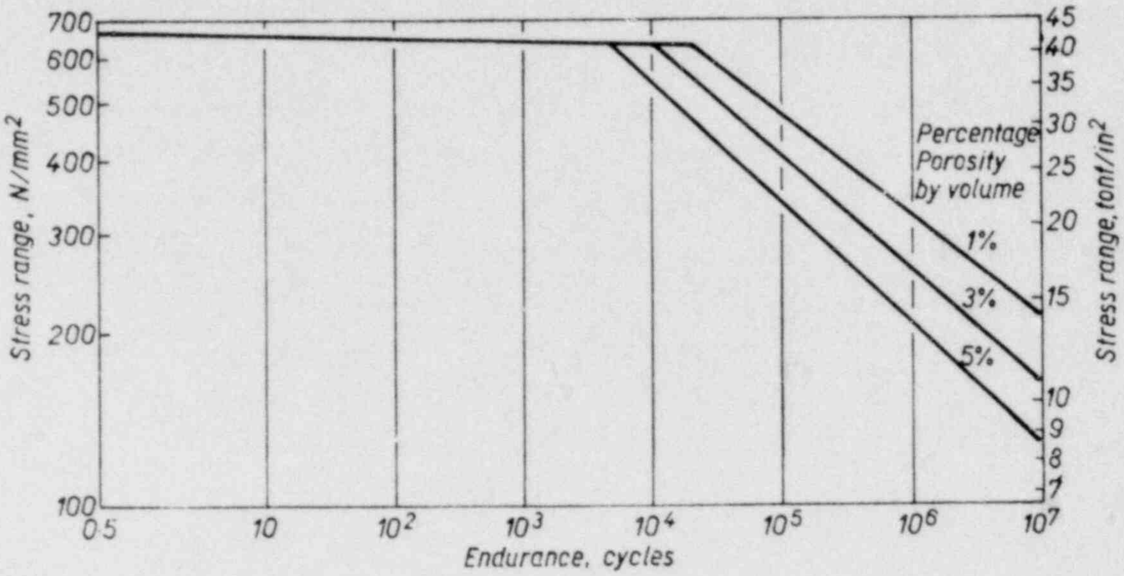

Fig. 9. Load-Controlled Fatigue Behavior of Porous Welds. Reprinted from: J. D. Harrison, "Basis for a Proposed Acceptance-Standard for Weld Defects. Part 1: Porosity," *Met. Constr. Br. Weld. J.* 4(3): 101 (March 1972).

Lack of Fusion

Lack of fusion (LOF) and lack of penetration (LOP) are similar and should be considered together. However, since we are concerned with full-penetration welds, lack of fusion is of primary importance. In full-penetration welds, lack of fusion will not occur by design but could be introduced in poor welding procedures that cause localized areas of nonfused metal. Incomplete fusion is also forbidden by ASME-III-NB-5000, although many investigators apparently[39] believe that certain amounts of LOP or LOF are acceptable. The effects of LOF or LOP are thought[39] to resemble those of slag inclusions (discussed below), although of perhaps slightly larger magnitude.[44,70-73]

Most direct investigations have concerned LOP rather than LOF, but the same conclusions should apply to both. Fillet welds often include lack of penetration by definition, so all work on these is somewhat applicable to the determination of LOP defects. Lack of penetration defects oriented parallel to the applied stress appear[74] to exert little or no effect on fatigue strength. Likewise, the effects of LOP are maximized when the defect is oriented perpendicular to the stress axis.

For example,[75] the reduction in fatigue strength resulting from transverse LOP in butt welds has been related to the percentage of the weld over area covered by the defect. The percent reduction in fatigue strength was found[74] to be roughly twice that in weld area by the defect. However, any significant lack of fusion defect would be revealed in preservice nondestructive examination and would thereby be excluded from service.

Slag Inclusions

Slag inclusions are allowed in welds that go into service in nuclear systems per the restrictions given in ASME-III-NB-5000. However, slag inclusions can detrimentally affect fatigue strength, especially in flush machined welds.[17,42,49,62,72] (In welds with a reinforcement or other geometric factor, the geometry effects become controlling and tend to negate any harmful effects of the slag inclusions.)

In a recent review of available information on the effect of slag inclusions, Harrison[49] notes that both the length and depth of a slag inclusion would be expected to affect the stress concentration resulting from the defect and thus to influence the resultant decrease in fatigue strength. Furthermore, as the length-to-depth ratio increases, the length becomes gradually less important, until finally (above $l/d \approx 10$) the stress intensity factor depends on depth alone. However, in real welds the depth of slag inclusions generally does not vary widely, with inclusions in multipass welds being limited to the depth of one weld pass. However, the length of the inclusions may vary considerably. Moreover, radiography is the most commonly used examination technique, and radiographs, being two-dimensional, cannot yield measures of all dimensions. Therefore, Harrison[49] recommended that the lengths of slag inclusions be used as the primary variable for practical estimation of effect on fatigue life.

Location of slag inclusions is also an important factor, with inclusions near the surface generally somewhat more damaging than those buried within the weld.[17,44,72,76] This effect may partially result from[58] the buried inclusions in a compressive stress field and the surface inclusions in a tensile residual stress field in nonstress-relieved welds. Citing published data[58,77-87] Harrison[49] also drew the following general conclusions about the influence of slag inclusions on fatigue behavior.

1. Welds containing similar slag inclusions appear to display higher fatigue strength when made with low-hydrogen electrodes than with rutile electrodes. This difference was attributed[81] to the deleterious effects of hydrogen in the slag-bearing welds.

2. Stress relief can be harmful or helpful, depending on the situation. For buried slag inclusions removal of residual compressive stresses may be harmful. However, stress relief may also result in the removal of hydrogen from the weld and thus be beneficial. In general,[58] stress relief has a net beneficial effect on discrete buried inclusions or on near-surface inclusions. The net effect of stress relief may be harmful for continuous slag lines near the center thickness.

3. A given size of slag inclusion becomes less detrimental as the section thickness is increased.

Arc Restarts

Very little information exists on the effect of arc restarts on fatigue behavior. However, Matting and Neitzel[45] considered such defects to be of minor importance. For nuclear pressure vessels and piping, they probably do not play a significant role.

Porosity

Although listed by Matting and Neitzel[45] as being even less consequential than arc restarts, porosity deserves more attention here for these reasons: (1) The effects of porosity can vary widely, and some data indicate that porosity can be very detrimental in fatigue behavior; (2) A significant amount of fatigue data on welds containing porosity is available for analysis; (3) Porosity is likely to occur in some amount in most production welds.

For butt welds porosity seems to have little effect on fatigue strength of a joint with reinforcement intact.[17,48,50,56,73,77] In addition, while small amounts of porosity can in some cases cause significant decreases in fatigue strength, further increases in porosity level have less and less effect.[17,56,73,77,88,89] Finally, the sensitivity to porosity increases as material strength increases.[17]

Probably the best available summary of information regarding effects of porosity is again from Harrison,[48] although his work deals only with ferritic steels. Available data[42,84,87,90,91] (again mostly for high-cycle fatigue) were compiled and examined for trends. However, limited data of Ishii and Iida[87] indicate that the effect of porosity on low-cycle fatigue is small. The conclusion from this compilation was that percentage reduction in cross-sectional area is the best criterion to use in assessing the significance of porosity to fatigue performance. However, one should exercise some caution in determining the relevant percentage. Since fatigue crack initiation is localized, localized clusters of porosity can result in early crack initiation and thus failure. Thus, if the defect area percentage is calculated based on the entire cross-sectional area of a component or test piece, results may be overly optimistic for clustered porosity. Several sets of test data[62,76,91] confirm this possibility.

Other conclusions from Harrison's work[48] include:

1. Surface pores are somewhat more damaging than buried ones.[56,92,93]

2. The effect of single large pores is unclear. However, within the range of pore sizes studied [up to about 3.5 mm (0.14 in.)] the effect of large pores is the same as that of a group of smaller pores.

3. Linear porosity is probably no worse than uniform porosity,[91] although linear porosity in a production weld may indicate planar defects, such as lack of fusion.

4. More data are needed to assess the effects of elongated pores, such as blowholes.

5. High levels of porosity may prevent detection of other defects by standard nondestructive examination techniques. (However, the level of porosity in welds in nuclear pressure vessels and piping is probably well below the level where such masking would occur.)

6. Typical levels of porosity in production welds appear to have little or no effect on low-cycle load-controlled fatigue behavior, while their effect on high-cycle behavior can be systematically related to percentage porosity by cross-sectional area or volume (Fig. 9).

Comparison of Harrison's porosity results with nuclear pressure vessel conditions suffers from the same limitations concerning size effects as mentioned above for slag inclusions. The high-cycle fatigue results can again be compared with those of Soete and Sys,[73] and again their data for a thick-walled test pressure vessel indicated that conclusions drawn from smaller test specimens should be conservative. However, tests on pressure vessels with nozzles[94] showed that effects of porosity on low-cycle fatigue might be more serious than for the small test specimens of Ishii and Iida.[87]

Design Philosophies for Treatment of Weld Defects

As noted elsewhere in this report, current ASME design procedures for nuclear power plants include no special methods for treatment of welds in fatigue design other than some purely geometric factors. Therefore, ASME ostensibly includes no direct treatment of the effect of weld defects on fatigue strength. However, such effects are indirectly addressed by the specification of allowable defect levels.

The ASME Sect. III-NB-5300 specifies allowable defect levels from preservice inspection of welds in Class 1 nuclear components. Any weld defect not meeting those standards must be removed or reduced to acceptable levels before going into service. Welds of concern to this study must be fully radiographed. In addition, the weld surfaces must be examined by either the magnetic particle or liquid penetrant method.

Unacceptable defects from radiographic examinations include:

1. any type of crack or zone of incomplete fusion or penetration;
2. any other elongated indication that is longer than 6 mm (1/4 in.) for thickness of thinner portion of weld (t) up to 19 mm (3.4 in.) inclusive and 19 mm (3/4 in.) for t over 57 mm (2 1/4 in.);
3. any group of aligned indications with an aggregate length greater than t in an area of length $12t$ unless successive indications are separated by a distance of greater than $6L$, L being the length of the largest indication; and
4. rounded indications in excess of those shown in Table 8 from Appendix VI, ASME-III.

Unacceptable defects by either the magnetic particle or liquid penetrant surface examinations are defined by the following standards:

1. Only indications with major dimensions exceeding 1.6 mm (1/16 in.) are considered relevant.

2. Relevant indications that are unacceptable include: any cracks and linear indications, rounded indications larger than 4.8 mm (3/16 in.), four or more rounded indications in a line separated by 1.6 mm (1/16 in.) or less edge to edge, and ten or more rounded indications in any 3870 mm^2 (6 $in.^2$) of surface with the major dimension of this surface area not to exceed 152 mm (6 in.) and with the area taken in the most unfavorable location relative to the indications being evaluated.

In addition, ASME Sect. XI includes acceptance standards for defects in welds from in-service nondestructive examinations. These standards include flaw indications representative of cracks, slag inclusions, porosity, lack of penetration, lack of fusion, and laminations. Separate standards are given for various component categories, such as reactor vessels, other vessels, vessel nozzles, piping, etc. Tables 2 and 3 reproduce tables from ASME Sect. III and Sect. XI, illustrating the philosophy of approach used by ASME. These tables refer to the allowable sizes of indications in reactor vessels, as defined in Fig. 10. In addition, laminar indications (planar indications oriented parallel to the vessel walls) are limited as shown below:

Component Thickness (mm)	Laminar Area (mm^2)
152	6,452
203	12,903
254	19,355
305	25,806

Finally, porosity in vessel welds is limited to 2% of the area of a radiograph of the weld, although porosity exceeding this level may be acceptable if an equivalent planar indication meets the restrictions given in Table 3. Standards for other components are similar to those for reactor vessels.

A brief review of service experience under the current ASME rules is given in the next section. That experience has generally been good.

Table 2. Examples of Maximum Sizes of Nonrelevant and Acceptable Rounded Indications from Preservice Nondestructive Examination[a]

Thickness, t (mm)	Maximum Size of Acceptable Rounded Indication, mm		Maximum Size of Nonrelevant Indication (mm)
	Random	Isolated	
<3.2	1/4 t	1/3 t	1/10 t
<3.2	0.79	1.07	0.38
<4.8	1.19	1.60	0.38
<6.4	1.60	2.11	0.38
<7.9	1.98	2.64	0.79
<9.5	2.31	3.18	0.79
<11.1	2.77	3.71	0.79
<12.7	3.18	4.27	0.79
<14.3	3.61	4.78	0.79
<15.9	3.96	5.33	0.79
<17.5	3.96	5.84	0.79
19–50.8	3.96	6.35	0.79
>50.8	3.96	9.52	1.60

[a]Source: *ASME Boiler and Pressure Vessel Code, Sect. III, Div. 1, Appendix VI*, Table VI — 1132-1, American Society of Mechanical Engineers, New York, 1977, p. 214.

Table 3. Allowable Planar Indications[a] from In-Service Nondestructive Examination of Reactor Vessel Welds[b]

Aspect Ratio (a/l)[c]	Surface Indications (a/t %)	Subsurface Indications (a/t %)
0	1.88	2.32
0.05	2.00	2.42
0.10	2.18	2.61
0.15	2.42	2.91
0.20	2.71	3.25
0.25	3.08	3.68
0.30	3.48	4.13
0.35	3.48	4.63
0.40	3.48	5.24
0.45	3.48	5.86
0.50	3.48	6.51

[a]$2a$ = flaw thickness;
l = flaw length;
t = component thickness.

[b]Source: *ASME Boiler and Pressure Vessel Code, Sect. XI, Div. 1*, Table IWB-3510, American Society of Mechanical Engineers, New York, 1977, p. 66.

[c]For intermediate ratios, linear interpolation is permitted.

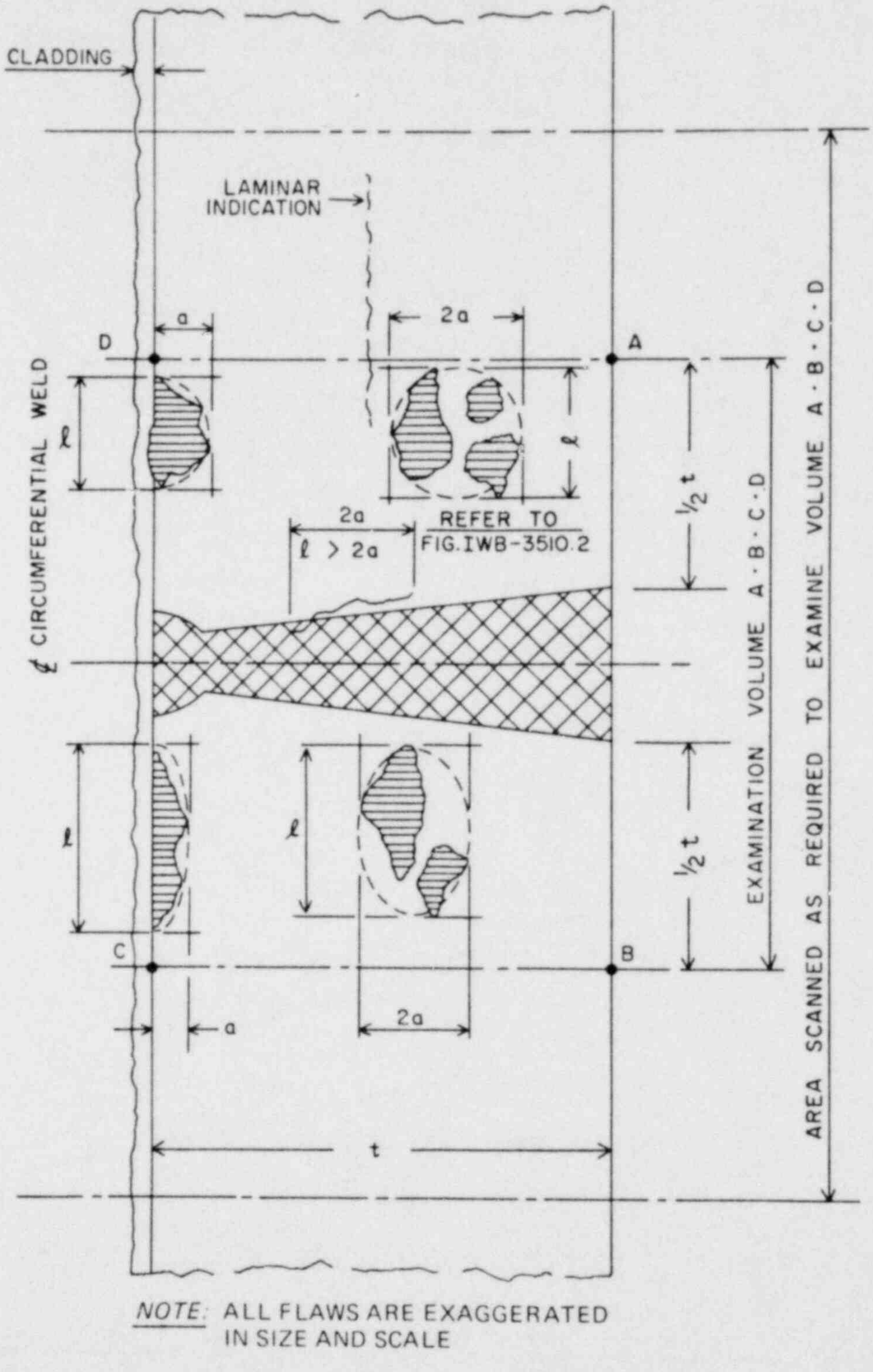

Fig. 10. Definition of Indication Dimensions for Indications Normal to Circumferential Stress. Reprinted fr..: *ASME Boiler and Pressure Vessel Code, Sect. XI, Div. 1*, American Society of Mechanical Engineers, New York, 1974, p. 41.

Various test results[95-100] on experimental pressure vessels have shown the ASME rules to be conservative, while Heald and Kiss[101] obtained the same result from tests of nuclear piping components, paying particular attention to welds. Figures 11 and 12 illustrate their results, in which failures in prototypic ferritic and austenitic pipe components were shown to occur well above the ASME-III design curve. (Of course, the times involved in the tests were much shorter than actual component design lives.)

However, Harrison[49,102] has warned that the ASME-III curves may not be conservative in all cases. Figure 13 shows a comparison of collected experimental data for butt welds with the ASME-III design curves for material with ultimate tensile strength (UTS) $\leq$ 550 MPa (80 ksi). At the nearest point the design curve yields a safety factor of only 1.25 on stress compared with the bottom of the experimental scatter band and about 1.8 compared with the middle of the scatter band. These welds were machined flush, so this apparent lack of conservatism was attributed[102] to

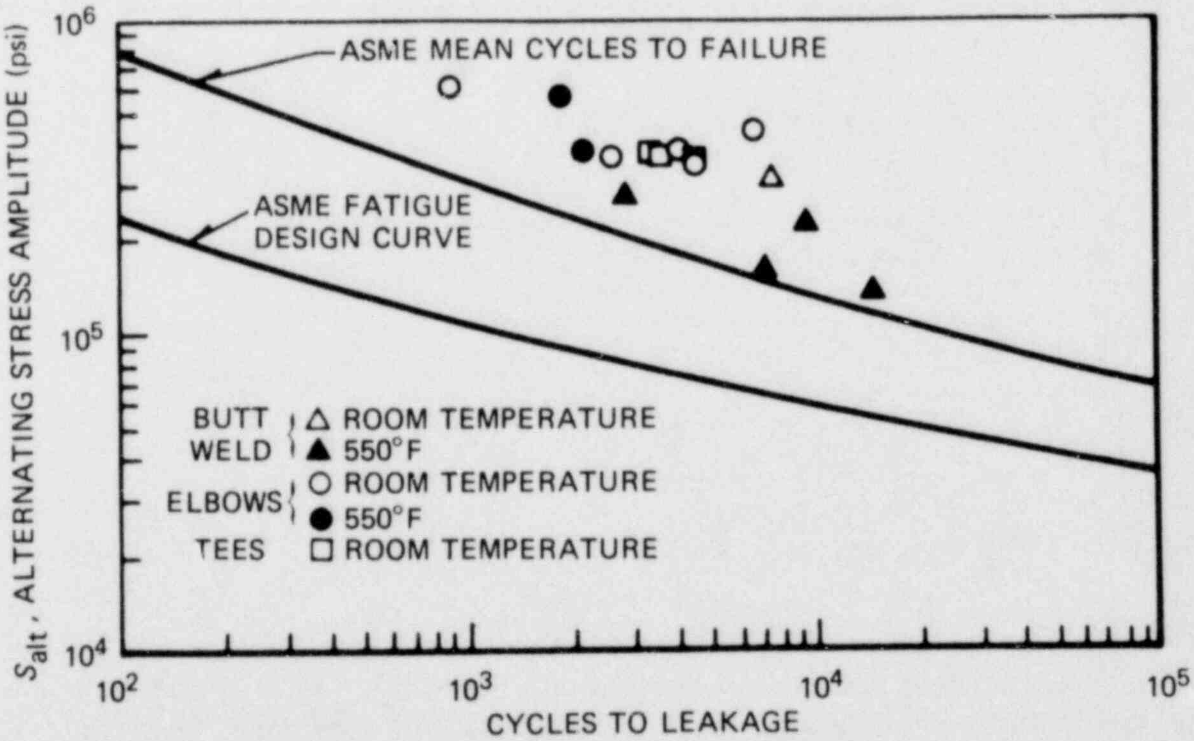

Fig. 11. Comparison of Fatigue Test Data on Stainless Steel Pipe Components with ASME Sect. III Fatigue Design Curves. Reprinted from: J. D. Heald and E. Kiss, "Low Cycle Fatigue of Nuclear Pipe Components," *J. Pressure Vessel Technol.* 96: 174 (August 1974)

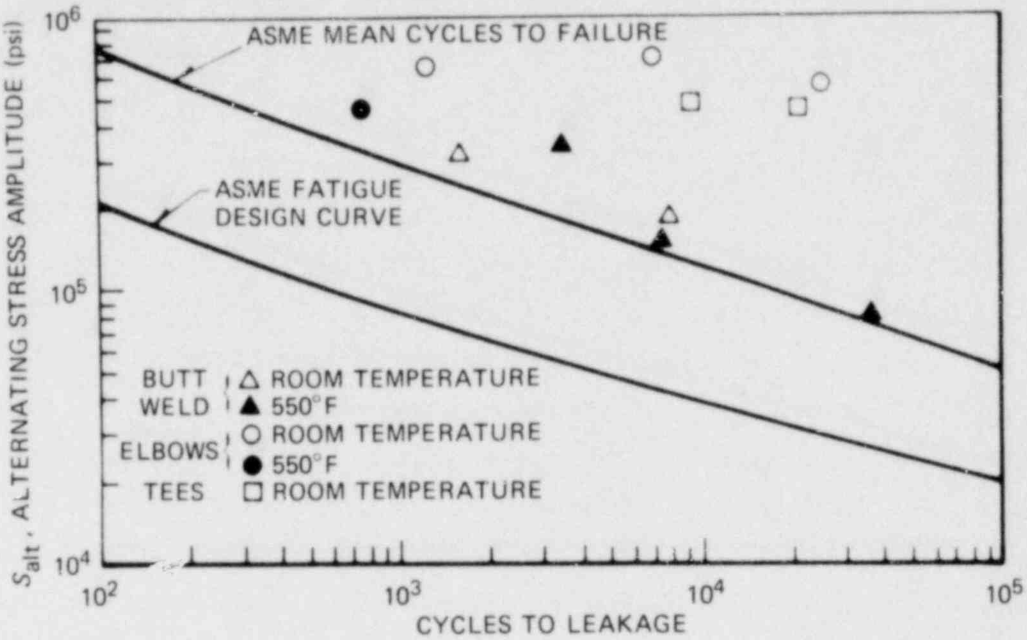

Fig. 12. Comparison of Fatigue Test Data on Carbon Steel Pipe Components with ASME Sect. III Fatigue Design Curves. Reprinted from: J. D. Heald and E. Kiss, "Low Cycle Fatigue of Nuclear Pipe Components," *J. Pressure Vessel Technol.* 96: 174 (August 1974).

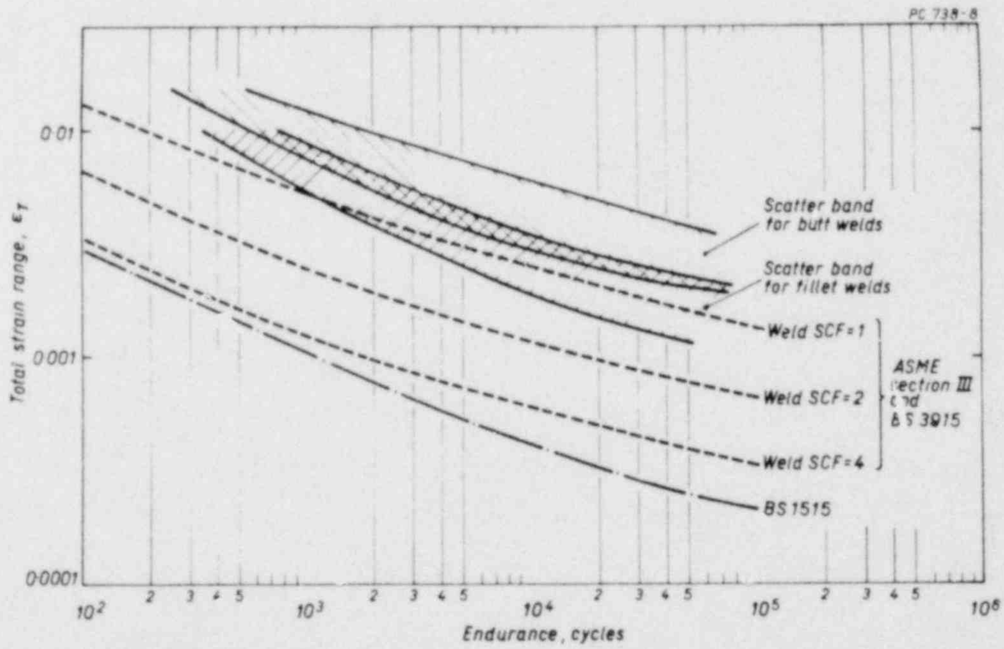

Fig. 13. Comparison of Fatigue Data on Butt Welds with ASME Sect. III Fatigue Design Curves. Reprinted from: J. D. Harrison, "Low Cycle Fatigue Tests of Nuclear Pipe Components," p. 207 in *Proc. Conf. Fatigue of Welded Structures*, The Welding Institute, Abington, England, 1971.

the effect of weld defects. The data shown in Fig. 13 were obtained from shielded metal-arc welds of several ferritic alloy steels, none of which are used in construction of nuclear pressure vessels and piping.

Harrison[49] has also called attention to Japanese[87] strain-controlled low-cycle fatigue data on weld specimens with slag inclusions. Those data show a marked effect of slag inclusions under strain-controlled loading to yield point stresses, which might resemble those occurring at either a pressure vessel nozzle or a similar point of stress concentration.

Little American work has been done toward quantification of the influence of weld defects on fatigue strength under conditions relevant to the nuclear industry. The prevailing American attitude has been typified by the work of Radziminski and Lawrence,[76] who expressed some pessimism concerning the possibility of assessing the effects of defects on crack initiation from a nondestructive examination of a weld joint.

On the other hand, workers at The Welding Institute (formerly the British Welding Research Association) have done a considerable amount of work[48,49,54,44,57,58,103,104] toward developing defect acceptance standards aimed at design on a fitness-for-purpose basis. Thus, they directly address the effect of various types and sizes of weld defects on fatigue strength. That work originally centered on the somewhat arbitrary definition of "quality bands" of fatigue strength, as illustrated in Figs. 14 and 15. Comparison of experimental data with these bands then allowed a rough correlation of defect size and type with expected fatigue strength (Table 4). Harrison[104] defines the results for main seams in pressure vessels. Later work[54] centered on a more statistical treatment of the data, but the essential aim is still a direct correlation among defect level, design stress, and design life. While the above work has not been used in the design of nuclear vessels at this writing, the approach still represents an interesting possibility. Lundin[39] reinforces this view, concluding that sufficient information is now available to begin a direct assessment of the influence of weld defects and that design codes should move in that direction or face the possibility of becoming antiquated in comparison with the state of the art of current technology.

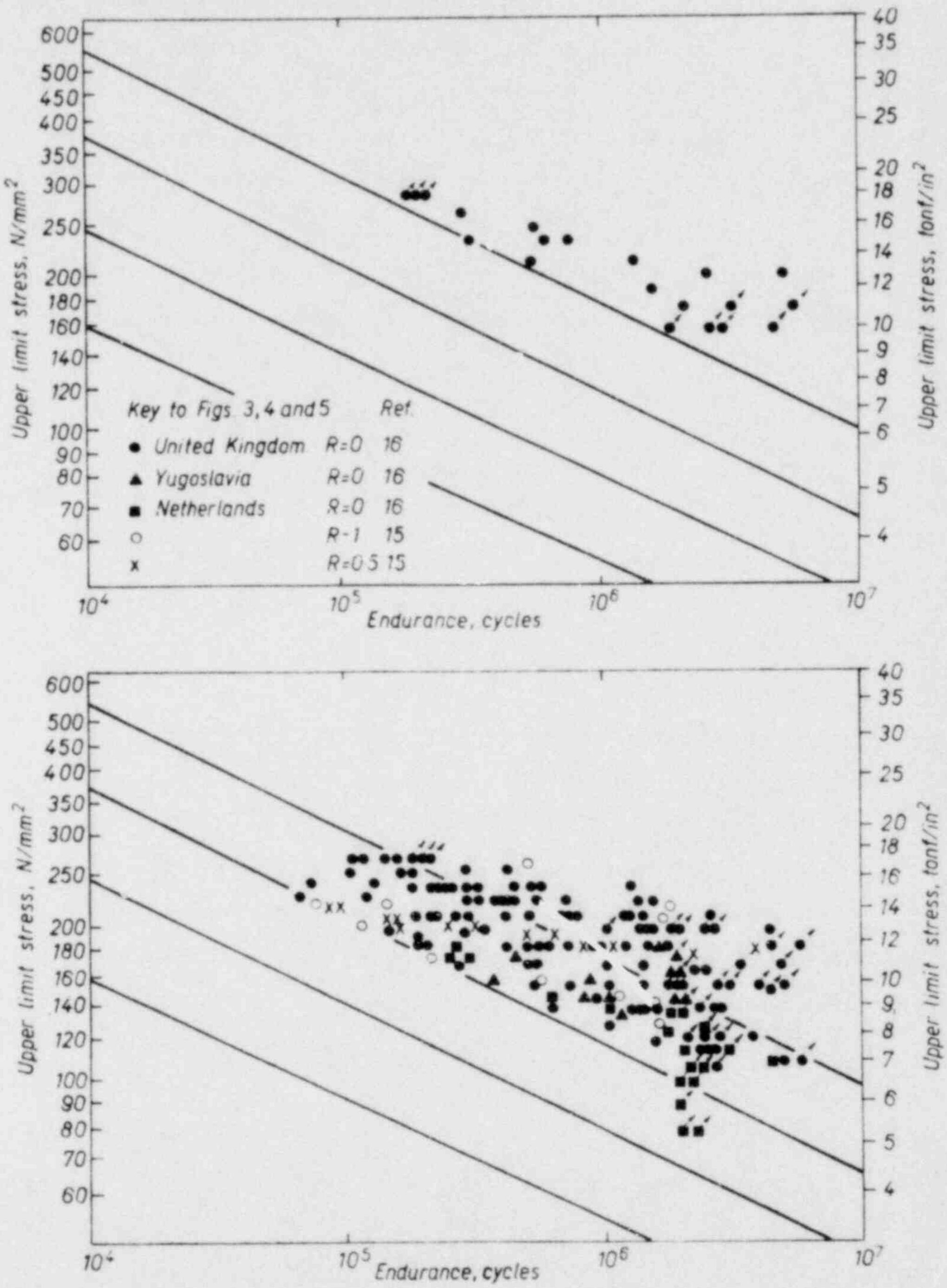

Fig. 14. Comparison of Fatigue Data for Welds with Slag Inclusions with Quality Bands Developed at The Welding Institute. Reprinted from: J. D. Harrison, "Basis for a Proposed Acceptance-Standard for Weld Defects. Part 2: Slag Inclusions," *Met. Constr. Br. Weld. J.* 4: 264 (July 1972).

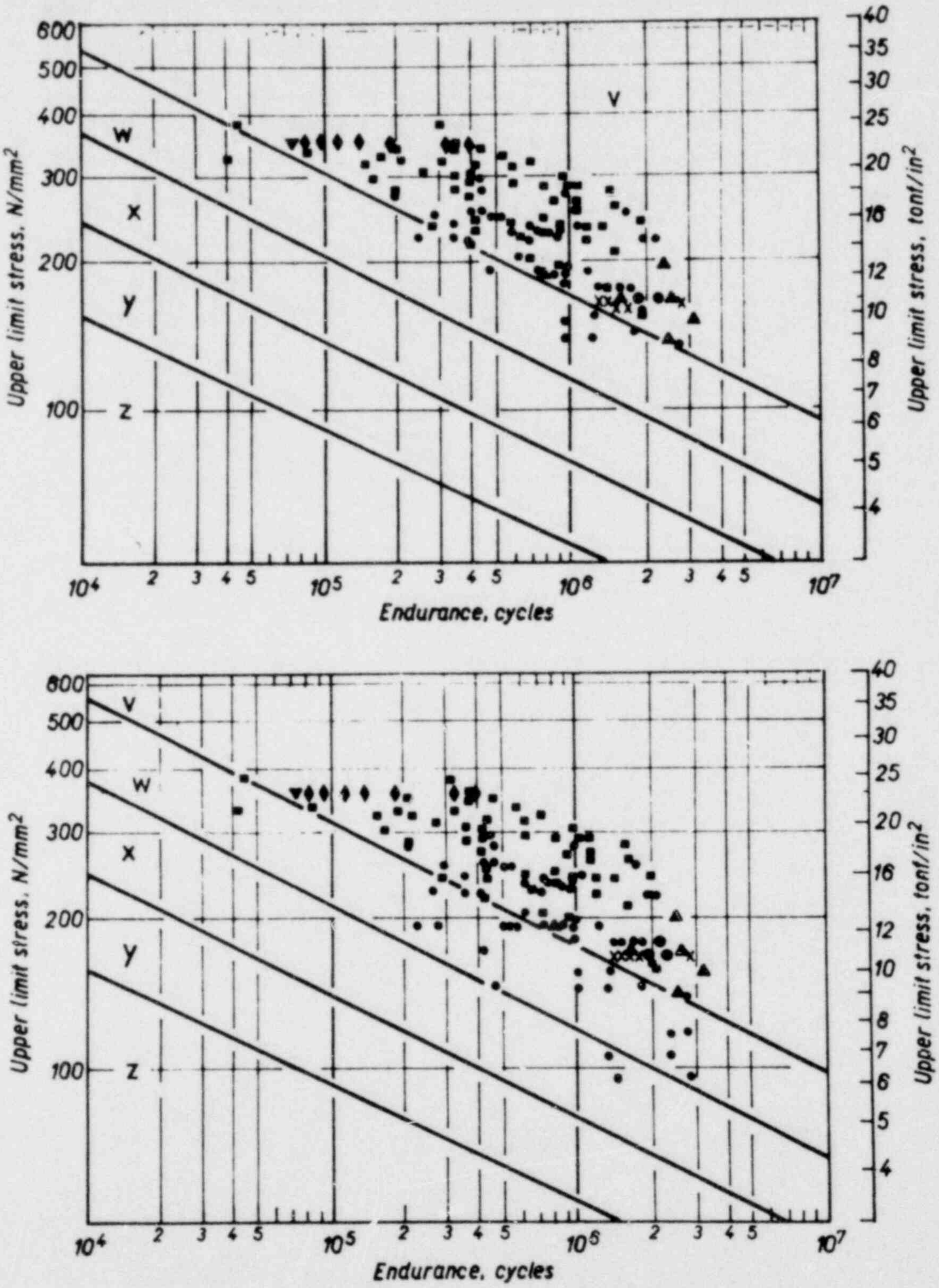

Fig. 15. Comparison of Fatigue Data for Porous Welds with Quality Bands Developed at The Welding Institute. Reprinted from: J. D. Harrison, "Basis for a Proposed Acceptance-Standard for Weld Defects. Part 1: Porosity," *Met. Constr. Br. Weld. J.* 4: 102 (March 1972).

Table 4. Maximum Allowable Defect Sizes and Volume Percentage Porosities for Various Weld Quality Levels[a]

Quality	Slag Inclusions, mm				Porosity, All Welds (%)
	Rutile Welds, as Welded	Low-Hydrogen Welds, as Welded	Stress Relieved Welds	As-Welded High-Restraint Welds	
V	0	0	0	0	0
W	1.5	5	5	0.75	3
X	10	25	No Max	4	8
Y	No Max	No Max	No Max	25	20
Z	No Max	No Max	No Max	No Max	20

[a]Minimum weld thickness 12.7 mm. Sources: J. D. Harrison, "Basis for a Proposed Acceptance-Standard for Weld Defects. Part 1: Porosity," *Met. Constr. Br. Weld. J.* 4: 99–107 (March 1972); and "Basis for a Proposed Acceptance-Standard for Weld Defects. Part 2: Slag Inclusions," *Met. Constr. Br. Weld. J.* 4: 262–68 (July 1972).

Welding Residual Stresses

Since various portions of a weld joint cool at different rates after welding, thermal gradients develop that result in the formation of residual stresses. Figure 16 schematically illustrates the patterns of residual stresses that develop for a common weld geometry.[105]

The exact effect of residual stresses on the fatigue life of welds is complex and difficult to predict, being different for different loading conditions, materials, etc.[106] This complexity partially results from the nature of weldment behavior and from several experimental complications, as listed below.[17]

1. The effects of residual stresses are often determined by testing before and after stress relief, but this heat treating process may introduce microstructural changes or result in the elimination of hydrogen[78,81] from the weld metal as well as removing residual stress.

2. Direct measurements of residual stress are often not available.

3. Cutting up a large weldment into small test specimens can result in a significant decrease in the level of residual stress.[24]

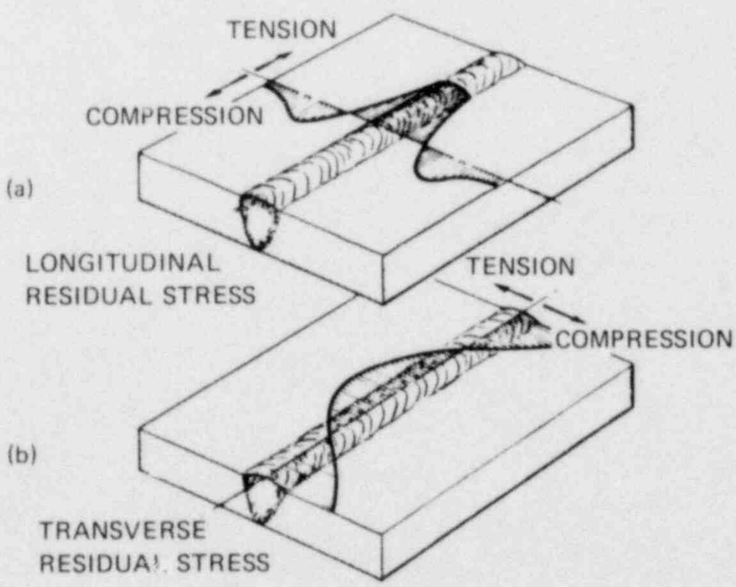

Fig. 16. Residual Stress Patterns Developed from Welding a Single-V Butt Weld. Reprinted from: B. Pollard and R. J. Cover, "Fatigue of Steel Weldments," *Weld. J. (Miami)* 51: 547-s (November 1972).

Despite the above complications, general conclusions can be drawn. Thermal stress relief of welding residual stresses generally has little or no effect on the fatigue behavior of steel weldments under pulsating tension loading.[31,32,107,108] However, stress relief has been found to improve fatigue strength under cyclic loading with an imposed compressive mean stress.[24,109]

In any event, all ferritic steel welds of concern to this investigation are required by the ASME Code to be given a postweld heat treatment. Thus, welding residual stresses should exist only for austenitic components. Still, the fatigue design curves in ASME-III contain corrections for the maximum effect of mean stress per the modified Goodman diagram.[110] Welding residual stresses are thus not expected to result in any design difficulties in nuclear pressure vessels and piping.

Microstructural Effects

During the welding process both the filler metal and the adjacent base metal are melted. Nearby base metal also sees very high temperature exposure up to the melting temperature. As a result significant microstructural changes occur, and the weld joint will of necessity be a region of inhomogeneous microstructure. Figure 17 illustrates typical variations in material properties across a weld joint. Thus, it is important to address the possible effects of microstructure on fatigue behavior of weldments.

If fatigue cracks initiate earlier in welds than in base metal, the above discussions show that the cause is usually either geometric stress concentrations or the presence of weld defects. Early initiation resulting from microstructural changes in the weld was not evident. Therefore, any microstructural effects would probably manifest themselves in crack propagation behavior. However, crack propagation behavior has generally been found[111,112] to be relatively independent of microstructure for materials such as those used in the construction of nuclear pressure vessels and piping. This result is supported by the general conclusion that crack propagation rates in weld metal, base metal, and heat-affected zones (HAZ) are similar.[23,112] Thus, based on available information, weld

Y-93367

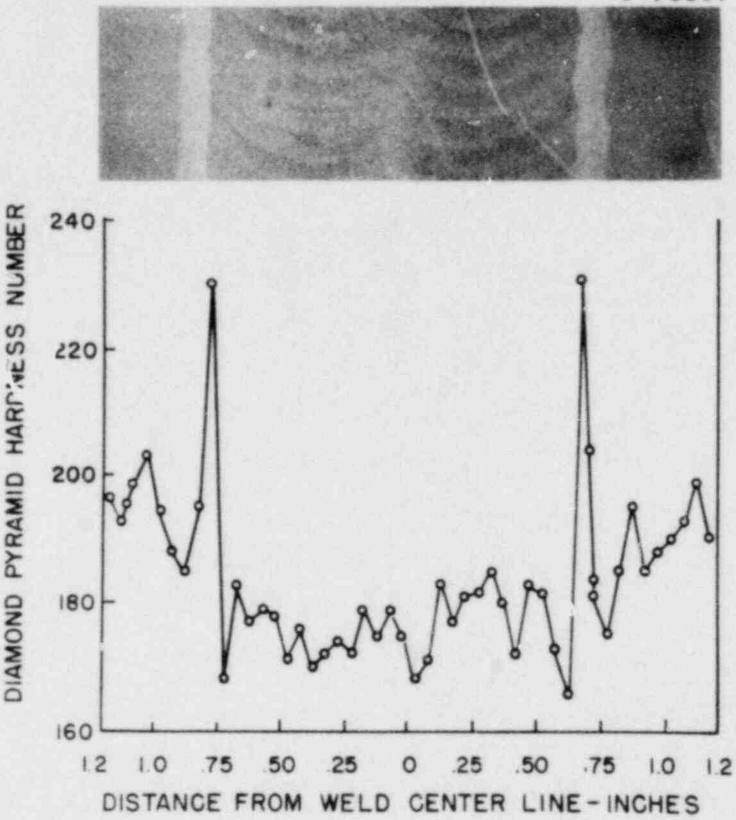

Fig. 17. Variation of Hardness Across a Submerged-Arc Weldment. (1 in. = 25.4 mm.)

microstructure would be expected to exert little influence on design allowable fatigue stresses. However, variable microstructure across the weld joint might result in differences in yield strength among base, HAZ, and weld metal that could cause localized strain concentration. However, no direct evidence of the magnitude of this effect is available.

Reheat Cracking

The problem of reheat cracking in the HAZs of pressure vessel steel weldments is treated here separately from weld defects in general because (1) these cracks cannot be detected by usual nondestructive techniques, and (2) there is no real evidence that the cracks represent defects that reduce fatigue strength. Such cracks have primarily been observed in the base metal HAZ underneath the stainless steel cladding on A508 Class 2 forgings, although they may also occur in the HAZs of structural welds.[113] Actually, the term "reheat" cracks may be misleading since there is some evidence[114] that the cracks occur as a result of liquation of low melting point constituents during welding. However, the prevailing view[114-116] is that the cracks do occur as a result of residual stress relaxation during stress relief heat treatment ("reheating").

Reheat cracks can occur either as microscopic grain boundary decohesions (Fig. 18) or as larger macrocracks. Their existence in A508 Class 2 HAZs is apparently fairly common[113] and probably results from the particular chemical composition of the HAZ.[115,117-118] Such cracks do not appear to be common in our other materials of interest. The reader is referred to refs. 113 through 120 for a general discussion on reheat cracking.

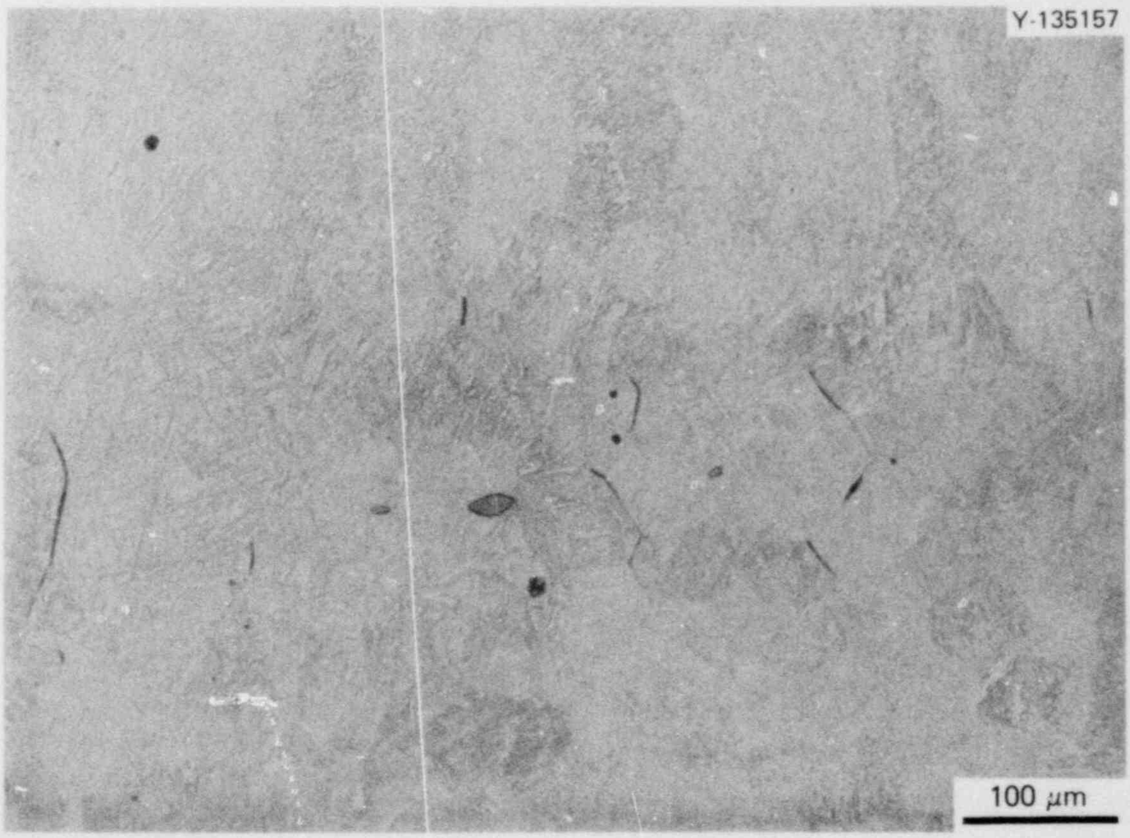

Fig. 18. Grain Boundary Decohesions in a SA508 Class 2 Heat-Affected Zone.

Our primary question is whether reheat cracks might cause a decrease in the fatigue resistance of the components involved. Fatigue data generated by Grotke et al.[121] on a high-heat-input submerged-arc structural weld are shown in Fig. 19. Axial fully reversed load-controlled fatigue tests were run at a frequency of 60 Hz on specimens whose axes were parallel to the weld, with the specimen cross section containing half weld metal and half HAZ-base metal. Five of the eight specimens tested were found to contain reheat microcracks oriented perpendicular to the specimen axis. There was no indication of any loss in fatigue strength resulting from these cracks over the limited range tested.

In a study of artifically induced flaws in the undercladding in a Russian reactor pressure vessel steel, Rahka and Forstén[122] expressed concern at the conservatism of the current ASME-III and ASME-XI fatigue and crack growth rules, although their results compared well with the

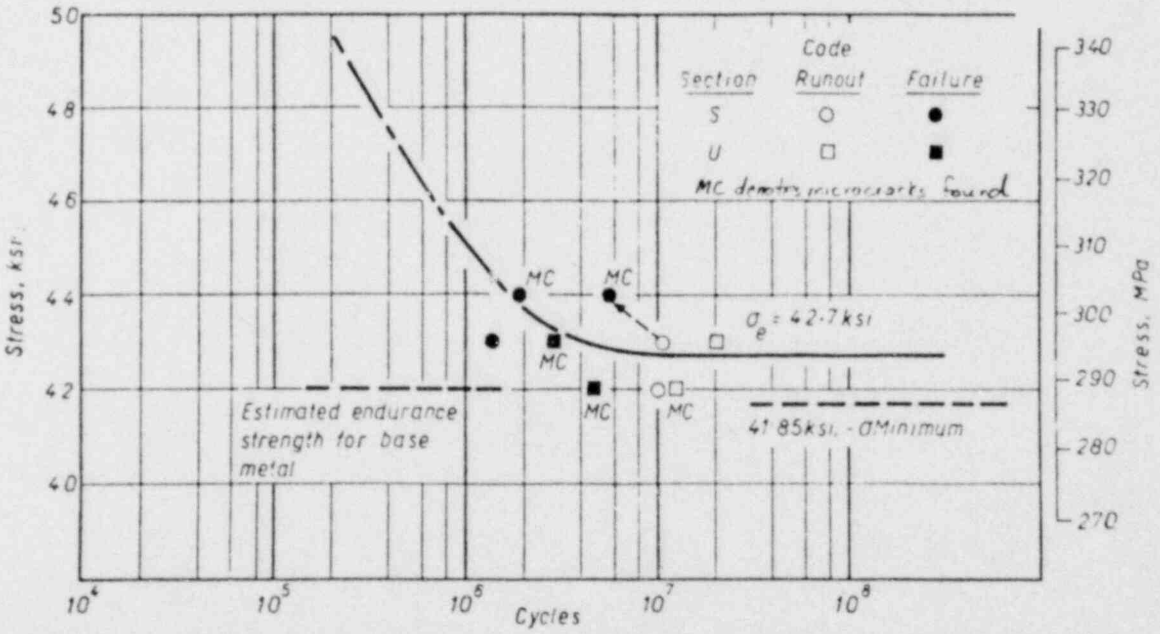

Fig. 19. Results of Fatigue Tests on Weld Specimens With and Without Reheat Cracks. MC denotes test on specimen in which reheat cracks were found after testing. Reprinted from: A. Dhooge et al., "A Review of Work Related to Reheat Cracking in Nuclear Reactor Pressure Vessel Steels," *Int. J. Pressure Vessels and Piping* 6(5): 329–409 (September 1978).

acceptance standard work of Harrison et al.[48,49,55] Their results show that Code-allowable defects could result in a degradation of the actual fatigue properties of their material. However, this steel was not representative of U.S. pressure vessel steels, and the artificially induced flaws may not have been representative of those to be expected in actual components.

More relevant are crack growth studies by Ayres et al.[114] and by Mager et al.[123-124] Table 5 summarizes predictions from the Ayres et al.[114] study. For the worst case studied, a 4-mm-deep crack in the undercladding would be expected to extend by only 1.15 mm in 40 years. The analysis in the Mager, Landerman, and Kubit[123] report for an internal wall surface flaw (though not specifically a reheat crack) predicted that a 3.2-mm-deep crack would grow by only 0.1 mm in 40 years. These calculations were revised by Mager et al.[124] to include frequency effects on crack growth rates and showed that a 3.2-mm-deep crack in the top head closure would grow by 0.51 mm in 40 years when exposed to pressurized-water reactor (PWR) water. Table 6 summarizes the results from the above two reports[123,124] for various critical locations in the vessel (Fig. 20).

Surveying available information on growth of cracks in the undercladding, Dhooge et al.[115] and Dolby and Saunders[120] drew the following conclusions.

1. Cracks in the undercladding may grow either into the cladding or into the base metal, but the amount of growth into the base metal would be small and would probably not extend beyond the HAZ.

2. Cracks growing into the cladding could conceivably break the cladding surface, exposing the crack to the reactor environment. However, the above-cited work[122-124] assumed surface cracks and ignored the cladding. Thus, while penetration of the cladding could lead to accelerated crack growth rates, the results in Tables 5 and 6 would still be applicable.

3. Under normal, upset, and test conditions reheat cracks would not be expected to grow to critical dimensions. However, under certain fault conditions (such as loss of coolant), it is conceivable that a crack in

Table 5. Summary of Babcock and Wilcox[a] Defect Significance Calculations

Assumed Crack Position, Size, and Orientation	Calculated Acting Stress on Inside Surface		Assumed Toughness		Critical Crack Size (mm)	Number of Cycles to Grow 4-mm-Deep Crack to 10.2 mm Depth	40-Year Crack Growth (1440 Cycles) (mm)
	(MPa)	(ksi)	($MPa\sqrt{m}$)	($ksi\sqrt{in.}$)			
Continuous circumferential 4-mm-deep crack at vessel core belt line	850	12.3	3120	90	346	52,559	0.13
Longitudinal crack, 12 mm long, 4 mm deep at vessel belt line	1910	27.8	3120	90	110	16,877	0.53
Continuous circumferential 4-mm-deep crack close to bottom closure to shell joint	2140	31.2	4860	140	130	7,817	1.15
Longitudinal crack 12.5 mm long, 4 mm deep close to bottom closure to shell joint	1910	27.8	4860	140	244	24,665	0.36

[a]Source of data: P. S. Ayres et al., *Study of Intergranular Separations in Low-Alloy Steel Heat-Affected Zones Under Austenitic Stainless Steel Weld Cladding*, BAW-10013 (December 1971).

Table 6. Results of Crack Growth Analysis for Three Internal Wall Surface Flaws at Critical Vessel Locations

Location	Location Thickness (mm)	Initial Flaw (mm)	Crack Growth, mm	
			Analysis 1[a]	Analysis 2[b]
Top Head	157	3.18	0.076	0.508
		6.35	0.203	1.219
		12.70	0.508	6.172
Nozzle Shell Course	234	3.18	0.102	0.305
		6.35	0.254	1.219
		12.70	0.838	5.131
Belt Line	195	3.18	0.000	0.254
		6.35	0.051	1.041
		12.70	0.102	2.083
Lower Head	121	3.18	0.000	0.127
		6.35	0.051	0.457
		12.70	0.127	2.108

[a]Source: T. R. Mager, E. Landerman, and C. Kubit, *Reactor Vessel Weld Cladding — Base Metal Interaction*, WCAP-7733 (July 1971).

[b]Source: T. R. Mager, J. D. Landes, D. M. Moon, and V. J. McLaughlin, *The Effect of Low Frequencies on the Fatigue Crack Growth Characteristics of A533 Grade B Class 1 Plate in an Environment of High-Temperature Primary Grade Nuclear Reactor Water*, WCAP-8256, HSST Tech. Rep. 35 (December 1973).

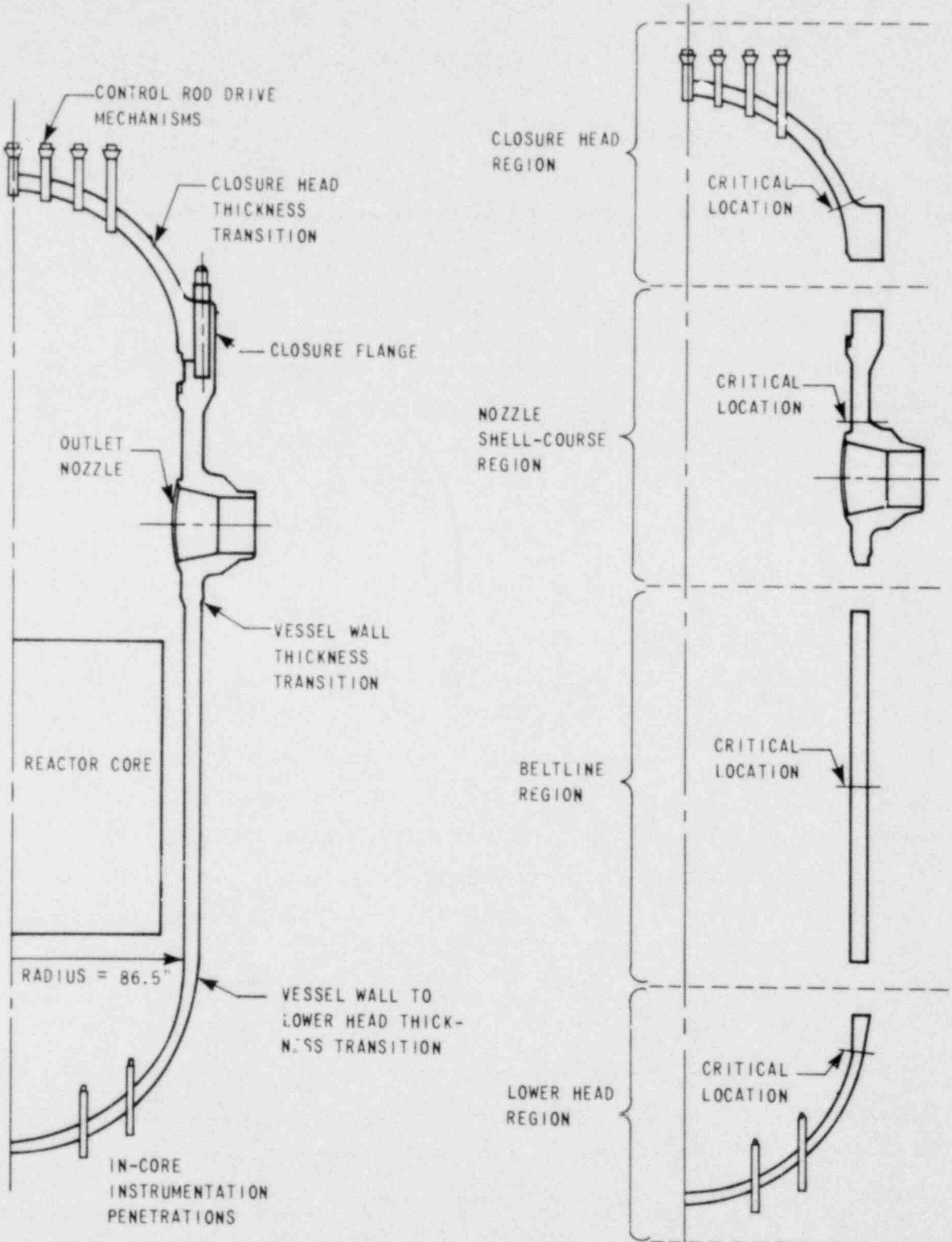

Fig. 20. Critical Vessel Locations in Crack Growth Analyses. Reprinted from: T. R. Mager, J. D. Landes, D. M. Moon, and V. J. McLaughlin, *The Effect of Low Frequencies on the Fatigue Crack Growth Characteristics of A533 Grade B Class 1 — Plate in an Environment of High-Temperature Primary Grade Nuclear Reactor Water*, WCAP-8256, HSST Tech. Rep. 35 (December 1973).

the undercladding would initiate a fracture, although this fracture would probably arrest near the vessel outer wall, where stresses approach zero.

4. More testing and analysis are needed to fully assess the possibilities of growth of reheat cracks by fatigue, including the risk of vessel rupture resulting from linkup of reheat crack networks.

Crack Propagation

All the above discussions have been concerned with fatigue crack initiation because most evidence indicates that the major possible difference between weld and base metal would be in initiation. Futhermore, the ASME Code rules are currently based primarily on crack initiation criterion. In-service examinations and evaluations can include crack growth considerations per ASME-XI, but no actual component design is done considering crack propagation aspects.

Still, most recent investigations[18,19,23,65,125,126] have concluded that a reasonable prediction of the fatigue life of welds must include a separate assessment of fatigue crack initiation and crack growth aspects. Furthermore, some investigators[51,76,127] have expressed that fatigue life of a weld should be reliably estimated for crack growth only and that the initiation period cannot be estimated conservatively, except by considering it as nil. However, more detailed investigations have found that the initiation life may not be negligible, especially in the absence of geometric stress concentrations.[65,128,129] In fact, the initiation portion of life can vary considerably and may easily be half the total life in a high-cycle situation.[19]

Crack propagation data for weld metal and HAZs have been treated by techniques of linear elastic fracture mechanics similar to those used for base material.[30] In essence, the data are treated by the Paris-Erdogan[130,131] law,

$$\frac{da}{dN} = A(\Delta K)^n \, , \tag{1}$$

where

ΔK = stress intensity factor range,

$\frac{da}{dN}$ = rate of crack extension per cycle.

Extensive results have been obtained concerning the fatigue crack propagation behavior of nuclear pressure vessel steels.[124,132-137] The conclusion from that work and from other work on similar materials[111,112,138] was that crack growth rates in weld metal and HAZs are probably no greater than those in base metal. Figure 21 shows typical results. Still, it should be emphasized that these results can be significantly influenced by such factors as environment, frequency, load ratio, and specimen geometry. Thus, while published results indicate no tendency

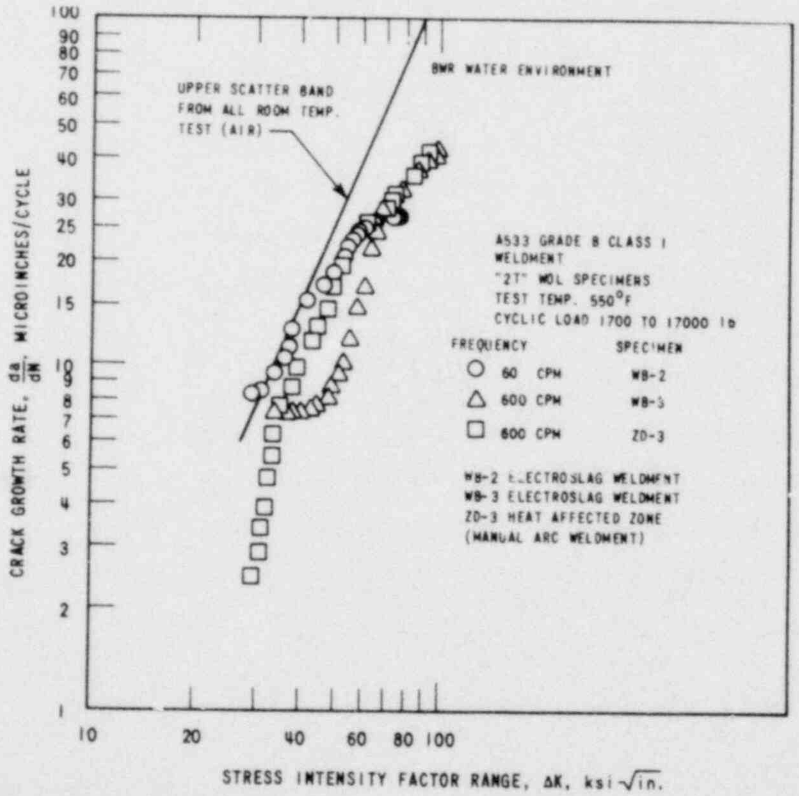

Fig. 21. Comparison of Fatigue Crack Propagation Data for Weld Metal and Heat-Affected-Zone Specimens with Behavior of Base Metal. Reprinted from: T. R. Mager, "Fatigue Crack Growth Characteristics of Nuclear Pressure Vessel Grade Materials," p. 47 in *Heavy-Section Steel Technology Semiannu. Prog. Rep. Feb. 28, 1971*, ORNL-4681 (December 1971).

for accelerated crack growth rates resulting from the presence of weld, it is not inconceivable that a combination of circumstances exists under which such acceleration would occur.

SUMMARY OF DESIGN AND OPERATING EXPERIENCE OF REACTOR PRESSURE VESSELS AND PIPING COMPONENTS

Pressure vessel failures have occurred in various industries, but the philosophy that these failures primarily result from the inherently inferior characteristics of weldments is an unfounded oversimplification.[111] Failures that have occurred have been documented in several recent compilations from service both in the U.S. and abroad.[139-141] Phillips and Warwick[140] and Bush[141] reached the conclusions that annual probability of service failures in pressure vessels in general was about 10^{-5}. However, for nuclear vessels Smith and Warwick[139] determined this probability to be more in the range of 10^{-2} to 10^{-3}. These results were influenced by the small sample size involved and by some confusion over the exact definition of failure used by various organizations responding to their survey. At any rate, none of the reported failures dealt with the vessel itself, with most being concerned with leaks or cracks in piping and nozzles. Only one failure was reported as being caused by a weld discontinuity, and none of the incidents resulted in sudden catastrophic failure, release of radiation, or loss of life.

To provide more detailed up-to-date information on service experience, the files of the Nuclear Safety Information Center (NSIC) were accessed for reported weld failures in U.S. nuclear power plants from 1965 to 1978. Over 360 entries are summarized in Table 7. Approximately 25 to 30% of the total number of failures of weldments resulted from fatigue-related phenomena. Of these, none were reported to have occurred in the reactor vessel itself. Most are associated with various instrument lines, feedwater piping systems, etc. By far the largest contribution to the number of failures was from pump vibrations. Interestingly, several components experienced repeated failures after repair. A detailed summary of the various incidents is given in Appendix B. None of the incidents were interpreted as being cause for alarm within the scope of this investigation.

Table 7. Summary of Reported Fatigue Failures in Light-Water Reactors[a]

Year	Total Reported Incidents	Fatigue Failures						Fraction Due to Fatigue
		Reactor Type[b]		Failure Cause				
		PWR	BWR	Fatigue	Vibration	Thermal	Pressure	
1965–1972	15							0/
1973	21	6	1	1	6			7/21 = 0.333
1974	47	7	5	3	8		$\sim$1	12/47 = 0.255
1975	75 + 2	14	7	2	18		$\sim$1	21/77 = 0.272
1976	65	18	2	6	12	$\sim$2		20/65 = 0.308
1977	82	16	4	4	15	$\sim$1		20/82 = 0.244
1978	56 + 1	13	1	3	8	$\sim$2	$\sim$1	14/57 = 0.246
TOTAL	361 + 3	74	20	19	67	$\sim$5	$\sim$3	94/364 = 0.258

[a]Source: Nuclear Safety Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee, January 9, 1979.

[b]PWR = pressurized-water reactor, BWR = boiling-water reactor.

Although not directly related to service experience, the work of Jerram[38] is probably worth mentioning here. A compilation of pressure vessel fatigue test data was compared with existing U.S. and British codes, as shown in Figs. 22 and 23. From these results Jerram concluded that the ASME-III fatigue design curves may not be conservative, even for parent metal. Figure 22 shows very little margin of safety between many of the test failures in base material and the design curves. He recommended additional safety factors of 1.25 on stress or 3.75 on cycles to introduce a more comfortable margin. However, it should be noted that these were not nuclear vessels and were not designed to the stringent requirements of ASME-III.

Figure 23 compares weld metal failures from Jerram's compilation[38] with applicable design curves. From this comparison Jerram concluded

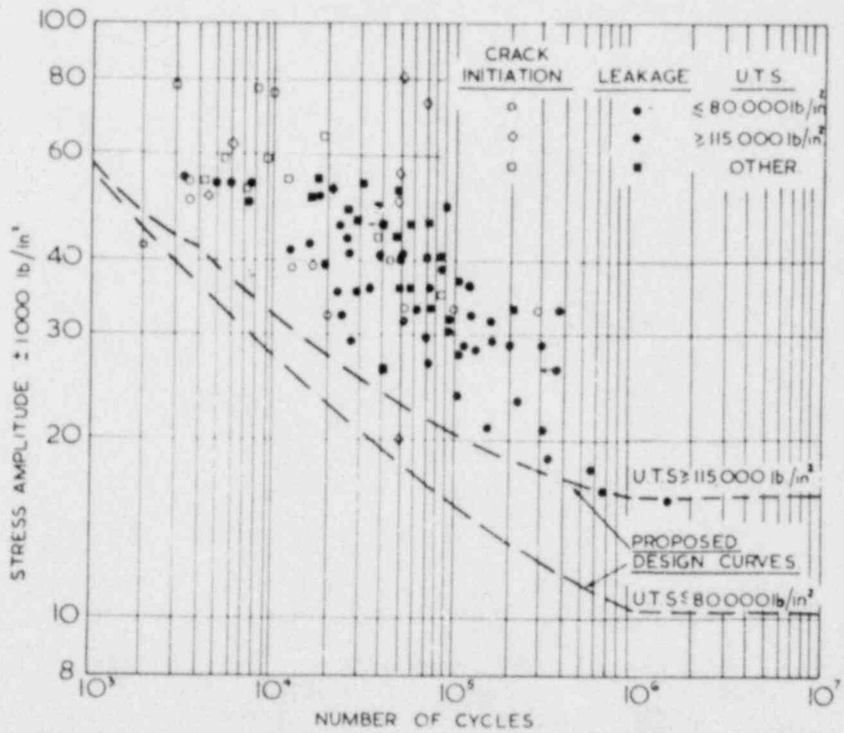

Fig. 22. Comparison of Pressure Vessel Fatigue Tests Failed in Base Material with ASME-III and BS-1515 Fatigue Design Curves. Reprinted from: K. Jerram, "An Assessment of the Fatigue of Welded Pressure Vessels," p. 1312 in *1st Int. Conf. Pressure Vessel Technol., Pt. II, Materials and Fabrication*, American Society of Mechanical Engineers, New York, 1969. (1 lb/in.^{2} ≅ 6.895×10^3 Pa.)

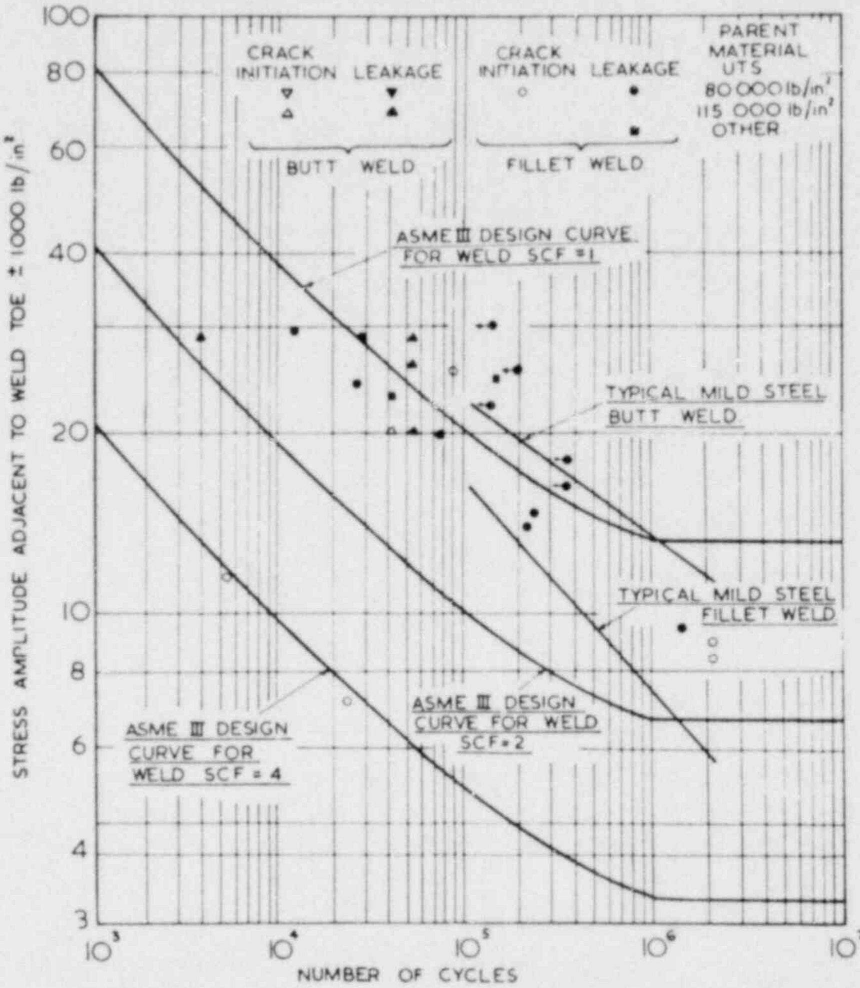

Fig. 23. Comparison of Pressure Vessel Fatigue Tests Failed at Weldments with ASME-III Fatigue Design Curves by Using Various Fatigue Strength Reduction Factors. Reprinted from: K. Jerram, "An Assessment of the Fatigue of Welded Pressure Vessels," p. 1314 in *1st Int. Conf. Pressure Vessel Technol., Pt. II, Materials and Fabrication*, American Society of Mechanical Engineers, New York, 1969. (1 lb/in.^{2} ≅ 6.895 × 10^3 Pa.)

that, if anything, the current ASME-III rules for use of fatigue strength reduction factors at welds are overconservative. Thus, while Jerram has questioned the conservatism of ASME-III, the validity of the comparison upon which he based his question is not clear. In any case his primary concern was not related to problems involving fatigue of weldments, although he did also question the use of variable ultimate tensile strength fatigue curves for weldments since the relationship between weld fatigue strength and ultimate tensile strength is not necessarily exact.[16,17,32] However, the tensile strength effect on the ASME-III fatigue design curves is not strong, and the curves for different tensile strengths do cross over (Fig. 2). Therefore, no changes in current design procedures appear to be dictated for ultimate tensile strength.

Decock[142] presented some similar comparative results. He found that the safety margin on fatigue failure varied as a function of cyclic life, as illustrated in Fig. 24. (Measured stress concentration factors have been incorporated into the plotted stress amplitude.) He noted that for cyclic lives of less than 10^4 cycles his data for a mild steel showed practically no safety margin in the ASME design curves. However, he considered that margin to be sufficient for life in excess of 5×10^4 cycles.

A survey by Mayfield, Rodebaugh, and Eiber[143] concluded that the ASME-III Code evaluation procedures were generally satisfactory when $S_n > 3S_m$ (elastic-plastic conditions). They recommended that current

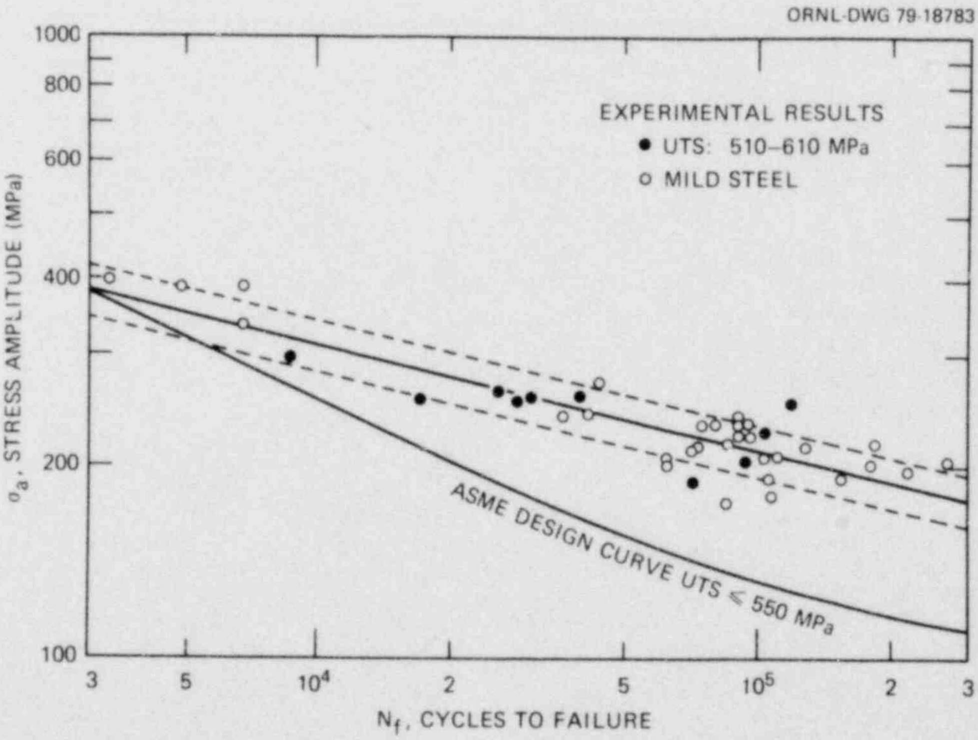

Fig. 24. Fatigue Results and Comparison with ASME Design Curve. Source: J. Decock, "Determination of Stress Concentration Factors and Fatigue Assessment of Flush and Extended Nozzles in Welded Pressure Vessels," p. 821–34 in *2d Int. Conf. Pressure Vessel Technol., Pt. II*, American Society of Mechanical Engineers, New York, 1973.

procedures for calculation or cyclic loading should be reviewed along with the procedures for defining the fatigue strength reduction factor, K_e, for elastic-plastic analysis. Finally, they demonstrated that the Code analyses were generally more conservative for ferrite steels than for austenitic stainless steels.

In the late sixties and early seventies a series of fatigue cracks were found in the undercladding in boiling-water reactor pressure vessels.[144] These cracks occurred at the interface between the clad and the inlet nozzle forging. It was later determined that the cause of these cracks was attributed to thermal cycling. A series of investigations and experimental programs verified this assumption, and subsequently the General Electric Company requested that the cladding be removed (ground) from the ASTM A508-2 forging material at the cold feed water inlet nozzle. The clad was left on the rest of the pressure vessel (A533B Class 1 plate material). There were no reported incidents of a through-crack being found in these vessels.

Available published information and documentable information obtained privately indicate that service experience in nuclear pressure vessels and piping has been satisfactory with respect to fatigue of weldments. However, there have been and continue to be numerous fatigue failures in weldments in other industries.[145] Although specific supporting data are apparently not available, W. J. O'Donnell of O'Donnell and Associates, Inc., also cites[145] from past experiences that when thermal fatigue is imposed on a weldment by the flow of fluids of changing temperature over the surface, the heat-affected zone will tend to exhibit premature fatigue cracking. He believes that welds may have inherently inferior fatigue properties for these reasons: (1) fatigue damage incurred from thermal cycling in the actual process of multipass welding and (2) additional strains developed at weldments from "metallurgical notches" or from changes in material properties across the weld joint. However, these possibilities are hard to quantify based on available data. However, since thermal transient stresses and stresses resulting from through-the-wall thermal gradients can reach the plastic range, the metallurgical notch effect should be explored further, although postweld heat treatment should greatly decrease the importance of this effect.

SUMMARY OF CURRENT RESEARCH

World-wide activities regarding fatigue evaluations and associated investigations include both experimental studies of reactor pressure vessel and piping steel and development of analytical techniques that are used to predict and analyze the performance of these components. Table 8 shows a compilation of the summary[146] of some research programs for fracture toughness, crack growth, and stress analysis for reactor pressure vessel steels being conducted internationally. Although prepared in 1977, this summary reflects the current concentrations of effort and illustrates the magnitude of attention being directed toward the investigation of pressure vessel steels.

Table 8. Summary of Research Programs for Fracture Toughness, Crack Growth, and Stress Analysis for Reactor Pressure Vessel Steels

Title and Aims	Location	Sponsor
Crack growth in LWR PV at complex geometries. Development LEFM computing procedures to predict crack growth at nozzle corners. Verification of procedures using codel tests. Determination of growth and related toughness values using U_{1c}. Evaluation of acoustic emission for location & detection and growth monitoring.	Delft University, Holland	Dutch Minister of Economic Affairs
Phase II of (1) above. Investigate crack growth in complex geometries. Extend Phase I to include Thermal stresses, biaxial loading. Elasto-plastic at complex geometries and biaxial loading.	Delft University, Holland	Dutch Minister of Economic Affairs

Table 8. (Continued)

Title and Aims	Location	Sponsor
Aim to study the application of J concept for predicting the behavior of cracks in areas of complex geometries, notably nozzle corner areas.	Delft University, Holland	Dutch Minister of Economic Affairs
Residual stress at girth-butt welds in PV and pipes. Develop analytical methods to calculate magnitude direction and distribution of residual stress.	Battelle Columbus Labs, USA	NRC
To characterize crack arrest as applied to LWR esp RPV. Experts to validate arrest theory, dynamic analysis, test procedures and data acquisition.	Battelle Columbus Labs, USA	NRC/EPRI
HSST Program Crack growth rated in LWR chemistry. Irrad effects on upper band for weldments. Method of analysis to predict crack propagation following LOCA's. Plan, analysis of model test experiments. Tests on concrete press stressing wires. Plan foreign research.	Oak Ridge Nat Lab, USA	NRC
To characterize crack arrest in terms of measurable physical properties. Correlate with crack arrest test. Aim code adoption.	University of Maryland, USA	NRC
Structural integrity of Water Reactor pressure boundary components. Develop dynamic testing practice and verify K_{1c}. Correlate C_v and K_{1c} upper shelf. Develop J-integral testing.	NRL, USA	NRC

Table 8. (Continued)

Title and Aims	Location	Sponsor
Larger capacity cyclic crack growth rate, esp welds, both unirradiated.		
Investigate irradiation effect and update Regulatory Guide 1.99.		
Observation benefits of warm prestressing. Provide technical basis for disregarding biaxiality of stresses during warm prestressing.		
Destructive tests, static, dynamic, cyclic, conventional and fracture mechanics tests.	MPA, Stuttgart, Germany	Germany
Small spec up to 10 in. - Sectional behaviour, esp local strength and fracture toughness.		
Large size - integral behaviour of fracture toughness tests on specimen, compact tensile, round and wide plate specimens. Specimens will include lower bound residual stress state.		
Intermediate vessels 2000 - 3500 mm D, up to 150 mm thick. Strength and fracture crack propagation, crack arrest and use of US and AE.	MPA, Stuttgart, Germany	Germany
Experimental assessment of stress and failure analysis of RPV, before and after test and in a defined weakened condition. Aim to do a comparison of measured and theoretical safety margins.	MPA, Stuttgart, Germany	German HTR Prog.

Table 8. (Continued)

Title and Aims	Location	Sponsor
Analogous tests to previous program will be done on pipework and selected piping parts including castings. Tests will be done on a rig and will look for leak-before-break conditions and crack-arrest behaviour.	MPA, Stuttgart, Germany	German HTR Prog.
The material is essentially X-10 austenitic and in the Phase I tests, ie abnormal loading, emphasis will be behaviour at cyclic cracks. Parallel component tests to the pipework tests are planned.		
For the final failure, again specifically weakened structures, ie weld cracks. The geometry will be such that a direct comparison will be possible with the unweakened pipe.		
The final test will be crack initiation in the loop near an 'S' pipe where crack-arrest conditions will be investigated.	MPA, Stuttgart, Germany	German
Rough and detailed specification research programme 'Structural Integruty of Components' (RS192)	MPA, Stuttgart, Germany	BMFT, Germany
Objects give basic information on quantification of safety margins.		
Determination of fracture mechanics safety criteria for Elastic plastic behaviour of metals (RS90)	KWU, Erlangen, Germany	BMFT, Germany
Use of COD, J integrals and R curve to extend elastic-plastic behaviour.		

Table 8. (Continued)

Title and Aims	Location	Sponsor
Fracture properties of through cracks (RS 102) Extension of part-through cracks in thick-walled vessels & tubes under fatigue loading on basis of fracture mechanics. (Cracking in plates and tubes of 22 Ni Mo Cr27.)	IFRM, Frieburg, Germany	BMFT, Germany
Investigation of crack initiation and arrest (RS 102-12) Use of DCB specimens Model dynamic effect on crack arrest. Measure crack initiation and toughness on 22 Ni Mo Cr 37. Measure crack velocity prior to arrest. (Determination of values of K_{Ia} dyn K_{Ia} stat; corrections with ASME XI-A5300.)	IFKM, Frieburg, Germany	BMFT, Germany
Use of COD/CTOD fracture criteria in the assessment of component strength (RS 102-13). Measure COD in component and SEN specimen on two mods of 22 Ni Cr Mo 37 as a function of thickness and crack length. (Determination of CTOD and J integral for increasing loads for a series of CT and SEN specimens.)	IFKM, Frieburg, Germany	BMFT, Germany
Fracture analyses at nozzle interactions (BROS) Programme covers Early detection of defects Detailed surveillance of growth Establish predictions of further growth (ASTM A508 Cl. 2 material)	Rotterdam Dry Dock, RTD, Delft University, Holland	Dutch Minister of Economic Affairs

Table 8. (Continued)

Title and Aims	Location	Sponsor
Dynamic - Fracture mechanics.	RisØ, Denmark	DAEC, Denmark
Model used is a crack propagating in low toughness materials surrounded by tough material conditions for propagation in tough material considered.		
Effect of crack length, propagating speed energy will be attempted.		
Fracture mechanics in ductile materials.	PISA-University, Italy	CNEN and CNR, Italy
Influence of Environment on fatigue crack growth	TNO - Metackinstitut, Holland	Dutch Ministry of Social Affairs
Influence of ageing on fracture-related material props. (Literature Survey)		
Literature survey of matl parameters which are necessary for the diagnosis of defect location with AE techniques.	TNO - Metackinstitut, Holland	Dutch Ministry of Social Affairs
Elastic-plastic fracture mechanics	A Atomenergi Studsvik, Sweden	Sweden
For nuclear pressure vessels - development of methods for determination of fracture toughness of pressure vessel steels at high temperature.		
Application of method to assess a base for fracture mechanics analyses - Project P27.	A Atomenergi Studsvik, Sweden	Sweden
Elastic-plastic fracture mechanics for reactor pressure vessels - Phase 2 - Project P51.		

Table 8. (Continued)

Title and Aim	Location	Sponsor
Applied to nuclear pressure vessels. Complementary investigation on plate material - determination of fracture toughness, as a function of temperature, at very low strain rate - Project B29/76.		
Applied to nuclear pressure vessels. Experimental investigation of the temperature dependence of the fracture toughness of weld metal and HAZ - fracture toughness, as a function of temperature, between 20°C and 350°C - Project B36/76.		
Cyclic strain embrittlement at a fatigue crack tip - Project P9.		
Crack propagation in welds - Project P10.		
Evaluation of uncertainties in defect detection - probablistic fracture mechanics - Project P65.		
Stresses in repaired welds - Project B27/77		
Statistical evaluation of fracture toughness data - critical review of measuring and evaluating J_{1c} for A533B - Project P58.	KTH, Sweden	Swedish Nuclear Power Inspectorate
Fracture mechanics analysis of pressure vessel nozzles - Project P39.		
Probabilistic fracture mechanics for reactor pressure vessels - Project P41.		
Fracture mechanics analysis of Weldments - Project B23/76.		

Table 8. (Continued)

Title and Aims	Location	Sponsor
Ageing of weld metal in nuclear pressure vessels - investigation of static and dynamic strain ageing of stress-relief-annealed and weld repair-treated A533B - Project 826/76.	Inst for Metal Research, Sweden	Sweden
Analysis of the reliability of quality assurance of welded nuclear pressure vessels with regard to catastrophic failure - Project S19/76.	Lund Institute of Technology, Sweden	Swedish Nuclear Power Inspectorate
A study of extreme values in statistical distributions of mechanical properties - Peoject S17/77.		
Model Pressure Vessel Studies in Japan	Technical Research Inst., Hitachi Shipbdg & Engg Co. Ltd., Japan	Japan
(i) Cyclic Internal Pressure Tests of Heavy Walled Pressure Vessel 1.5m dia, 5m long, 150mm thick, containing machined defects. Fracture mechanics crack growth laws examined. Ultrasonic and Acoustic Emission measurements - Hora et al, 3rd International Conference on Pressure Vessel Technology, Tokyo, April 1977, p717.		
(ii) Fatigue crack initiation and growth at nozzle corners. 5 model vessel tests. 1m dia, 2m long, 23mm thick. 3 types of nozzle (BWR) - Miyazono et al, 3rd Conf on Pressure Vessel Technology 1977, p741.	JAERI, Japan	
(iii) Internal pressure rupture tests. A533B Cl.I, 1.0m dia, 2m long, 50mm thick, with and without nozzles. Not yet tested.	Ship Research Institute, Japan	

Table 8. (Continued)

Title and Aims	Location	Sponsor
Dynamic Fracture Toughness for Primary System Materials Re-evaluation of K_{1d}/K_{1c} of ASME III/XI. Correlation of crack arrest and various transient temperatures.	Ship Research Institute, Japan	Japan
Evaluation of finite element method for strain in integral vessel.	Delft University, Holland	Dutch Ministry of Economic Affairs
Development of computational methods for stress & deformation analysis. Cylinder - cylinder interactions. Large connections in LWR. Technique development ie high temp strain gauges.	Delft University, Holland	Dutch Ministry of Economic Affairs
Experimental & Analystic stress analysis of piping, close spaced cycles.	Oak Ridge National Lab, USA	NRC
Stress analysis of primary steel components. A comparison between calculated and measured stresses and strains in BWR main circulating pipe nozzle.	RisØ, Denmark	DAEC, Denmark
Structural and acoustic vibration in piping system. (A multi-purpose computer programme has been developed aimed at studying structural and acoustic vibration in piping system.)	PISA University, Italy	Italy
Temperature distribution measurements and thermal stress analysis of the feed water nozzle of tank of Borsebäck 2.	ASEA-ATOM, Sweden	Swedish Nuclear Power Inspectorate

Domestically the bulk of the research activities are sponsored by the Electric Power Research Institute (EPRI) and the U.S. Nuclear Regulatory Commission (NRC). A list of pertinent EPRI sponsored projects is presented below.

RP231 – Fracture of Nuclear Reactor Piping: The University of Washington will perform a detailed analytical study of subcritical flaw growth in some of the critical nuclear piping regions and a theoretical investigation of the arrest of propagating brittle cracks. A preliminary analytical investigation of the depressurization phenomena in a high-pressure, high-temperature water system to supplement the crack growth and arrest problem will be included. This is a 2-year effort. A report on the first year's effort is available (Report No. EPRI 231, No. 1, June 1975).

RP232 – Fracture Toughness in Nuclear Power Plant Components: Testing of materials will be performed by Effects Technology, Inc., Combustion Engineering, Inc., and Babcock & Wilcox, with Effects Technology acting in a program management role. Sufficient data will be obtained to provide statistical assurance of the validity of an existing fracture toughness design curve in the ASME Reactor and Pressure Vessel Code used in the design of pressure vessels for utility nuclear reactors.

RP299 – Numerical Evaluation of Stress-Intensity Factors by Global-Local Finite Element Method (GLFEM): This project will extend the global-local finite element method (GLFEM) for application to crack problems in fracture mechanics and seismic wave propagation. The essence of the proposed method is to combine generalized displacement functions, which span the entire domain, with local details described via finite elements. This method has been successfully applied to linear fracture mechanics and will be described in a forthcoming EPRI report. University of California, Los Angeles, is the contractor.

RP303 – Crack Arrest Studies: The objective of the final 18-month phase of this project is to determine the invariance of K_{Ia}, crack arrest toughness. Refinement of experimental technique will be emphasized in order to define better the fracture initiation, propagation, and arrest event. Materials Research Laboratory is the contractor.

+ **RP312 – NRL/EPRI Cooperative Fracture Toughness Research Program**: This 5-month program supplemented RP232 by including investigation of steels unique to high temperature gas-cooled reactors (HTGRs) (A508 Class 1 and A537 Class 1 and 2, etc.) and by extending the tests to larger specimens that can be handled "commercially." The contractors were Naval Research Laboratory and Effects Technology, Inc.

RP447 – Fatigue Crack Growth of Pressure Vessel Steels: This 3-year program will investigate the effect of stress, chemical environments, frequency, wave form, and radiation of fatigue crack growth of nuclear reactor pressure vessel steels. Crack growth rate information will be obtained under conditions typical of those encountered in service. The University of Missouri is the contractor.

RP498 – BWR Feedwater Nozzle Fracture Mechanics Analysis: A three-dimensional fracture mechanics analysis will be performed for a boiling water reactor feedwater nozzle to determine stress intensity factors as required by the ASME Code, Section XI. Improved sophistication of the analysis over previous efforts will result in better definition of flaw size for specific flaw sizes, shapes, and orientations of practical interest. Teledyne Engineering Services is the contractor.

+ **RP499 – Nucleation and Growth of Microfractures**: This program consisted of three tasks: (1) performance of computer simulations of the crack arrest experiments performed in RP303, Crack Arrest Studies, using the Nucleation and Growth Model to predict and model experimental results; (2) sectioning of several test specimens from RP303 and metallographically examining them to quantitatively characterize the observed fracture damage with the predicted fracture damage; and (3) analysis of the results of the above work and assessment of the feasibility of predicting crack propagation and arrest using the Nucleation and Growth Model. Stanford Research Institute was the contractor. Final Report No. NP412 (Project No. RP499), April 1977.

+ **RP585 – Predictions of Fracture from Circumferential Cracks in Boiling-Water Reactor Piping**: The objective of this project was to demonstrate that the critical flaw sizes for unstable fracture in ductile piping material are of sufficient size that detection or leakage will occur prior to fracture. Data and analytical methodology from the project should minimize or avoid the need for extended shutdowns. Battelle Memorial Institute was the contractor. Final Report No. NP192 (Project No. 585–1)), September 1976.

RP601 – Methodology for Plastic Fracture: This 3-year project is the development of methodology for making engineering predictions concerning the behavior of flawed structures in the plastic regime. A fracture mechanics methodology, having application in the plastic regime, is required to permit more quantitive assessment of the margins of design inherent in nuclear reactor pressure vessels. Analytical techniques for evaluating stable crack growth criteria have been developed for flat shear and mixed mode fractures. Specimen testing has proceeded to describe material behavior for plastic properties. General Electric Company and Battelle Memorial Institute are the contractors.

RP602 – Numerical Analysis of Welds: The objective of this project is to eventually provide a reliable analysis procedure to calculate the residual stresses induced by the welding process. Two analysis techniques — finite element and finite difference — will be compared against experimental data. The features of each method and their sensitivity to variations of physical conditions will be established. The contractors are the Marc Analysis Research Corporation and Science Applications, Inc.

RP603 – Fundamental Study of Crack Initiation and Propagation: This 30-month study will establish basic predictive capability requirements for potential failure of the pressure vessel in the presence of material flaws. A coupled calculational and experimental approach will be used to predict crack initiation and propagation. Success in this program will allow for calibration of simplified design methods in fracture problems. Lawrence Livermore Laboratory and Science Applications, Inc., are the contractors.

+**RP696 – Fracture Toughness Statistical Analysis**: This 1-year program was an attempt to overcome some procedural and data difficulties in order to develop a statistically based KIR fracture toughness design curve. Fracture Control Corporation was the contractor. Final Report No. NP372 (Project No. RP696–1) May 1977.

RP697 – Theoretical and Experimental Analysis of Residual Stresses In Reactor Components: Improvement in the predictability of inelastic deformation in reactor components at elevated temperatures is the theme of this 2-year project. Objectives are to develop inelastic constitutive relations for type 304 stainless steel, install the equations in a general purpose finite element stress code, and demonstrate the capabilities of the code in predicting test data. Cornell University is the contractor.

RP1022 – Influence of Dynamic Effects of Crack Arrest: The objectives of this 17-month project are: (1) to evaluate the effect of geometry on the effects of crack arrest; (2) to evaluate the effect of crack velocity on fracture toughness; and (3) to develop an experimental procedure for measuring fracture toughness. If the limits of the present simplified code crack arrest analysis can be adequately justified, extensive dynamic analyses of each defect discovered during in-service inspection will not be necessary. The contractor is the Institut fur Festkorpermechanik (Germany).

RP1023 – Small Specimen Fracture Mechanics Analysis: The objective of this 2-year project is to establish a quantitative, phenomena-based computational fracture model capable of predicting ductile fractures. With such a model, it should be possible to remove some of the existing overconservatism in toughness limits for reactor pressure vessels. SRI International is the contractor.

RP1123 – Evaluation of Multiaxial Fatigue: The objectives of this 24-month project are: (1) to develop and to evaluate an engineering method for analysis of multiaxial fatigue; (2) to evaluate conservative restraints in the present ASME Boiler and Pressure Vessel Code with respect to multiaxial fatigue; and (3) to develop multiaxial fatigue data on nuclear pressure vessel materials for development of new multiaxial fatigue design curves. Stanford University is the contractor.

RP1174 – Effects of Weld Parameters on Residual Stresses in Pipes: Welding-produced residual stresses are the major causes of pipe cracking in boiling-water reactor systems. In addition, fatigue and fracture behavior lifetimes are reduced by weld residual stresses. The most effective method to optimize weld properties is by systematic testing and analysis. The objective of this project is to refine a multipass welding code (developed by the contractor, Battelle, Columbus Laboratories) in order to incorporate additional parameters believed to be important in pipe welding and to use the program to rationalize optimized welding procedures. A second objective is to evaluate welding remedies studies in RP701.

RP1237 – Simplified Prediction of Elastic-Plastic Fracture: Research efforts are developing techniques for assessing design margins for pressure vessel components. The ultimate goal is to develop an engineering procedure for ductile fracture analysis of nuclear components intended to meet both design and regulatory needs, and reduce the attendant conservatisms associated with the present elastic flaw evaluation procedures. The principal objective of this project is to produce a simplified engineering tool in the form of a plastic fracture handbook and analytic procedure for the analysis of fracture in generalized structural components, a handbook similar to those currently used in the analysis of elastic fracture. General Electric Company is the contractor.

RP1238 - The Effect of Specimen Size and Configuration on Fracture Toughness and Ductile Instability: This work will help to predict accurately the behavior of reactor vessel materials and, thereby, avoid some of the economic penalties associated with overconservatism in fracture toughness analyses. The project objectives are: (1) to procure a suitable project material with a low charpy energy upper shelf (approximately 50 ft-lb), a yield strength greater than 50 KSi, and acceptable for reactor vessel fabrication prior to 1973; (2) to measure experimentally the relationship between plastic toughness and the "elastic" toughness; (3) to develop guidelines for stable crack growth testing for submission to the ASTM; (4) to evaluate the simplified test procedure for stable crack growth and the applicability of a new tearing instability analysis. Westinghouse Electric Corporation and Washington University are the contractors.

RP1241 - Feedwater (FW) Nozzle Instability Analysis: The project objective is to demonstrate analytically that thermal fatigue cracks in FW nozzles of BWR systems are not a safety problem, e.g., catastrophic failure of the vessel is impossible, even for very large cracks and high internal pressures. The task is to develop a two-dimensional instability analysis of a FW nozzle, incorporating the effects of the reactor vessel and flaw location. Washington University is the contractor.

***RP1325 - Corrosion Fatigue Characterization of Reactor Pressure Vessel Steels**: Present methods for evaluating crack growth rates under corrosion fatigue condition in reactor pressure vessels have proven to be inadequate. To prevent excessively conservative regulatory reaction in evaluating the safety of flaws found in future inservice inspections, a corrosion fatigue materials data base will be developed and used, in turn, to develop the predictive methodology for determining crack growth rates under actual reactor operating conditions.

***RP1394 - Techniques to Mitigate BWR Pipe Cracking in Existing Plants**: Residual tensile stress has been identified as a major cause of pipe cracking. It has also been established that resistance can be improved by altering the stress distribution associated with pipe welds. The objective of this project is to provide a practical method of treating any surface of BWR piping with heat induction, which would redistribute residual stresses and thus prevent pipe cracking. General Electric Company is a contractor.

These projects do not include those that are wholly associated with the experimental investigation of the intergranular stress corrosion cracking (IGSCC) problems found in boiling-water reactor (BWR) piping systems. This separate program includes an examination of the susceptibility of the current generations of austenitic stainless steels and their weldments to IGSCC and to the development of alternate techniques to limit the propensity of IGSCC in the weldments. This effort also includes programs to investigate the employment of alternate ferritic and austenitic steel materials that would not be susceptible to IGSCC. Project management of the wide scope of these related EPRI sponsored projects has been given to the EDAC Company of Palo Alto, California.

The parallel but smaller scope activities being conducted by Westinghouse are:

1. joint testing program with Framatome of France on fatigue in stainless steel and on crack growth in stainless and ferritic steels,
2. joint testing program with Naval Research Laboratory on crack growth in ferritic steels in a pressurized-water reactor environment for the Nuclear Regulatory Commission, and
3. in-house testing programs on crack growth in stainless and ferritic steels in air and steam.

This company is an active participant in several "publically funded" EPRI and NRC projects as well as being a sponsor-cosponsor of internally funded activities. Some of the internally funded projects are being coordinated with Framatome, which is conducting fatigue life experiments on stainless steel weldments in appropriate environments. The General Electric Company also has a similar committment to the fracture analysis of reactor components. Their 72 fatigue units in the Pipe Test Laboratory (PTL) are wholly committed to the EPRI funded projects surrounding the BWR pipe IGSCC problems.

The Heavy Section Steel Test (HSST) program being directed by ORNL is an example of a large scale NRC sponsored effort regarding fracture of pressure vessel steels under cyclical loads. Westinghouse and the Naval Research Laboratory (NRL), among others, are major participants in this program. A parallel and complimentary effort is being funded by the German Federal Republic at the University of Stuttgart's Materealprüfungsanstalt (MPA) facilities. The efforts for these two major programs are being jointly coordinated by ORNL and MPA.

In reviewing the scope of the aforementioned research and investigation activities, it appears that little attention is currently being directed toward the generation of data that could improve the credibility of ASME-III-Class 1-type evaluations and refine "S-N" curves for ferritic and austentic steel weldments.

Analytical development efforts are overwhelmingly concentrated upon the evaluation of the performance of steels with respect to crack propagation and to a lesser extent, crack initiation characteristics. Although designs are based upon crack-free components, most investigations recognize the potential for the cracking in the weldment areas, and thus there is a concentrated effort to examine the weldments during the in-service inspection periods for the reactors. In this regard linear elastic fracture mechanics (LEFM) techniques and the acquisition of an applicable data base have been given the most attention. Although this ASME III-1, Appendix G, and ASME XI, Appendix A, methodology is the most advanced of the analytical predictive techniques, it has been acknowledged that uniform applications of the procedures have not been universally accepted.

An attempt to remedy this deficiency is being made by a 25-member international group, which will, via a series of round-robin investigations, define standardized procedures for employment of LEFM techniques. It is projected that these (eventual) standardized procedures will be employed to further refine the understanding of the specific mechanisms and ingredients that govern crack initiation and growth in base metals and their weldments. Furthermore, it is anticipated that these standardized procedures may be the vehicles that will assist in the definition of future ASME Code requirements regarding fatigue of components and their weldments.

For the austenitic steel materials and operating temperatures in light-water reactor (LWR) piping systems, the anticipated failure mode, should failure occur, is likely to be either plastic structural instability or unstable ductile tearing before general plastic instability. As large amounts of plasticity will accompany such a failure, and LEFM methods may not be useful.

In summary, it appears that today only marginal efforts are being directed toward reassessing the adequacy of employing the current ASME-III techniques for fatigue evaluation of weldments. Experimental validation is minimal and analytical investigations are almost nonexistant.

CONCLUSIONS AND RECOMMENDATIONS

Several general conclusions can be derived from the information gathered in this investigation, although in many cases the existing information is incomplete. These conclusions include:

1. Existing ASME Code design procedures do not distinguish between base metal and weldments in the fatigue design criteria.

2. A general survey of data on fatigue of weldments indicates that any loss of fatigue strength resulting from a weldment would be from early crack initiation rather than from accelerated crack growth. For the particular case of pressure vessels and piping, the major factor that might cause such early crack initiation appears to be the existence of defects in the weld.

3. Design experience with nuclear pressure vessels and piping appears to have been generally satisfactory under the existing Code rules. However, weldment fatigue has been a problem in some other industries, and experience in the nuclear industry is relatively limited.

4. Significant research activities are currently under way in related fields, but little effort is presently being directed toward experimental or analytical verification of current ASME fatigue design procedures.

5. Within the limitations of available information, there is no reason to suggest any changes in the ASME Code fatigue design procedures at this time.

6. Limitations in available information make it impossible to fully verify the current fatigue procedures. These limitations include: virtually no data are available on thermal fatigue of weldments, and available mechanical fatigue data are very limited under low-cycle strain-controlled conditions; few fatigue data are available for the exact materials of interest to this investigation, and current design curves were developed primarily from data from other (although similar) materials; little information is available on the effects of repair welding; few data are available for fatigue under variable loads and other cumulative damage situations; detailed quantitative data for the effects of weld defects on fatigue strength are not available for the materials of interest, however, current technology appears capable of assessing such effects; the significance of the effects of reheat cracking in the HAZs of A508 Class 2 base material has not been fully assessed under all conditions; and analytical studies to assess the impact of "metallurgical notches" (changes in material properties across the weld joint) have not been undertaken, though they appear technically feasible.

Specific recommendations for future research as determined from this investigation are:

1. Studies should be undertaken to gain a more thorough understanding of the relative contributions of crack initiation and crack propagation to fatigue life. These studies could include experimental investigations as well as analytical-fracture mechanics studies aimed at a better definition of initiation, that is, "when is a crack a crack?"

2. Attention should be given to generation of fatigue data for pressure vessel and piping materials under conditions relevant to nuclear service. Such studies should include cumulative damage and thermal fatigue data. Tests should involve material from prototypic commercial weldments.

3. Consideration should be given to a fuller characterization of the properties of repair welds.

4. Further studies are needed to fully assess the significance of reheat cracks in A508 Class 2 HAZs. Such studies should include considerations such as those given in the above three items.

5. An investigation of the feasibility of directly addressing effects of weld defects on a quantitative basis should be undertaken. Such an investigation should include detailed examination of available data and generation of systematic new data where necessary.

6. An analytical investigation should be undertaken to determine the effect of "metallurgical notches" on fatigue strength. Such analyses could determine differences in maximum local strain ranges for weldments vs homogeneous materials under the same loading conditions. If these differences are significant, such analyses would lead to development of "metallurgical notch" factors to account for the difference.

ACKNOWLEDGMENTS

The authors would like to acknowledge the help and cooperation provided by the many organizations and individuals contacted in this investigation. We would also like to thank D. A. Canonico, K. C. Liu, C. R. Brinkman, and G. M. Slaughter for reviewing the contents of this report; B. G. Ashdown for editing; and S. G. Frykman for preparing the final report for publication.

REFERENCES

1. *ASME Boiler and Pressure Vessel Code, Sect. III, Div. 1*, American Society of Mechanical Engineers, New York, 1977.
2. J. Heuschkel, *Weldment Fatigue — A Bibliography of Cited USA References*, Document XIII-819-77, International Institute of Welding, Copenhagen (April 1977).
3. H. S. Reemsnyder, "Bibliography on Fatigue of Weldments," *Weld. J. (Miami)* 57: 178-s–182-s (June 1978).

4. C. F. Larson, "Bibliography on Fatigue Strength of Welded Joints, 1950–1971," *Weld. J. (Miami)* 51: 457-s–460-s (September 1972).

5. M. K. Booker and B.L.P. Booker, *Development and Implementation of a Mechanical Properties Data Storage and Retrieval System*, ORNL/TM-5330 (June 1976).

6. *ASME Boiler and Pressure Vessel Code, Sect. II, Pt. A*, American Society of Mechanical Engineers, New York, 1977, pp. 499–501.

7. *ASME Boiler and Pressure Vessel Code, Sect. II, Pt. A*, American Society of Mechanical Engineers, New York, 1977, pp. 469–73.

8. *ASME Boiler and Pressure Vessel Code, Sect. II, Pt. A*, American Society of Mechanical Engineers, New York, 1977, pp. 237–40.

9. *ASME Boiler and Pressure Vessel Code, Sect. II, Pt. A*, American Society of Mechanical Engineers, New York, 1977, pp. 477–78.

10. J. A. Rinebolt and W. J. Harris, Jr., "Effect of Alloying Elements on Notch Toughness of Pearlitic Steels," *Trans. Am. Soc. Met.* 43: 1175–214 (1951).

11. T. G. Gooch and J. Honeycombe, "Microcracking in Fully Austenitic Stainless Steel Weld Metal," *Met. Constr. Br. Weld. J.* 7: 146–48 (March 1975).

12. G. M. Slaughter, *Welding and Brazing Techniques for Nuclear Reactor Components*, Rowman and Littlefield, New York, 1964.

13. W. H. Kearns, Ed., *Welding Handbook*, 7th ed., American Welding Society, Miami, 1978.

14. G. E. Linnert, *Welding Metallurgy*, Vol. 1, 3d ed., American Welding Society, New York, 1967.

15. M. A. Miner, "Cumulative Damage in Fatigue," *J. Appl. Mech.* 12: 159–64 (September 1945).

16. T. R. Gurney, *Fatigue of Welded Structures*, Cambridge University Press, Cambridge, 1968.

17. B. Pollard and R. J. Cover, "Fatigue of Steel Weldments," *Weld. J. (Miami)* 51: 544-s–554-s (November 1972).

18. W. H. Munse, "Fatigue of Weldments — Tests, Design, and Service," pp. 89–112 in *Fatigue Testing of Weldments, Am. Soc. Test. Mater. Spec. Tech. Publ.* 648, American Society for Testing and Materials, Philadelphia, 1978.

19. H. S. Reemsnyder, "Development and Application of Fatigue Data for Structural Steel Weldments," pp. 3–21 in *Fatigue Testing of Weldments, Am. Soc. Test. Mater. Spec. Tech. Publ.* 648, American Society for Testing and Materials, Philadelphia, 1978.

20. T. J. Dolan, "Significance of Fatigue Data in Design of Pressure Vessels," *Weld. J. (N.Y.)* 35: 255-s–260-s (May 1956).

21. J. Dubuc and G. Welter, "Investigation of Static and Fatigue Resistance of Model Pressure Vessels," *Weld. J. (N.Y.)* 35: 329-s–337-s (July 1956).

22. T. R. Gurney, "Finite Element Analyses of Some Joints with the Welds Transverse to the Direction of Stress," *Weld. Res. Abroad,* 22: 2–29 (December 1976).

23. S. J. Maddox, "Fatigue Crack Propagation in Weld Metal and HAZ," *Weld. Res. Abroad,* 16: 15–19 (October 1970).

24. M. R. Baren and R. P. Hurlebaus, "The Fatigue Properties of a Welded Low Alloy Steel," *Weld. J. (N.Y.)* 50: 207-s–212-s (May 1971).

25. J. H. Gross and R. D. Stout, "Plastic Fatigue Properties of High-Strength Pressure-Vessel Steels," *Weld. J. (N.Y.)* 34: 161-s–166-s (April 1955).

26. W. D. Doty, "Properties and Characteristics of a Quenched and Tempered Steel for Pressure Vessels," *Weld. J. (N.Y.)* 34: 425-s–441-s (September 1955).

27. C. E. Bowman and T. J. Dolan, "Resistance of Low-Alloy Steel Plates to Biaxial Fatigue," *Weld. J. (N.Y.)* 35: 102-s–109-s (February 1956).

28. K. Kobayoshi et al., "Improvement in the Fatigue Strength of a Fillet Welded Joint by Use of a New Welding Electrode," *Weld. Res. Int.* 8: 43–74 (January 1978).

29. K. Ikeda, S. Denoh, T. Godai, and T. Ogawa, "Improvement of the Fatigue Strength of Fillet Welded Joints in 780 N/mm^2 High Strength Steel," *Weld. Res. Int.* 8: 18–25 (January 1978).

30. M. Hanzawa et al., "Improvement of Fatigue Strength in Welded High Tensile Steel by Toe Treatment," *Weld. Res. Int.* 8: 28–51 (January 1978).

31. H. S. Reemsnyder, "Some Significant Parameters in the Fatigue Properties of Welded Joints," *Weld. J. (N.Y.)* 48: 213-s–220-s (May 1969).

32. H. S. Reemsnyder, *Fatigue Properties of Welded Roller-Quenched and Tempered (RQ) Steels*, Soc. Aut. Eng. [Tech. Pap.] 790519, Society of Automotive Engineers, New York, April 1979.

33. W. M. Wilson and A. B. Wilder, "Fatigue Tests of Welded Joints in Structural Plates," *Weld. J. (N.Y.)* 19:100-s–108-s (March 1940).

34. W. M. Wilson, "Fatigue Strength of Commercial Butt Welds in Carbon-Steel Plates," *Weld. J. (N.Y.)* 21: 491-s–496-s (October 1942).

35. W. W. Sanders, Jr., A. T. Derecho, and W. H. Munse, "Effect of External Geometry on Fatigue Behavior of Welded Joints," *Weld. J. (N.Y.)* 44: 49-s–55-s (February 1965).

36. G. L. Archer, *The Significance of Defects in Welded Pressure Vessels: Fatigue Tests on Seam Welds in Cylindrical Pressure Vessels Containing Slag, Porosity, and Lack of Penetration*, Research Report 3322/18/74, The Welding Institute, Abington, England (May 1974).

37. C. F. Boulton, "The Influence of Surface Porosity on the Fatigue Strength of Butt Welded Joints," *Weld. Inst. Res. Bull.* 18: 41–45 (February 1977).

38. K. Jerram, "An Assessment of the Fatigue of Welded Pressure Vessels," pp. 1303–17 in *1st Int. Conf. Pressure Vessel Technol. Pt. II, Materials and Fabrication*, American Society of Mechanical Engineers, New York, 1969.

39. C. D. Lundin, *The Significance of Weld Discontinuities — A Review of Current Literature, Weld. Res. Counc. Bull.* 222, Welding Research Council, New York, 1976.

40. T. W. Crooker and E. A. Lange, "How Yield Strength and Fracture Toughness Considerations Can Influence Fatigue Design Procedures for Structural Steels," *Weld. J. (N.Y.)* 49: 488-s–496-s (October 1970).

41. T. J. Dolan, "Fatigue as a Factor in Pressure Vessel Design," *Weld. J. (N.Y.)* 33: 265-s–275-s (June 1954).

42. O. Masi and A. Erra, "The Reduction in Fatigue Strength of Weld Metal by Defects," *Weld. Res. Abroad* 2: 33–36 (January 1956).

43. S. J. Maddox, "Assessing the Significance of Flaws in Welds Subject to Fatigue," *Weld. J. (Miami)* 53: 402-s–409-s (September 1974).

44. F. V. Lawrence and W. H. Munse, "Fatigue Crack Propagation in Butt Welds Containing Joint Penetration Defects," *Weld. J. (Miami)* 52: 221-s–225-s (May 1973).

45. A. Matting and M. Neitzel, "The Influence of Weld Defects on the Fatigue Behavior of Welded Constructions," *Br. Weld. J.* 14: 3–12 (January 1967).

46. Y. Tobe and F. V. Lawrence, Jr., "Effect of Inadequate Joint Penetration on Fatigue Resistance of High-Strength Structural Steel Welds," *Weld. J. (Miami)* 56: 259-s–266-s (September 1977).

47. A. R. Jack and A. T. Price, "The Use of Crack Initiation and Growth Data in the Calculation of Fatigue Lives of Specimens Containing Defects," *Met. Const. Br. Weld. J.* 3: 416–19 (November 1971).

48. J. D. Harrison, "Basis for a Proposed Acceptance-Standard for Weld Defects. Part 1: Porosity," *Met. Constr. Br. Weld. J.* 4: 99–107 (March 1972).

49. J. D. Harrison, "Basis for a Proposed Acceptance-Standard for Weld Defects. Part 2: Slag Inclusions," *Met. Constr. Br. Weld. J.* 4: 262–68 (July 1972).

50. A. Matting and M. Neitzel, "The Evaluation of Weld Defects in Fatigue Testing," *Weld. Res. Abroad*, 12: 34-60 (August-September 1966).

51. E. G. Signes, R. G. Baker, J. D. Harrison, and F. M. Burdekin, "Factors Affecting the Fatigue Strength of Welded High Strength Steels," *Br. Weld. J.* 14: 108–16 (March 1967).

52. W. R. Miller and H. S. Amin, "Low-Cycle Fatigue in Welds," *Exp. Mech.* 15: 230–33 (June 1975).

53. F. Watkinson, P. H. Bodger, and J. D. Harrison, "The Fatigue Strength of Welded Joints in High Strength Steels and Methods for its Improvement," pp. 97–113 in *Proc. Conf. Fatigue of Welded Structures*, The Welding Institute, Abington, England, 1971.

54. J. D. Harrison and J. Doherty, *Re-Analysis of Fatigue Data for Butt Welded Specimens Containing Slag Inclusions*, Research Report 37/1977/E, The Welding Institute, Abington, England, (April 1977).

55. J. D. Harrison, F. M. Burdekin, and J. G. Young, "A Proposed Acceptance Standard for Welded Defects Based Upon Suitability for Service," *Weld. Res. Abroad* 15: 72–85 (November 1969).

56. R. P. Newman, "Effect on Fatigue Strength of Internal Defects in Welded Joints," *Weld. Res. Abroad* 5: 45–50 (May 1959).

57. H. Moller and M. Hempell, "The Effect of Defects on the Fatigue Strength and Submerged-Arc Welds," *Weld. J. (N.Y.)* 37: 113-s and 128-s (March 1958, abstracted translation).

58. J. D. Harrison, "Further Fatigue Tests of 1.5 in Thick Butt Welds Containing Slag Inclusions," *Weld. Res. Abroad* 14: 56–65 (April 1968).

59. S. Yokoi, "The Effect of Radiographically Detected Porosity on the Fatigue Strength of Butt Welded Joints," *Weld. Res. Abroad* 10: 90 (August-September 1964).

60. G. E. Linnert, *Welding Metallurgy*, Vol. 2, 3d ed., American Welding Society, New York, 1967.

61. W. G. Warren, "Fatigue Tests on Defective Butt Welds," *Weld. Res.* 6(6): (1952), as cited in ref. 16.

62. W. H. Munse et al., *Studies of Fatigue Behavior of Butt-Weld Joints in HY-80 and HY-100 Steels*, Civil Engineering Studies Structural Research Series No. 285, University of Illinois, Urbana-Champaign, Ill., November 1964.

63. S. J. Maddox, "The Application of Fracture Mechanics Techniques to Fatigue in Welded Joints, Part 1: Linear Elastic Fracture Mechanics," *Weld. Inst. Res. Bull.* 12(11): 307–12 (November 1971).

64. S. J. Maddox, "A Fracture Mechanics Approach to Service Load Fatigue in Welded Structures," *Weld. Res. Int.* 4(2): 1–30 (1974).

65. S. J. Maddox, "Fracture Mechanics Applied to Fatigue in Welded Structures," pp. 73–96 in *Proc. Conf. Fatigue of Welded Structures*, Vol. I, The Welding Institute, Abington, England, 1971.

66. S. J. Maddox, "Calculating the Fatigue Strength of a Weld Joint Using Fracture Mechanics," *Met. Constr. Br. Weld. J.* 2: 327–31 (1970).

67. F. V. Lawrence, "Estimation of Fatigue-Crack Propagation Life in Butt Welds," *Weld. J. (Miami)* 52: 212-s–220-s (May 1973).

68. F. V. Lawrence, "Fatigue Crack Propagation in Butt Welds Containing Joint Penetration Defects," *Weld. J. (Miami)* 52: 221-s–225-s (May 1973).

69. H. Thielsch, "Weld Surface Contour Variations," *Weld. Eng.* 51: 60, 62–67 (November 1966).

70. F. Guyot, H. Louis, J. Martelee, and W. Soete, "Fatigue Tests on Transverse Butt Welds Characterized by an Incomplete Root Penetration — Series 1," *Weld. Res. Abroad* 14: 2–55 (April 1968).

71. F. Guyot, J. Martelee, and W. Soete, "Fatigue Tests on Transverse Butt Welds Characterized by an Incomplete Root Penetration — Series 2," *Weld. Res. Abroad* 14: 2–22 (May 1968).

72. A. L. Lowenberg, E. B. Norris, and A. R. Whiting, *Evaluation of Discontinuities in Pipeline Weld Joints — Summary Report No. 1*, Pressure Vessel Research Committee of the Welding Research Council, New York, November 1968.

73. W. Soete and A. Sys, "Influence of Weld Defects on High-Fatigue Behavior," pp. 1147–56 in *1st Int. Conf. Pressure Vessel Technol.*, American Society of Mechanical Engineers, New York, 1968.

74. W. M. Wilson et al., *Fatigue Strength of Various Types of Butt Welds Connecting Steel Plates, Ill. Univ. Eng. Exp. Stn. Bull.* 384, University of Illinois, Urbana-Champaign, Ill., March 1950.

75. R. P. Newman and M. G. Dawes, "Exploratory Fatigue Tests on Transverse Butt Welds Containing Lack of Penetration," *Br. Weld. J.* 12: 117–20 (March 1965).

76. J. B. Radziminski and F. V. Lawrence, Jr., "Fatigue of High-Yield Strength Steel Weldments," *Weld. J. (N.Y.)* 49: 365-s–375-s (August 1970).

77. F. M. Burdekin, J. D. Harrison, and J. G. Young, "The Effect of Weld Defects with Special Reference to BWRA Research," *Weld. Res. Abroad* 14: 58–67 (August-September 1968).

78. T. R. Gurney, "Effect of Stress Relief on the Fatigue Strength of 1/2 inch (12 mm) Thick Transverse Butt Welds Containing Slag Inclusions," *Br. Weld. J.* 11: 593–603 (November 1964).

79. T. R. Gurney and G. C. Smith, "Fatigue Tests on 1 1/2 inch (36 mm) Thick Transverse Butt Welds Containing Slag Inclusions," *Br. Weld. J.* 14: 17–38 (January 1967).

80. J. D. Harrison and T. R. Gurney, "Some Fatigue Tests on 1 1/2 inch (36 mm) Thick Butt Welds Containing Slag Lines," *Br. Weld. J.* 14: 121–31 (March 1967).

81. J. D. Harrison and G. C. Smith, "Effects on Hydrogen on the Fatigue Behavior of Mild Steel Bar and Weld Metal," *Br. Weld. J.* 14: 493–502 (September 1967).

82. "The Effect of Slag Inclusions on the Fatigue Strength of Machined Butt Welds," *Weld. World* 7: 212–38 (Winter 1969).

83. *Report of Tests on 1/2 inch (12 mm) Thick Butt Welds Containing Slag Inclusions (Series B1–B4)*, German report to the International Institute of Welding (Copenhagen) Commission XIII Working Group on Defects, as cited in ref. 49.

84. J. E. Zimmerman and W. H. Munse, *Fatigue Behavior of Defective Welded Joints in HY-80 Steel Subjected to Axial Loading*, Civil Engineering Studies Structural Research Series No. 252, University of Illinois, Urbana-Champaign, Ill., July 1962.

85. R. P. Meister et al., *Effects of Weld-Joint Imperfections on Piping Performance*, Battelle Memorial Institute Report (July 1964), as cited in ref. 49.

86. *Alternating Fatigue Tests on 1/2 inch (12 mm) Thick Joints Butt-Welded with Low Hydrogen Electrodes Containing Slag Inclusions at Mid-Thickness*, French report to International Institute of Welding (Copenhagen) Commission XIII Working Group on Effect of Weld Defects on Fatigue Behavior, as cited in ref. 49.

87. Y. Ishii and K. Iida, "Low and Intermediate Cycle Fatigue Strength of Butt Welds Containing Defects," *J. Soc. Non-Destr. Test. (Japan)* 18(10): 1–32 (1969). Also International Institute of Welding (Copenhagen) Document XIII-560-69.

88. R. Clough, "Application of Weld Performance Data," *Br. Weld. J.* 15: 319–25 (July 1968).

89. F. daKazinczy, "Fatigue Properties of Repair Welded Cast Steel," *Br. Weld. J.* 15: 447–50 (September 1968).

90. A. J. Hartment, W. H. Bruckner, J. Mooney, and W. H. Munse, *Effect of Weld Flaws on the Fatigue Behavior of Butt-Welded Joints in HY-80 Steel*, Civil Engineering Studies Structural Research Series No. 275, University of Illinois, Urbana-Champaign, Ill., December 1963.

91. F. Guyot and J. Martelee, *Essais de Fatigue sur Soudunes Bout à Bout Transversales Contenant des Soufflures*, C.R.I.F. Brussels Report MT53 (June 197), as cited in ref. 48.

92. Y. Ishii, H. Kihara, and Y. Tada, "On the Relation Between the Non-Destructive Testing Information of Steel Welds and their Mechanical Strength," *Weld. Res. Abroad* 13: 2–26 (June-July 1967).

93. M. G. Dawes, "Ferritic Steel Shafts Reclaimed by Welding and Metal Spraying," *Br. Weld. J.* 10: 428–38 (August 1963).

94. E.H.K. Wellinger and D. Sturm, *Influence of Weld Seam Quality on the Fatigue Resistance under Pulsating Internal Pressure in the Case of Vessels with Nozzles,* Document XIII-578-70, International Institute of Welding, Copenhagen, 1970.

95. K. D. Ives, L. F. Kooistra, and J. T. Tucker, "Equibiaxial Low Cycle Fatigue Properties of Typical Pressure Vessel Steels," *Pressure Vessels and Piping Design and Analysis,* Vol. 1, American Society of Mechanical Engineers, New York, 1969.

96. L. F. Kooistra, E. A. Lange, and A. G. Pickett, "Full Size Pressure Vessel Testing and Its Application to Design," *Pressure Vessels and Piping Design and Analysis,* Vol. 1, American Society of Mechanical Engineers, New York, 1969.

97. B. F. Langer, "Criteria of ASME Boiler and Pressure Vessel Code for Design by Analysis in Sect. IV and VIII, Division 2," *Pressure Vessels and Piping Design and Analysis,* Vol. 1, American Society of Mechanical Engineers, New York, 1969.

98. A. G. Pickett and S. C. Grigory, *Cyclic Pressure Tests of Full-Size Pressure Vessels,* Weld. Res. Counc. Bull. 135, Welding Research Council, New York, 1968.

99. G. Welter and J. Dubuc, "Fatigue Resistance of Simulated Nozzles in Model Pressure Vessels of Steel," *Weld. J. (N.Y.)* 14: 368-s–374-s (August 1962).

100. L. F. Kooistra and M. M. Lemcoe, "Low Cycle Fatigue of Full Size Pressure Vessels," *Weld. J. (N.Y.)* 41: 297-s–306-s (July 1962).

101. J. D. Heald and E. Kiss, "Low Cycle Fatigue of Nuclear Pipe Components," *J. Pressure Vessel Technol.* 96: 171–76 (August 1974).

102. J. D. Harrison, "Low Cycle Fatigue Tests on Welded Joints in High Strength Steels," pp. 194–207 in *Proc. Conf. Fatigue of Welded Structures,* The Welding Institute, Abington, England, 1971.

103. J. D. Harrison, "The Analysis of Fatigue Test Results for Butt Welds with Lack of Penetration Defects Using a Fracture Mechanics Approach," *Weld. Res. Abroad* 18: 68–77 (March 1971).

104. J. D. Harrison, "Proposals for an Acceptance Standard for Defects in the Main Seams of Pressure Vessels," *Br. J. Non-Destr. Test.* 12: 12–16 (March 1970).

105. K. G. Richards, *Fatigue Strength of Welded Structures,* The Welding Institute, Abington, England, May 1969.

106. J. D. Burk and F. V. Lawrence, Jr., *The Effect of Residual Stresses on Weld Fatigue Life,* FCP Rep. 29, University of Illinois, College of Engineering, Urbana-Champaign, Ill., January 1978.

107. K. A. Selby, J. E. Stallmeyer, and W. H. Munse, *Influence of Geometry and Residual Stress on Fatigue of Welded Joints,* Civil Engineering Studies Structural Research Series No. 297, University of Illinois, Urbana-Champaign, Ill., June 1965.

108. R. P. Newman and T. R. Gurney, "Fatigue Tests of Plain Plate Specimens and Transverse Butt Welds in Mild Steel," *Br. Weld. J.* 6: 569–94 (June 1959).

109. T. R. Gurney, "Fatigue of Welded Military Structures," *Br. Weld. J.* 15: 276–82 (June 1968).

110. B. F. Langer, "Design of Pressure Vessels for Low-Cycle Fatigue," *J. Basic Eng.* 84: 389–402 (September 1962).

111. J. M. Barsom, *Fatigue Behavior of Pressure Vessel Steels, Weld. Res. Counc. Bull.* 194, Welding Research Council, New York, May 1974.

112. L. A. James, "Fatigue-Crack Propagation Behavior of Several Pressure Vessel Steels and Weldments," *Weld. J. (Miami)* 56: 386-s–391-s (December 1977).

113. D. A. Canonico, "Significance of Reheat Cracks to the Integrity of Pressure Vessels for Light-Water Reactors," *Weld. J. (Miami)* 58(5): 127-s–142-s (May 1979).

114. P. S. Ayres et al., *Study of Intergranular Separations in Low-Alloy Steel Heat-Affected Zones Under Austenitic Stainless Steel Weld Cladding,* BAW-10013 (December 1971).

115. A. Dhooge et al., "Review of Work Related to Reheat Cracking in Nuclear Reactor Pressure Vessel Steels," *Int. J. Pressure Vessels and Piping* 6: 329–409 (1978).

116. R. Kume, H. Okabayashi, and M. Amano, "Mechanism of Underclad Cracking — Combined Effects of Residual Strain and Heat-Affected Zone Ductility," *J. Eng. Mater. Technol.* 98: 348–56 (October 1976).

117. H. Nakamura, T. Naiki, and H. Okabayashi, "Fracture in the Process of Stress-Relaxation Under Constant Strain," p. 863 in *1st Int. Conf. Fracture*, Vol. 1, The Japanese Society for Strength and Fracture of Materials, 1966.

118. R. Kume, H. Okabayahsi, and M. Amano, "Susceptibility to Underclad Cracking in Relation to Chemical Composition," *J. Eng. Mater. Technol.* 98: 342–47 (October 1976).

119. A. Vinckier and A. Pense, *A Review of Underclad Cracking in Pressure Vessel Components, Weld. Res. Counc. Bull.* 197, Welding Research Council, New York, August 1974.

120. R. E. Dolby and G. G. Saunders, *A Review of the Problem of Reheat Cracking in Nuclear Vessel Steels*, Research Report 18/1976/M, The Welding Institute, Abington, England (October 1976).

121. G. E. Grotke et al., *Reheat Cracking and Heat-Affected-Zone Toughness in SA 508 Class 2 Steel Weldments*, Westinghouse Research Laboratory Report 76-IDA-REHET-R1 (February 1976).

122. K. Rahka and J. Forstén, *The Growth of Underclad Flaws in Fatigue Loading*, Reactor Materials Research Report 64, Technical Research Centre of Finland (August 1976).

123. T. R. Mager, E. Landerman, and C. Kubit, *Reactor Vessel Weld Cladding — Base Metal Interaction*, WCAP-7733 (July 1971).

124. T. R. Mager, J. D. Landes, D. M. Moon, and V. J. McLaughlin, *The Effect of Low Frequencies on the Fatigue Crack Growth Characteristics of A533 Grade B Class 1 Plate in an Environment of High-Temperature Primary Grade Nuclear Reactor Water*, WCAP-8256, HSST Tech. Report 35 (December 1973).

125. D. W. Hoeppner, "Summary," pp. 285–87 in *Fatigue Testing of Weldments, Am. Soc. Test. Mater. Spec. Tech. Publ.* 648, American Society for Testing and Materials, Philadelphia, 1978.

126. R. V. Salkin, "Low Cycle Fatigue of Welded Structural Steels: A Material Manufacture and Design Approach," pp. 183–93 in *Proc. Conf. Fatigue of Welded Structures*, The Welding Institute, Abington, England, 1971.

127. S. J. Maddox, "Calculating the Fatigue Strength of a Welded Joint Using Fracture Mechanics," *Met. Constr. Br. Weld. J.* 2: 327–31 (August 1970).

128. R. J. Mattos and F. V. Lawrence, *Estimation of the Fatigue Crack Initiation Life in Welds Using Low Cycle Fatigue Concepts*, SAE Sp. Publ. 424, Society of Automotive Engineers, Warrendale, Pa., 1977.

129. F. V. Lawrence, Jr., R. J. Mattos, Y. Higashida, and J. D. Burk, "Estimating the Fatigue Crack Initiation Life of Welds," pp. 134–58 in *Fatigue Testing of Weldments, Am. Soc. Test. Mater. Spec. Tech. Publ.* 648, American Society for Testing and Materials, Philadelphia, 1978.

130. P. C. Paris, "The Fracture Mechanics Approach to Fatigue," pp. 107–26 in *Fatigue — An Interdisciplinary Approach*, Syracuse University Press, New York 1964.

131. P. C. Paris and F. Erdogan, *A Critical Analysis of Crack Propagation Laws*, Am. Soc. Mech. Eng. Pap. 62-WA-234, The American Society of Mechanical Engineers, New York, 1962.

132. W. G. Clark, *Fatigue Crack Growth Characteristics of Heavy Section ASTM A533 Grade B, Class 1 Steel Weldments*, Scientific Paper 70-1E7-FMPWR-56, Westinghouse Research Laboratories, Pittsburgh, February 1970.

133. P. C. Paris et al., "Extensive Study of Low Fatigue Crack Growth Rates in A533 and A508 Steels, pp. 141–76 in *Stress Analysis and Growth of Cracks, Am. Soc. Test. Mater. Spec. Tech. Publ.* 513, American Society for Testing and Materials, Philadelphia, 1972.

134. W. G. Clark, "Effect of Temperature and Section Size on Fatigue Crack Growth in Pressure Vessel Steel," *J. Mater.* 6: 134–39 (March 1971).

135. T. Kondo et al., "Fatigue Crack Propagation Behavior of ASTM A533B and A302B Steels in High Temperature Aqueous Environment," Paper 6, HSST Program 6th Annual Information Meeting, April 25–26, 1976, Oak Ridge National Laboratory, Oak Ridge, Tenn.

136. T. R. Mager and V. J. McLoughlin, *The Effect of an Environment of High Temperature Primary Grade Nuclear Reactor Water on the Fatigue Crack Growth Characteristics of A533 Grade B Class 1 Plate and Weldment Material*, WCAP-7776, HSST Tech. Rep. 16 (October 1971).

137. W. H. Bamford et al., "Effect of High-Temperature Primary Reactor Water on the Subcritical Crack Growth of Reactor Vessel Steels," *Heavy Section Steel Technol. Prog. Quart. Prog. Rep.* ORNL/NUREG/TM-28, -49, -64, -94, -120, -147, -166, and -194 (1976–1977).

138. S. J. Maddox, "Fatigue Crack Propagation Data Obtained from Parent Plate, Weld Metal, and HAZ in Structural Steels," *Weld. Res. Int.* 4: 36–60 (January 1974).

139. T. A. Smith and R. G. Warwick, "A Survey of Defects in Pressure Vessels Built to High Standards of Construction and Its Relevance to Nuclear Primary Circuits," *Int. J. Pressure Vessels and Piping* 2: 283–322 (October 1974).

140. C.A.G. Phillips and R. G. Warwick, *A Survey of Defects in Pressure Vessels Built to High Standards of Construction,* AHSB(S)R-162 (1968).

141. S. H. Bush, "Pressure Vessel Reliability," *J. Pressure Vessel Technol.* 97:54–70 (February 1975).

142. J. Decock, "Determination of Stress Concentration Factors and Fatigue Assessment of Flush and Extended Nozzles in Welded Pressure Vessels," pp. 821–34 in *2d Int. Conf. Pressure Vessel Technol.*, American Society of Mechanical Engineers, New York, 1973.

143. M. E. Mayfield, E. C. Rodabaugh, and R. J. Eiber, *Relevance of Fatigue Tests to Cold Leg Piping,* NUREG/CR-0325 (September 1978).

144. H. Wantanabe, *BWR Feedwater Nozzle/Sparger Final Report,* NEDO-21821 (March 1978).

145. Private communication, W. J. O'Donnell, O'Donnell and Associates, Inc., Pittsburgh, Pa., to Keith Booker, Oak Ridge National Laboratory, Oak Ridge, Tenn., February 1979.

146. Private communication, S. Bush, Battelle Northwest Laboratory, Richland, Wash., to H. B. Meieran, O'Donnell and Associates, Inc., Pittsburgh, Pa., January 1979.

APPENDIX A

SCOPES FOR APPLICABLE *ASME Boiler and Pressure Vessel Code* SECTIONS

APPENDIX A

SCOPES FOR APPLICABLE ASME B&PV CODE SECTIONS

ASME-II-A

This Section provides the material specifications for ferrous materials.

ASME-II-B

This Section provides the material specifications for nonferrous materials.

ASME-III-1-NB

(a) Subsection NB contains rules for the material, design, fabrication, examination, testing, overpressure relief, marking, stamping, and preparation of reports by the Certificate Holder of items which are intended to conform to the requirements for Class 1 construction.

(b) The rules of Subsection NB cover the requirements for strength and pressure integrity of items, the failure of which would violate the pressure retaining boundary. The rules cover deterioration which may occur in service as a result of corrosion, radiation effects, or instability of material.

ASME-III-Appendix I

This appendix contains the design stress intensity values, allowable stresses, material properties, and design fatigue curves for materials allowed in the construction of ASME-III-1-NB components.

ASME-III-Appendix II

This appendix applies to the critical or governing stresses in parts for which theoretical stress analysis is inadequate or for which design rules are unavailable shall be substantiated by experimental stress analysis.

Reevaluation is not required for configurations for which there are available detailed experimental results that are consistent with the requirements of this appendix.

The test procedures followed and the interpretation of the results shall be such as to discount the effects of material added to the thickness of members such as corrosion allowance or of other material which cannot be considered as contributing to the strength of the part.

Tests conducted in accordance with this appendix need not be witnessed by the inspector. However, a detailed report of the test procedure and the results obtained shall be included with a stress report. The report shall show that the instrumentation used was within calibration.

ASME-III-Appendix G

This Appendix presents a procedure for obtaining the allowable loadings for ferritic pressure retaining materials in components. This procedure is based on the principles of linear elastic fracture mechanics. At each location being investigated a maximum postulated flaw is assumed. At the same location the mode I stress intensity factor,[1] K_I, is produced by each of the specified loadings as calculated and the summation of the K_I values is compared to a reference value, K_{I_R}, which is the highest critical value of K_I which can be assured for the material and temperature involved. Different procedures are recommended for different components and operating conditions.

---

[1]The Stress Intensity Factor as used in fracture mechanics has no relation to and must not be confused with the stress intensity used in Articles of this Section. Furthermore stresses referred to in this Appendix are calculated normal tensile stresses not stress intensities in a defect free stress model at the surface nearest the location of the assumed defect.

ASME-V

(a) This Section of the Code contains requirements and methods for nondestructive examination which are Code requirements to the extent they are specifically referenced and required by other Code Sections. These nondestructive examinations methods are intended to detect surface and internal discontinuities in materials, welds, and fabricated parts and components. They include radiographic examination, ultrasonic examination, liquid penetrant examination, magnetic particle examination, eddy current examination, visual examination, and leak testing.

(b) Methods described or referenced are included in Subsection A. Subsection B lists Standards covering nondestructive examination methods which have been accepted as ASME Code Standards, and are included for direct use, or for reference, or as sources of technique details which may be selected, as appropriate, in the preparation of manufacturers' procedures. Acceptance standards for these methods and procedures shall be as stated in the referencing Code Sections.

(c) The nondestructive examination methods included in this Section are applicable to most geometric configurations and materials encountered in fabrication under normal conditions. However, special configurations and materials may require modified methods and techniques, in which case the manufacturer shall develop special procedures which are equivalent or superior to the methods and techniques described in this Code Section, and which are capable of producing interpretable examination results under the special conditions. Such special procedures may be modifications or combinations of methods described or referenced in this Code Section, and shall be proved by demonstration to be capable of detecting discontinuities under the special conditions, which are equivalent to the capabilities of the methods described in this Code Section when used under more general conditions. Depending on the quality assurance or quality control system requirements of the referencing Code Section, these special procedures shall be submitted to the Inspector for approval where required, and shall be adopted as part of manufacturer's quality control program.

ASME-IX

The rules in this Section apply to the preparation of welding procedure specifications, and the qualification of welding procedures, welders, and welding operators for all types of manual or machine welding processes.

ASME-XI-1

(a) This Division provides rules and requirements for inservice inspection of Class 1, 2, and 3 pressure retaining components (and inservice testing of pumps and valves in light-water cooled nuclear power plants.

(b) This Division categorized the areas subject to inspection and defines responsibilities, provisions for accessibility, examination methods and procedures, personnel qualifications, frequency of inspection, record keeping and reporting requirements, procedures for evaluation of inspection results and subsequent disposition of results of evaluation, and repair requirements.

(c) This Division provides for the design, fabrication, installation, and inspection of replacements.

ASME-XI-1-Appendix A

This Appendix provides a procedure for determining the acceptability of flaws that have been detected during inservice inspection (excluding preservice in-inspection) that exceed the allowable flaw indication standards. The procedure is based upon the principles of linear elastic fracture mechanics.

This procedure applies to ferritic materials 4 in. (102 mm) and greater in thickness with specified minimum yield strengths of 50.0 ksi (345 MPa) or less in components having simple geometries and stress distributions. The basic concepts of the procedure may be extended to other ferritic materials (including clad ferritic materials) and more complex geometries; however, they are not intended to apply to austenitic or high nickel alloys. For purposes of analysis, indications that exceed the standards are considered as cracks or flaws.

APPENDIX B

SUMMARY OF FATIGUE AND/OR VIBRATION-INDUCED FAILURES IN REACTOR COMPONENTS WELDS

APPENDIX B

SUMMARY OF FATIGUE AND/OR VIBRATION-INDUCED FAILURES IN REACTOR COMPONENTS WELDS

NOTES FOR TABLE B1

Utilities	Full Name
Bal. G&E	Baltimore Gas & Electric
N.E. Nucl. Energy	Northeast Nuclear Energy
Ark. P.&.L.	Arkansas Power & Light
Con. Ed.	Consolidated Edison
N. States Power	Northern States Power
Cons. Pwr.	Consumer's Power Co.
Niag. Moh. Pwr.	Niagra Mohawk Power
PSE&G	Public Service Electric & Gas
Met. Ed.	Metropolitan Edison

Table Bl. Summary of Fatigue and/or Vibration Induced Failures in Reactor Components Welds
Ref. NRC-NSIC-9711-1, 1-9-79 Request

Item No.	Survey No.	Accession No.	Reactor Type	Reactor Mfg.	Reactor Name	Utility	Date of RPT	Date of Event	Source of Fatigue Event	Component Description	Corrective Action
1	21	138334	PWR	CE	Cal Cliffs-1	Bal. G&E	1978	4-1-78	Vibration-Pump	RCF M-S Pressure Line	Ground, temp. plugged
2	27	137251	BWR	GE	Millstone-1	N.E. Nucl. Energy	1978	3-19-78	Fatigue	Vent Line	
3	38	135885	PWR	BW	Ark. Nucl-1	Ark. P&L	1978	12-3-77	Vibration	Vent Line	Replaced
4	60	129369	BWR	GE	Millstone-2	N.E. Nucl. Energy	1977	2-19-77	Fatigue-Long	Safety Inj. Valve	Ground, reweld
5	62	129314	PWR	W	Ind. Pt-2	Con.Ed	1977	4-27-77	Vibration-Excess	RHR Pump	- - -
6	73	126009	PWR	GE	Dresden-3	Comm. Ed	1977	6-7-77	Fatigue	Mois. Sep. Tank	- - -
7	76	125433	PWR	CE	Cal Cliffs-2	Bal. G&E	1977	5-11-77	Vibration-Pump	Cool. Pump Line	- - -
8	82	125083	PWR	CE	Cal Cliffs-2	Bal. G&E	1977	5-1-77	Vibration-Pump	Cool. Pump Line	- - -
9	83	125082	PWR	CE	Cal Cliffs-2	Bal. G&E	1977	4-27-77	Vibration-Pump	Disch. Line-Control Tank	-
10	84	125070	PWR	W	Ind. Pt.-2	Con.Ed	1977	4-27-77	Vibration-HX	RHR Pump Seal	Removed
11	87	124327	BWR	GE	Millstone-2	N.E. Nucl. Energy	1977	4-20-77	Fatigue	Pressure Tap	Yes (2)
12	98	122317	PWR	CE	Cal Cliffs-2	Bal. G&E	1977	1-15-77	Vibration-Excess	Control Tank Line	W. R. TBM (3)
13	100	122141	PWR	W	Ind. Pt.-2	Con.Ed	1977	2-1-77	Vibration	Vent Valve	Vent removed, cap
14	109	120489	PWR	CE	Cal Cliffs-1	Bal. G&E	1976	12-7-76	Fatigue Pump (Thermal)	CVCS Charge Pump	Yes
15	110	120480	PWR	CE	Cal Cliffs-2	Omaha Public Pwr.	1976	12-7-76	Vibration-Pump	CVCS Charge Pump	Yes
16	114	119781	PWR	CE	Millstone-2	N.E. Nucl. Energy	1976	11-12-76	Vibration	Pipe-Vol Cont. Tank (1)	Ground
17	125	118733	PWR	CE	Calhoun-1	Omaha	1976	10-12-76	Vibration	Pump Bleep Off Line	- - -
18	127	118260	PWR	CE	Millstone-2	N.E. Nucl. Energy	1976	7-1-76	Fatigue	Instr. Line	Ground, reweld
19	128	118244	PWR	BW	Ark. Nucl-1	Ark. P&L	1976	9-27-76	Vibration-Excess	Decay Heat Dislave Line	
20	129	118243	PWR	BW	Ark. Nucl-1	Ark. P&L	1976	9-27-76	Vibration	Decay Heat Center	Yes
21	137	114638	BWR	GE	Monticello	N. States Power	1976	6-11-76	Fatigue	Mois. Sep. Drain Line	Yes
22	138	114081	BWR	GE	Peach Bottom-3	Philadelphia Elec.	1976	5-21-76	Vibration-Pump	Recir.Pump Instr Line	Reweld
23	142	112678	PWR	W	Ind. Pt.-3	Con.Ed	1976	3-31-76	Fatigue-Poor Weld	Acc. Vent Line	- - -
24	145	112155	PWR	CE	Millstone-2	N.E. Nucl. Energy	1976	3-17-76	Fatigue-high Cycle	Coolant Pump	Ground, reweld
25	146	112148	PWR	W	Cook-1	Ind/Mich. Pwr.	1976	3-15-76	Vibration	Instr. Pipe	Ground, reweld
26	153	109579	PWR	BW	Oconee-3	Duke Power	1975	12-17-75	Vibration	LPCI Sample Line	Replaced
27	158	108854	PWR	CE	St. Lucie-1	Florida P&L	1975	12-23-75	Vibration	Piping	Restraints
28	164	108248	BWR	GE	Millstone-1	N.E. Nucl. Energy	1975	11-25-75	Fatigue	Instr. Pipe	Replaced w/long pipe
29	165	108081	BWR	GE	Peach Bottom-2	Philadelphia Elec.	1975	11-17-75	Vibration	1 in. Pipe	Repaired, add Support
30	170	106976	PWR	CE	Millstone-2	N.E. Nucl. Energy	1975	10-14-75	Vibration	Instr. Root Tap-Reactor Cool	--
31	183	103874	PWR	W	Cook-1	Ind/Mich. Pwr.	1975	6-30-75	Vibration	Eq. Line Valve	Redesign
32	185	103096	BWR	GE	Quad Cities-2	Comm. Ed	1975	5-30-75	Vibration	3/4 in. Test Connec. Line	Removed, Plugged
33	193	102125	PWR	W	Cook -1	Ind/Mich. Pwr.	1975	4-10-75	Vibration-Excess	Isolation Valve	Removed, reweld
34	193	102125	PWR	W	Cook -1	Ind/Mich. Pwr.	1975	4-10-75	Vibration-Excess	Isolation Valve	Removed, reweld
35	193	102125	PWR	W	Cook -1	Ind/Mich. Pwr.	1975	4-10-75	Vibration-Excess	Isolation Valve	Removed, reweld
36	201	099707	BWR	GE	Br. Ferry-1	TVA	1975	2-15-75	Vibration	3/4 in. C-Steel Test Connec.	Ground, reweld
37	213	097597	PWR	BW	Ark. Nucl-1	Ark. P&L	1974	11-26-74	Vibration-Pump	Decay Heat Pipe	Flow orifice
38	218	097074	PWR	CE	Palisades	Cons. Pwr	1974	9-9-74	Fatigue	Letdown Line Vent	Removed
39	220	096480	BWR	GE	Quad Cities-1	Comm. Ed	1974	10-22-74	Vibration-Slag	1 in. C-Steel Instr. Line	Ground, reweld

(1) Bimetallic Failure
(2) Multiple Failures of Same Weld (failure of a reweld which replaced original weld which had failed).
(3) W.R. TBM = Weld Repair To Be Made

Table B1. (Continued)

Item No.	Survey No.	Accession No.	Reactor Type	Reactor Mfg.	Reactor Name	Utility	Date of RPT	Date of Event	Source of Fatigue Event	Component Description	Corrective Action
40	221	096426	PWR	BW	Ark. Nucl.-1	Ark. P&L	1974	10-28-74	"ibrat. .	Cold Leg Drain Pipe	Ground, reweld
41	223	095884	BWR	GE	Br. Ferry-1	TVA	1974	10-2-74	Vibrati . . 'low	1 in. SS Recirc. Line	Split Sleeve Hgr mod.
42	223	095884	BWR	GE	Br. Ferry-1	TVA	1974	10-2-74	Vibration-Flow	Sensing Lines	
43	230	095026	PWR	W	Ginna	Rochester G&F	1974	8-12-74	Fatigue-Design	Vent Pipe Nipple	Larger weld, relocate
44	239	093545	BWR	GE	Fitzpatrick	Niag. Moh Pwr.	1975	1-29-75	Vibration	LPCI Drain Line	Pipe length
45	244	090422	PWR	CE	Ft. Calhoun	Omaha Public Pwr.	1974	4-15-74	Vibration	Flow element Sensing	Welds repaired, alt. fill proced.
46	245	089742	BWR	GE	Dresden-2	Comm. Ed	1974	4-4-74	Fatigue	C.R.D. Water Filter	Ground, pad weld
47	252	087290	PWR	W	Zion-1	Comm. Ed	1973	12-6-73	Vibration-Pump	Flow meter orifice tap	Replaced,reweld
48	254	087013	PWR	W	Ind. Pt.-2	Con. Ed	1973	10-31-73	Vibration	RHP Vent Valve	Replaced vent Couplings
49	258	085747	PWR	W	Ind. Pt.-2	Con. Ed	1973	11-12-73	Vibration	RHR Vent Valve	Removed, plugged
50	260	084737	BWR	GE	Dresden-3	Comm. Ed	1973	8-16-73	Fatigue	3 in. Pipe Stub	- - -
51	263	083019	PWR	W	Zion	Comm. Ed	1973	8-1-73	Vibration-Pump	304 SS Elbow/Tube	- - -
52	267	081483	PWR	W	Zion-1	Comm. Ed	1973	6-20-73	Vibration-Pump	3/4 in Pipe/Dis.Valve	Replaced, Add support
53	270	080745	PWR	W	Ind. Pt-2	Con. Ed	1973	5-25-73	Vibration	RHR Pipe	Ground, reweld, Short.
54	1	140458	PWR	CE	Cal Cliffs-2	Bal G&E	1978	6-25-78	Vibration-Pump/ Ther. Cycling	Y Connec. Iso. Valve	Repaired
55	2	139994	PWR	CE	Ft. Calhoun-1	Omaha Public Pwr.	1978	6-20-78	Vibration	Water inlet Valve	Ground, reweld, mod.
56	3	139820	PWR	CE	Cal Cliffs-1	Bal G&E	1978	5-31-78	Vibration/Therm. Cycling	Pump Flange	Repaired
57	4	138966	PWR	CE	Ft. Calhoun	Omaha Public Pwr.	1978	5-20-78	Vibration	Charging Header	Ground, repaired
58	5	137912	PWR	CE	Cal Cliffs-1	Bal G&E	1978	4-12-78	Vibration	Pump Valve	Add Pipe Support
59	5	137912	PWR	CE	Cal Cliffs-1	Bal G&E	1978	4-12-78	Vibration	Pump Valve	Add Pipe Support
60	6	137879	PWR	CE	Ft. Calhoun	Omaha Public Pwr.	1978	4-8-78	Vibration	Pump Valve	Ground, reweld, mod.
61	7	137832	PWR	W	Turkey Pt-4	Florida P&L	1978	3-20-78	Pressure Cycling	Pump Relief Line	Weld repair
62	10	135993	PWR	CE	Ft. Calhoun	Omaha Public Pwr.	1978	2-15-78	Vibration	Pump Valve	Ground, reweld, mod.
63	11	135111	BWR	GE	Farley-1	Alabama Pwr.	1978	12-31-77	Vibration	Steam Supply Line	Repaired
64	13	134516	PWR	W	Ind. Pt.-3	Con. Ed	1978	1-10-78	Vibration	CVCS Line	Repaired
65	14	134515	PWR	CE	Ft. Calhoun	Omaha Public Pwr.	1978	1-17-78	Vibration	Pump Valve	Ground, reweld, mod.
66	15	134505	PWR	W	Ginna	Rochester G&E	1978	1-11-78	Fatigue	Pump Pipe	Cut, replaced with larger fillet
67	17	132742	PWR	CE	Cal Cliffs-2	Bal. G&E	1977	6-30-77	Vibration	Pump Valve	Repaired
68	18	129845	PWR	CE	Cal Cliffs-2	Bal. G&E	1977	8-25-77	Vibration	Pump Valve	Gasket, repair
69	19	129796	PWR	CE	Cal Cliffs-2	Bal. G&E	1977	9-19-77	Vibration	Pump Valve	Repaired
70	21	129572	PWR	CE	Cal Cliffs-2	Bal. G&E	1977	9-6-77	Vibration Pump/ Therm. Cycle	Pump Pipe	Repaired
71	22	129313	PWR	W	Ind. Pt.-2	Con.Ed	1977	3-29-77	Vibration-Excess	Pump Valve	
72	23	129312	PWR	W	Ind. Pt.-2	Con.Ed	1977	3-9-77	Fatigue	Clamp	Repaired
73	24	126477	PWR	CE	Cal-Cliffs-2	Bal. G&E	1977	6-13-77	Vibration	Pump Line	Modified
74	25	124874	PWR	W	Ind. Pt.-2	Con.Ed	1977	3-15-77	Vibration-Excess	Pump Valve	
75	26	124873	PWR	W	Ind. Pt.-2	Con.Ed	1977	3-11-77	Vibration-Excess	Pump Elbow	

Table B1. (Continued)

Item No.	Survey No.	Accession No.	Reactor Type	Reactor Mfg.	Reactor Name	Utility	Date of RPT	Date of Event	Source of Fatigue Event	Component Description	Corrective Action
76	33	121655	PWR	CE	Millstone-2	N.E. Nucl. Energy	1977	12-12-76	Fatigue	Pump Valve	Ground, reweld
77	35	119523	PWR	CE	Cal Cliffs-2	Bal. G&E	1976	10-22-76	Thermal Fatigue	Pump Valve Inlet Line	Temp. weld repair
78	36	119520	PWR	BW	Ark. Nucle-1	Ark. P&L	1976	11-1-76	Vibration	Pump Relief Line Soc. Weld	Repaired
79	37	118734	PWR	W	Ind. Pt.-3	Con.Ed	1976	10-1-76	Fatigue	Charge Pump Vent Valve	Flange replaced
80[1]	40	117062	PWR	CE	Millstone-2	N.E. Nucl. Energy	1976	8-23-76	Fatigue	Charge Pump Relief Valve	Ground, reweld
81	42	116872	PWR	W	Pt. Beach-2	Wis./Mich. Pwr. Co.	1976	8-12-76	Vibration	Charge Pump Borate Line	Ground, reweld
82	47	112159	PWR	W	Salem-1	PSE & G	1976	3-31-76	Vibration-Pump	Charge Pump 4 in. Line	Replaced
83	48	112156	PWR	CE	Millstone-2	N.E. Nucl. Energy	1976	3-17-76	Fatigue	Charge Pump Relief Valve	Reweld
84	48	112156	PWR	CE	Milistone-2	N.E. Nucl. Energy	1976	3-17-76	Fatigue	Charge Pump Relief Valve	Reweld
85	50	111787	PWR	W	Yankee Rowe	Yankee Atomic	1976	3-4-76	Vibration	Charge Pump Relief Valve Flange	Flange replaced
86	55	107224	BWR	GE	Quad Cities-2	Comm. Ed	1975	10-17-75	Vibration	Feedwater Flush Line	Ground, reweld
87	56	106311	PWR	BW	3 Mile Island-1	Met Ed	1975	8-29-75	Vibration	Makeup Pump Vent Line	Repaired with guides
88	57	105546	PWR	W	Ind. Pt.-2	Con Ed	1975	8-18-75	Vibration	Charge Line Vent Connec.	Vent shortened
89	58	104995	PWR	W	Yankee Rowe	Yankee Atomic	1975	8-15-75	Vibration-Pump	Charge Pump Relief Valve Flange	Valve flange replaced
90	61	102538	PWR	W	Zion-2	Comm. Ed	1975	4-17-75	Fatigue	Charge Pump Indicator Pipe	- - -
91	62	102536	PWR	W	Zion-2	Comm. Ed	1975	4-28-75	Fatigue	Charge Pump Indicator Pipe	Press. pulsating filter installed
92	64	100889	BWR	GE	Quad Cities-2	Comm. Ed	1975	3-21-75	Vibration Flow, Thermal	Feedwater Sparger	New design sparger
93	69	097756	PWR	W	Zion-2	Comm. Ed	1974	11-27-74	Pressure Pulsations	Charge Pump Indicator Pipe	Repairs later
94	70	097105	PWR	W	Zion-2	Comm. Ed	1974	11-1-74	Vibration	Charge Pump Indicator Pipe	Add restraints
95	71	095592	BWR	GE	Dresden-3	Comm. Ed	1974	9-26-74	Vibration	Pipe Support	Ground, reweld
96	75	093253	PWR	W	Kewaunee	Wis. Pub. Serv.	1975	1-27-75	Vibration	Conn. Tee Vent Valve Pipe	Pulsation dampener

[1] Multiple failures of same weld (failure of a reweld which replaced original weld which had failed.)

NUREG/CR-1351
ORNL/NUREG-64
Distribution
Category R5

INTERNAL DISTRIBUTION

1–2. Central Research Library
3. Document Reference Section
4–5. Laboratory Records Department
6. Laboratory Records, ORNL RC
7. ORNL Patent Section
8. Nuclear Safety Information Center
9–17. M. K. Booker
18. R. S. Booth
19. C. R. Brinkman
20. R. H. Bryan
21. W. R. Corwin
22. J. E. Cunningham
23. G. R. Gessel
24. G. M. Goodwin
25. R. J. Gray
26. P. M. Haas
27. R. F. Hibbs
28. M. R. Hill
29. P. P. Holz
30. J. A. Horak
31. C. J. Long, Jr.
32. R. W. McClung
33. H. E. McCoy, Jr.
34. J. W. McEnerney
35. S. E. Moore
36. F. R. Mynatt
37. R. K. Nanstad
38. G. W. Parker
39. P. Patriarca
40. H. E. Reesor
41. J. E. Selle
42. G. M. Slaughter
43. J. H. Smith
44. J. P. Strizak
45. R. W. Swindeman
46. G. D. Whitman

EXTERNAL DISTRIBUTION

47. NRC, Office of Nuclear Regulatory Research, Washington, DC 20555
    Program Sponsor

48. DOE, Oak Ridge Operations Office, P.O. Box E, Oak Ridge, TN 37830
    Office of Assistant Manager, Energy Research and Development

49–50. DOE, Technical Information Center, P.O. Box 62, Oak Ridge, TN 37830

51–410. For Distribution Category R5.

☆ U.S. GOVERNMENT PRINTING OFFICE: 1980-640-245-439