Enclosure 2

June 1982

THREADED FASTENER EXPERIENCE IN NUCLEAR POWER PLANTS

by

William H. Koc
Operating Reactors Assessment Branch
Division of Licensing

31 pp

TABLE OF CONTENTS

ABSTRACT

This report identifies forty-four (44) incidents of threaded fastener degradation and failure in nuclear power plants for the period from 1964 to March 1982. It provides an overview of some of the various types of threaded fastener problems that have occurred since 1964. Safety implications of these incidents are discussed and recommended short term Regulatory actions and ongoing long term Regulatory actions are described. It should be noted that information included in this report represents the current NRC staff understanding of each issue.

ACKNOWLEDGEMENTS

The author wishes to thank K. Wichman, W. Hazelton, D. Seller, H. Conrad, and B. Turovlin of NRR, J. Collins of IE, and H. Vander Molen of DST, for their input to this report, Sherry Holden and Carolyn Wilson for their excellent typing job.

LIST OF FIGURES

1.0 INTRODUCTION

There are numerous threaded fasteners* in a nuclear power plant. The most important applications are those constituting an integral part of the reactor coolant pressure boundary such as pressure retaining closures in reactor vessels, pressurizers, reactor coolant pumps, and steam generators. In recent years, there have been an increasing number of degraded threaded fastener incidents reported in both operating reactors and reactors under construction. A large number of reported threaded fastener incidents involve reactor coolant pressure boundary applications and major component supports. Therefore, there is increasing concern regarding the integrity of the reactor coolant pressure boundary in operating nuclear power plants and the structural integrity of component supports.

The scope of this report is limited in that there is no intent to describe each event in detail and only relatively significant events are covered. This report covers a total of forty-four (44) threaded fastener incidents reported by the licensees of operating nuclear power plants and the applicants of plants still under construction during the period of 1964 to March 1982. This information is derived from pertinent Licensee Event Reports (LER), Reportable Occurrence Reports, Operating Reactor Event Memoranda (OREM), failure analysis reports and other relevant documents. As stated earlier, the coverage of the threaded fastener incidents is not exhaustive and may be incomplete; only relatively significant events that occurred during this period of time have been investigated by the staff.

---

*Bolts, studs, capscrews, etc.

This report was initiated as a result of the Executive Director for Operations response to R.F. Fraley's memorandum dated October 20, 1981 regarding the safety concern of threaded fastener failures in nuclear power plants (References 1 and 2). This report provides a perspective and an overview of threaded fastener problems in operating nuclear power plants and describes recommended short term Regulatory actions and ongoing long term Regulatory actions addressing this problem.

2.0 DESCRIPTION OF THREADED FASTENER PROBLEMS

2.1 Types of Threaded Fastener Degradation and Failures

Threaded fastener failures and degradation described in this report refer to the loss of the integrity of threaded fasteners including bolts, studs and capscrews due to any mechanical, chemical or electro-chemical causes. The principal types or modes of threaded fastener degradation and failures discussed in this report are defined below:

Stress Corrosion

Failure of material due to stress corrosion refers to cracking caused by the simultaneous presence of tensile stress and a specific hostile environment. In this report all cracking that occurs in a hostile environment under steady state tensile load is classified as stress corrosion cracking. Stress corrosion cracking is a concern because it can occur below the design stress.

Fatigue

Failure of material under repeated cyclic loads is defined as fatigue. Corrosion fatigue refers to reduction of fatigue resistance due to the presence of a corrosive or hostile environment. This report

2.2.1 Stress Corrosion

For all the threaded fastener incidents covered in this report, stress corrosion is the most common cause of failure. A total of nineteen (19) threaded fastener incidents attributed to stress corrosion are listed in Table 1. Relevent information when available such as material of parts, contributing factors, and the corrective action taken are also tabulated. It should be noted that only one threaded fastener incident (LaCrosse) was reported by a boiling water reactor (BWR); all the other incidents were experienced by pressurized water reactors (PWR).

Table 1 lists six (6) incidents associated with the cracking of primary pressure boundary closure studs: the failure of steam generator manway studs in Arkansas Unit 1 (1978 and 1981), Oconee Unit 3 (1980), San Onofre Unit 1 (1977), and Maine Yankee (1982), and the failure of reactor vessel closure studs in LaCrosse (1970).

During a 1970 refueling outage at LaCrosse, two reactor vessel closure studs failed in a head removal operation after applying a tension of 46 ksi and 35 ksi respectively. The main cause for the failure of two reactor vessel closure studs made of 12% Cr martensite stainless steel was attributed by the licensee to exposure to an aqueous environment while under load during an outage. The other contributing factors were improper heat treatment of the material resulting in a high susceptibility to intergranular stress corrosion and galvanic action resulting from localized breakdown of the silver plating on the threads.

During the fifth refueling outage at San Onofre Unit 1 in 1977, selected components of the reactor primary coolant system were subjected to an in-service inspection. Eight steam generator manway studs were visually identified to be showing crack indications. The cause of the cracking was not determined.

Two steam generator manway studs were sheared off at the base of the nut at Arkansas Unit 1 in 1978 during re-installation of the lower manway cover following tube plugging in steam generator A. Visual inspection indicated a crack of 90% of the stud diameter had existed prior to retensioning. UT inspection on other studs of both steam generator upper and lower manways did not show any crack indications. The cause of the stud failure was not determined because meaningful failure analysis could not be performed on the excessively deteriorated fracture surface of the failed studs.

During a scheduled steam generator inspection in Oconee Unit 3 in 1980, cracking in nine of sixty-four upper and lower manway studs was identified by visual and UT examination. At Arkansas Unit 1 during an outage as a result of a steam generator tube leak in 1980, UT inspection identified three lower manway studs with crack indications on "A" steam generator. For the steam generator manway stud failures at Arkansas Unit 1 and Oconee Unit 3, the cause was attributed to the use of thread lubricant containing molybdenum disulfide (MoS_2). Molybdenum disulfide is known to decompose at high temperature to form corrosive hydrogen sulfide. This resulted in accelerated cracking of the closure studs. Experiments performed at Brookhaven National Laboratory have shown a pronounced embrittling effect on carbon steel and low alloy steel when the material is in contact with MoS_2 in a steam environment.

During a routine disassembly of steam generator #2 primary manway in Maine Yankee in 1982, six of twenty manway studs had failed with another five showing crack indications. The studs were exposed to leaking borated water and Furmanite, a sealing compound containing leachable sulfur, fluorine, and chlorine which are known SCC promoters. The cause of leakage was due to an interference contact between the gasket retainer lip and vessel cladding which prevented proper compression of the flexitablic gasket during reinstallation of the manway cover. Furmanite was injected to the manway to control the leakage when increasing the torque in the studs to hydrotest level failed to stop the leakage. UT inspection on steam generator #1 and #3 manway studs did not show any indications. The cause of the failure of steam generator manway studs at Maine Yankee is currently under study.

Four incidents occurring at Surry Units 1 and 2 (1975) and Prairie Island Units 1 and 2 (1980) are related to the cracking of threaded fasteners in steam generator supports. All the threaded fasteners were made of maraging steel (Vascomax 250). The exact cause for the failure at Surry Units 1 and 2 (1975) was not determined. The failure of steam generator column support bolts at Prairie Island Units 1 and 2 was attributed to an excessive pretorque of 1400 ft-lb. Laboratory test results have shown that high strength maraging steels and low alloy steels heat treated to high hardness are highly susceptible to stress corrosion especially under large preload.

Four incidents reported by Ginna (1970), Haddam Neck (1973), Midland Unit 1 (1979), and Palo Verde (1981) are related to failure of imbeded anchor bolts in steam generator supports, reactor vessel skirt flange and piping restraints, respectively. The exact cause for the failure of steam generator support anchor bolts at Ginna and Haddam Neck was not determined. Midland Unit 1 and Palo Verde are plants still under construction and the cause of their anchor bolt failures is attributed to excessive hardness as a result of improper heat treatment of the low alloy steel material.

One threaded fastener incident related to some valve studs made of stainless steel type 416 in Rancho Seco (1980) is also attributed to improper heat treatment of the material. The studs of main steam isolation valve internals made of AISI 4140 material failed at D.C. Cook Unit 1 in 1981. The failure was attributed to over-torque during installation and the use of thread lubricant containing molybdenum disulfide (MoS_2) as the probable causes.

The failure of two bonnet to body studs of a 6 inch gate valve in the spent fuel cooling system during a routine disassembly for maintenance was reported by Main Yankee (1982). The studs were made of type 416 stainless steel material and were corroded due to exposure to boric acid from a small body-to-bonnet leak. The licensee reported that there are about 150 valves with type 416 stainless steel studs in the plant. Inspection of 12 such valves did not show any evidence of borated water corrosion. The licensee has replaced the degraded studs with AISI 4140 low alloy steel studs and in the longer term, all AISI 4140 studs will be replaced with studs made of 17-4 PH material.

Several broken thermal shield bolts in the reactor vessel internals of Oconee Unit 1 were visually observed while performing a 10 year ISI during a refueling shutdown. Subsequent UT inspection identified

94 of a total of 96 bolts showing crack indications. The results of failure analys s on the broken thermal shield bolts of Oconee Unit 1 had suggested stress corrosion as well as fatigue to be potential failure mechanisms. The fracture surfaces of the broken bolts were so severely corroded that it was not possible to determine which of the two failure mechanisms was responsible. The corrective action for Oconee Unit 1 consists of a redesign of the lower thermal shield and use of Inconel X-750 material for bolts and nuts.

UT inspection performed on reactor vessel thermal shield bolts of Oconee Unit 2 also identified three broken bolts and another twenty-four bolts showing crack indications. The cause of thermal shield bolt failures in Oconee Unit 2 is currently under study.

2.2.2 Fatigue

All identified fatigue failures of threaded fasteners in nuclear power plants have been associated with reactor vessel internals. Table 2 lists three threaded fastener failure incidents attributed to fatigue, namely, the failure of thermal shield bolts in Big Rock Point and Yankee Rowe, and the failure of hold down bolts for the ring shim in Palisades. The failure of hold down bolts in Palisades was identified after discovering a broken bolt head in the steam generator #2 inlet plenum; the cause for the failure was attributed to improper pretorque of the bolts. The cause for the thermal shield bolt failures in Big Rock Point (1964) and Yankee Rowe (1968) was attributed to the flow induced vibration.

2.2.3 Borated Water Corrosion

Borated water corrosion is the second most numerous type of threaded fastener failure or degradation covered in this report. It occurs only in pressurized water reactors. A total of fourteen (14) threaded fastener incidents resulting from borated water corrosion are listed in Table 3. Almost in every case, the cause of threaded fastener degradation is due to corrosive attack by borated water leaking from closure gaskets or seals, and the degraded fasteners were discovered during the process of correcting the leakage. With prolonged exposure to borated water, the affected closure studs or hold down bolts can be corroded sufficiently to impair their load carrying capability.

One of the severe cases of borated water threaded fastener degradation reported was a reactor coolant pump closure stud made of low alloy steel (AISI 4140) at Fort Calhoun. The diameter of the stud was reduced from 3.5 inches to 1.1 inches.

Among the fourteen incidents resulting from borated water corrosion, six incidents related to reactor coolant pump closure studs occurred in Fort Calhoun (1980)(1981), Calvert Cliffs Units 1 (1980) and 2 (1981), and Oconee Units 2 (1981) and 3 (1981); three incidents related to steam generator manway closure studs occurred in Arkansas 1 (1981), Calvert Cliffs Unit 1 (1981) and St. Lucie (1977); two incidents related to pressurizer manway studs occurred in St. Lucie (1978) and Calvert Cliffs Unit 2 (1981); and three incidents related to various types of valve studs occurred in Calvert Cliffs Unit 2 (1981), Kewaunee (1981) and D.C. Cook Unit 2 (1981).

2.2.4 Erosion-Corrosion

One threaded fastener incident resulting from erosion-corrosion by borated water is listed in Table 4. This incident occurred at Zion 1 in 1979 which resulted in leakage from a valve bonnet joint in the chemical and volume control system (CVCS). The result of the failure analysis concluded that the failure of the bolts to seal the bonnet was probably caused by "wire drawing" as a result of improper assembly of the bonnet.

2.2.5 Other Threaded Fastener Degradation and Failures

A total of seven threaded fastener incidents which cannot be classified into the previous categories are listed in Table 5. At Sequoyah Units 1 and 2 in 1977, some steam generator support bolts failed during hammering for seating into helicoils. Both plants were still under construction at that time. This premature failure of the support bolts was attributed by the applicant to the presence of quench cracks as a result of improper heat treatment of the bolting material during fabrication.

Reactor coolant pump support bolt failures occurred at Waterford in 1981. Waterford is still under construction. The cause of the failure was attributed to the improper torquing of the support bolts and that some bolts were too short. It was also discovered that the torquing equipment was not properly calibrated.

Some motor hold-down bolts on valve limit-torque operators were reported to have failed in Vermont Yankee (1981) and Pilgrim Unit 1 (1981). The exact cause of these failures is not known. One probable cause for the failure in Pilgrim Unit 1 was attributed by the licensee to vibration which loosened the hold-down bolts which subsequently sheared during operation.

The failure of emergency feedwater turbine steam inlet bolts in Arkansas Unit 1 (1980) was due to the bolts made of wrong material, carbon steel (C-1117), instead of the originally specified low alloy steel (AISI 4140). The carbon steel bolts were not strong enough to withstand the water hammers that occurred. Surry Unit 2 reported the failure of a capscrew in a service water pump impeller in 1981. The impeller capscrew was corroded in an aqueous environment and caused the pump to be inoperable. The broken carbon steel capscrew was replaced with a stainless steel capscrew.

2.2.6 Summary of the Causes of Threaded Fastener-Degradation and Failures

Based upon the available information concerning the incidents covered in this report, the major causes for threaded fastener degradation and failures are summarized below:

Stress Corrosion

(1) Borated water leakage

(2) Wet or humid environment

(3) High preload

(4) Use of lubricant containing molybdenum disulfide

(5) Improper heat treatment of material

Fatigue

(1) Flow induced vibration

(2) Improper preload

Borated Water Corrosion and Erosion-Corrosion

(1) Borated water leakage

Other Threaded Fastener Degradation and Failures

(1) Improper heat-treatment

(2) Improper preload

(3) Wrong material

In summary, we can conclude that the majority of the reported threaded fastener problems were corrosion related and caused by a combination of any of the following three factors:

(1) Presence of hostile environment such as borated water, wet or humid environment, and sulfur or chloride contamination.

(2) Application of high preload

(3) Use of material susceptible to stress corrosion cracking such as high strength maraging steel and low alloy steels heat treated to high strength levels.

3.0 SAFETY IMPLICATIONS

Most of the incidents reported in Tables 1 through 5 were discovered during refueling outages, scheduled in-service inspections, or maintenance/repair outages. As a result, such reported incidents have as yet, had no impact on public health and safety. Of those threaded fastener failure incidents discovered during normal operation, there has not been any challenge to plant engineered safety features. In spite of limited safety consequences to date, many incidents involve threaded fasteners that constitute an integral part of the reactor coolant pressure boundary. As shown in Table 6, a total of nineteen of the reported forty-four incidents relate to reactor coolant pressure boundary applications. Degradation and failure of such threaded fasteners constitute a potential loss of integrity of the reactor coolant pressure boundary, and could lead to malfunction or failure of the affected components. In the extreme case, a LOCA could occur if extensive threaded fastener failures in a pressure retaining closure were not detected.

As shown in Table 7, there are a total of eleven (11) threaded fastener incidents related to component supports. Failure of such threaded fasteners will not impair the normal operation of the plant, however, under the extreme loads associated with a LOCA or earthquake, extensive failures of support or anchor threaded fasteners can result in component up-lift and possible failure.

For threaded fastener incidents related to component internals, the major safety concerns are (i) the degradation of the component performance and (ii) the effect of loose parts on the safe operation of the plant when the physically separated internal bolts or studs are not captured. A total of seven (7) threaded fastener incidents related to component internals are listed in Table 8. Although those seven incidents did not result in any problem of component performance or plant operation, they do constitute a potential safety concern.

The safety implications discussed above are a cause for concern, especially in view of the increased number of reported threaded fastener incidents in recent years as shown in Figure 1. This concern is further compounded by the fact that the UT methods used in inservice inspection programs are not sensitive enough to detect initial cracking in the threaded fasteners resulting from stress corrosion and fatigue without development and use of special techniques. The UT sensitivity as required by the calibration standard in ASME section V is not high enough to detect the critical crack size of low alloy high strength steel material which is typically below 0.10 inch. Furthermore, for detecting threaded fastener degradation by borated water corrosion or erosion-corrosion, visual examination at this time is the only reliable method to discover such degradation; in almost all cases this requires disassembly of the component in order to have direct visual access to the threaded fasteners. Therefore, degradation of such visually inaccessible threaded fasteners by borated water corrosion or erosion-corrosion can potentially be left undetected when there is no clear evidence of leakage in the surrounding area. Under the present in-service inspection programs, visual inspection is not a mandatory requirement and UT insepction is also not required on pressure retaining bolts or studs with diameters of less than two inches. Maine Yankee steam generator manway studs for example, have diameters less than two inches.

Since each bolt, stud or capscrew in a component has a unique purpose or function, the loss of service of any particular part threatens the design basis of the component. In turn, this erodes the safety margin for the plant.

4.0 SHORT TERM REGULATORY ACTIONS

The following are recommended actions that could be implemented in the short term and could have significant effect in reducing both the number and severity of threaded fastener failure incidents. The recommended actions should be considered for threaded fasteners in RCPB components, in RCPB component supports and internals, and in other safety system components, when appropriate. Recommendations 3 through 5 are in IE Bulletin No. 82-02 dated June 2, 1982 (Reference 4). The scope of the Bulletin is limited to licensees of operating PWRs and the Bulletin requirements only address RCPB closures.

1. Leakage through reactor coolant pressure boundary bolted closures should be corrected when such leakage is detected. All threaded fasteners of the closure assemblies where leakage has been detected should be examined thoroughly to determine their suitability for continued service.

2. Licensees should supplement the requirements listed in Table IWB-2600, "Components, Parts and Methods" of ASME Section XI, with a visual examination of all applicable reactor coolant pressure boundary pressure retaining threaded fasteners of closure seal assemblies with insulation removed in each applicable plant's ISI program. This examination should occur after initial fill and vent, and at normal operating pressure and temperature upon recovery for each refueling outage.

3. The use of sealant compounds and fastener lubricants should be under quality control to assure proper selection, procurement and application to minimize fastener susceptibility to SCC environments.

4. Licensees should develop maintenance procedures detailing the instructions for removal (detorque) and treatment (cleaning-handling) of fasteners, as well as detailed tensioning techniques during assembly and disassembly of closure seal systems.

5. Studs or bolts of manway closure assemblies should be cleaned, visually inspected, and magnetic particle or dye penetrant (for nonmagnetic material) examined each outage in which the closure seal is removed for equipment inspection/maintenance.

5.0 LONG TERM REGULATORY ACTIONS

The following summarizes the long term Regulatory actions in progress at this time:

(1) Work sponsored by Materials Engineering Branch, Division of Engineering - NUREG/CR-2467 entitled "Lower-Bound K_{ISCC} Values for Bolting Materials - A Literature Study" was issued in February 1982. It was based on the work performed by Lawrence Livermore National Laboratory (LLNL). NUREG/CR-2467 documented the available test data, in the form of K_{ISCC} versus yield strength, of various low alloy steels, maraging steels and stainless steel in water, aqueous chloride, aqueous sulfide and other environments. This report will be used jointly by the staff and NRC contractor, Brookhaven National Laboratory (BNL), to prepare an NRC position on the actions that should be taken to prevent stress corrosion cracking in threaded fasteners and fastener materials.

A contract pertaining to evaluating and establishing threaded fastener application requirements was recently placed (March 1982) with Brookhaven National Laboratory (BNL). The objective of this work is to obtain information leading to a regulatory position on material selection, installation, and inspection of threaded fasteners and threaded fastener material used in water reactors. The BNL work is expected to be completed by October of 1982. Based on that input, a NUREG report on this subject will be issued by the NRC. This report will serve as the principle vehicle for evaluating the safety significance of threaded fastener degradation and failures and for developing and implementing new and improved regulatory requirements applicable to threaded fasteners.

(2) Work sponsored by Chemical Engineering Branch, Division of Engineering - A draft report dated March 3, 1982, entitled "Boric Acid Corrosion of Ferritic Reactor Components," was completed by Brookhaven National Laboratory (BNL). This report summarized the material degradation experience resulting from boric acid corrosion of seven nuclear power plants (Fort Calhoun, Calvert Cliffs Units 1 and 2, Oconee Units 2 and 3, Kewaunee and Zion 1) based on review of applicable Licensing Event Reports (LER) and other relevant reports. This report also reviewed the available corrosion rate data of H_3BO_3, H_3BO_3-KOH and H_3-BO_3-LiOH solutions on various low alloy steels in the literature including some of BNL's own work and determined that a corrosion rate of at least 112 mils/year can be attained at 212^{o}F.

The BNL report will be used in licensing reviews and operating reactor licensing actions to establish the basis for analysis of effects of borated water corrosion on carbon and low alloy steel components including threaded fasteners.

(3) Prioritization by the Division of Safety Technology (DST) - DST has completed an assessment of the safety implications of stud corrosion in PWR systems (Reference 3). Their priority was based primarily on the borated water corrosion phenomenon.

DST estimated that the frequency of small-break (SZ) LOCAs that could result from corrosion of stud bolts would be about 6×10^{-3} per PWR-year, based on the operating experience to date. Based on the WASH-1400 distribution of release categories resulting from small-break LOCAs, the estimated risk is 1.5×10^{-4} Ci/PWR-year. If regular visual inspection of studs is required, the risk can be reduced by at least a factor of ten. Over the life of a plant, the additional cost of visual inspection is estimated to be approximately $110,000. Thus the priority score for this issue is 1 x 10^5 Ci/RY/10^6 dollars, which is a high score relative to other issues that have been prioritized.

6.0 <u>REFERENCES</u> (attached)

(1) Memorandum from R.F. Fraley, ACRS to W.J. Dircks, Subject: Bolt Failures In Nuclear Power Plants, dated October 20, 1981.

(2) Memorandum from W.J. Dircks, NRC, to R.F. Fraley, ACRS, Subject: Bolt Failures In Nuclear Power Plants, dated December 2, 1981.

(3) Memorandum from S.H. Hanauer, NRC, to D.G. Eisenhut, NRC, Subject: Corrosion of RCP Studs, dated April 6, 1982.

(4) IE Bulletin No. 82-02, dated June 2, 1982; Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants."

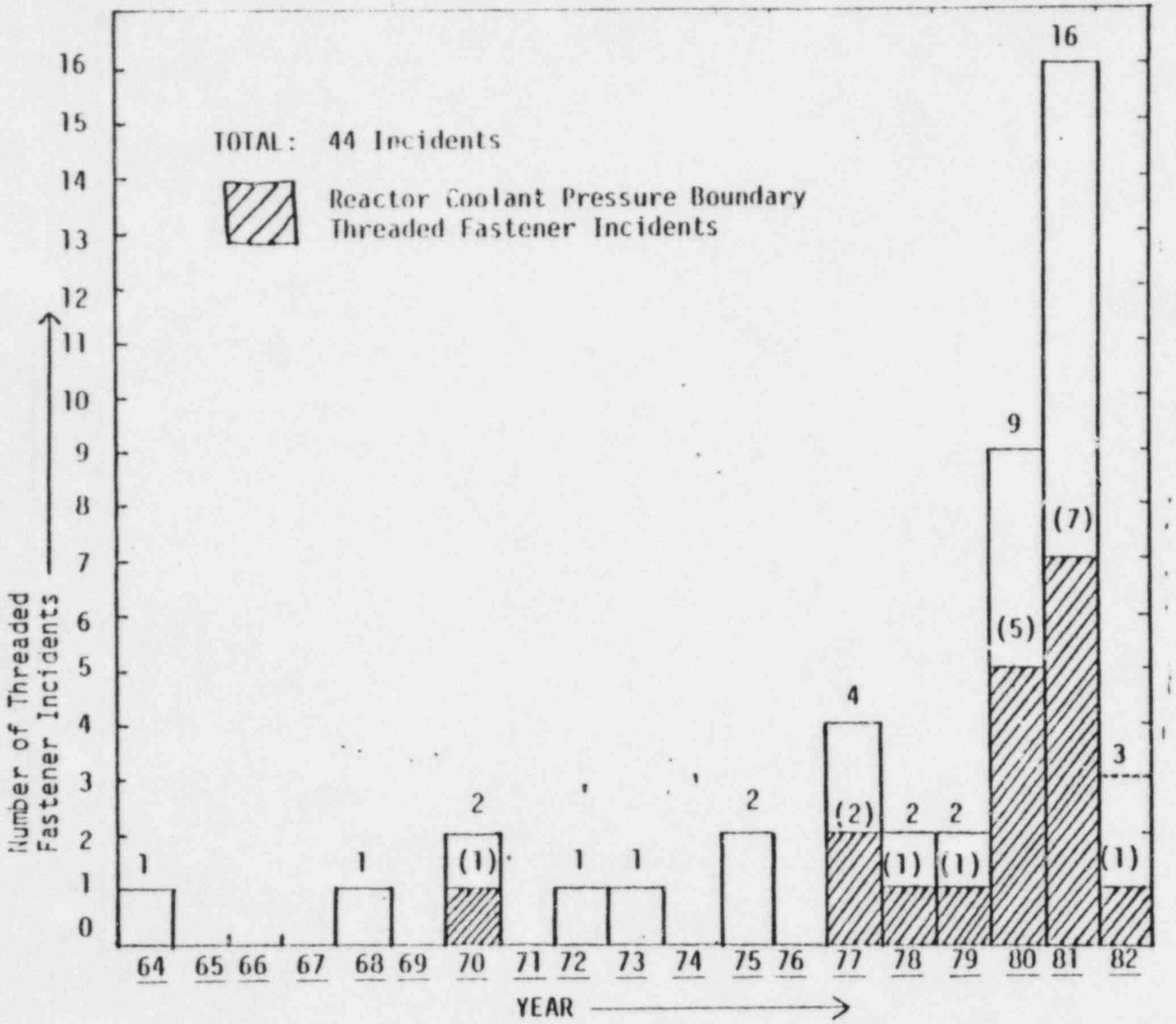


FIGURE 1 - A plot of the number of threaded fastener incidents covered in this report from 1964 to March 1982. The numbers in parentheses refer to reactor coolant pressure boundary threaded fastener incidents for that year.

TABLE 1. STRESS CORROSION OF THREADED FASTENERS

Plants	Year Reported	Components and Parts	Materials of Parts	Contributing Factors	Corrective Action
LaCrosse (BWR)	1970	Reactor Vessel Closure Studs (3.5 inch diameter)	12% Cr Martensitic Stainless Steel (ASTM-A-437-B4B)	(1) Aqueous environment during outage (2) Improper heat treatment of material (3) Galvanic action due to silver plating breakdown (4) Pretension	(1) Replaced with studs made from A-540-B23, Class 4 material (2) Augmented ISI UT surveillance
Ginna	1970	Steam Generator Support Anchor Studs (1-3/8 inch diameter)	Low alloy steel (AISI-4140)	(1) 160 KSI pretension (2) Humid/wet borated water	(1) Replaced with studs made from A-490 material (2) No pretension
Haddam Neck	1973	Steam Generator Support Anchor Bolts (2 inch diameter)	Low alloy steel	(1) Pretension (2) Water leakage	(1) 24 of 256 bolts replaced (2) Pretension reduced on replaced bolts (3) Installed microswitch on all bolts for monitoring
Surry 1	1975	Steam Generator Support Bolts	Maraging Steel (Vascomax 250)		(1) Replaced with Cd plated Vascoma: 250 bolts
Surry 2	1975	Steam Generator Support Bolts	Maraging Steel (Vascomax 250)		(1) Replaced with Cd plated Vascoma: 250 bolts

TABLE 1. STRESS CORROSION OF THREADED FASTENERS

Plants	Year Reported	Components and Parts	Materials of Parts	Contributing Factors	Corrective Action
San Onofre 1	1977	Steam Generator Manway Studs	Low alloy steel (AISI 4140)(A193-B7)		(1) 8 studs replaced
Midland 1	1979	Reactor Vessel Skirt Flange Imbed Anchor Studs (2-1/2 inches diameter)	Low alloy steel (AISI 4140, 4145)	(1) Improper heat treatment of material (2) Excessive Preload of 87-92 KSI	(1) Detension remaining studs to 6 KSI (2) Install upper lateral support on vessel
Prairie Island Unit 1	1980	Steam Generator Column Support Bolts (1-1/2 inch diameter)	Maraging Steel (Vascomax 250) (A538 grade B)	(1) Excessive Preload (1400 ft-lb torque)	(1) Replaced with studs made from same material (2) Pretension reduced
Prairie Island Unit 2	1980	Steam Generator Column Support Bolts (1-1/2 inch diameter)	Maraging Steel (Vascomax 250) (A538 grade B)	(1) Excessive Preload (1400 ft-lb torque)	(1) Replaced with studs made from same material (2) Pretension reduced
Rancho Seco	1980	Valve Studs	Stainless Steel Type 416 (A-193-B6)	Improper heat treatment of material	
Oconee 3	1980	Steam Generator Manway closure studs (2-inch diameter)	Low alloy steel (SA-320, Grade L-43)(AISI 4340)	(1) Use of thread lubricant containing molybdenum disulfide (2) Trapped moisture	(1) All studs replaced (thread lubricant containing molybdenum disulfide was applied)

TABLE 1. STRESS CORROSION OF THREADED FASTENERS

lants	Year Reported	Components and Parts	Materials of Parts	Contributing Factors	Corrective Action
rkansas 1	1980	Steam Generator Manway Closure Studs	Low alloy steel (AISI 4340)	(1) Use of thread lubricant containing molybdenum disulfide (2) Preload	(1) 3 cracked studs replaced
	1978	Steam Generator Manway Closure Studs	Low alloy steel (AISI 4340)		(1) 2 cracked studs replaced
.C. Cook 1	1981	Main steam isolation valve internals - studs	Low alloy steel (AISI 4340)	(1) Primary steam (2) Possible use of thread lubricant containing molybdenum disulfide (3) Possible over torque	
aio Verde	1981	Piping Restraint Embed Anchor Bolts (1-1/2 inch diameter)	Low alloy steel (AISI 4140)(A-354 Grade BD)	(1) Improper heat treatment of material	
conee 1	1981	Reactor Vessel Internals - Thermal Shield Bolts	A-286 stainless steel	(1) Borated water environment (2) Preload of 32 KSI and 32 KSI bending	(1) Redesigned lower thermal shield (2) Inconel X-750 studs and nuts

TABLE 1. STRESS CORROSION OF THREADED FASTENERS

ants	Year Reported	Components and Parts	Materials of Parts	Contributing Factors	Corrective Action
onee 2	1981	Reactor Vessel Internals - Thermal Shield Bolts	A-286 stainless steel	(1) Borated water environment (2) Preload of 32 KSI and 32 KSI bending	(1) Redesigned lower thermal shield (2) Inconel X-750 studs and nuts
aine Yankee	1982	Steam Generator Manway Closure Studs (1-1/2 inch diameter)	Low alloy steel (SA540-B24)	(1) Gasket leakage of borated water (2) Use of Furmanite sealing compound (3) Use of thread lubricant containing molybdenum disulfide (4) Preload of 900-1100 ft-lb	(1) 10 failed studs replaced with same stock
	1982	6-inch gate valve bonnet to body studs (5/8-inch diameter)	Stainless steel	Valve body to bonnet gasket leakage of borated water	(1) Proposed short-term action - replace with AISI 4140 (A-196-B7) studs (2) Proposed long-term action - use 17-4 PH studs

TABLE 2. FATIGUE OF THREADED FASTENERS

Plants	Year Reported	Components and Parts	Materials of Parts	Contributing Factors	Corrective Action
Palisades	1972	Reactor Vessel Internals - hold down bolts for Ring Shim (1/2 inch diameter)	Type 304 stainless steel	Improper torque	(1) Broken bolts replaced (2) Use proper torque and clearance
Yankee Rowe	1968	Reactor Vessel Internals - thermal shield bolts	Type 316 stainless steel	Flow Induced Vibration	(1) Added clamp to each thermal shield joint
Big Rock Point (BWR)	1964	Reactor Vessel Internals - thermal shield bolts	Type 316 Stainless Steel (ASTM A-276)	Flow induced vibration	(1) Modify support and flow pattern

TABLE 3. BORATED WATER CORROSION OF THREADED FASTENERS

Plants	Year Reported	Components and Parts	Materials of Parts	Contributing Factors	Corrective Action
St. Lucie	1978	Pressurizer manway closure studs	Low carbon low alloy steel (SA-540-B24)	Manway leakage of borated water	(1) 5 corroded studs were replaced
	1977	Steam generator manway closure studs (1-1/2 inch diameter)	Low carbon low alloy steel (SA-540-B24)	Manway gasket leakage of borated water	(1) 3 studs replaced (2) Gasket replaced
Fort Calhoun	1980	Reactor coolant pump closure studs (3-1/2 inch diameter)	Low alloy steel (AISI 4140)(SA-193-B7)	Flexitallic flange gasket leakage	(1) 9 studs replaced
	1981	Reactor coolant pump closure studs (3-1/2 inch diameter)	Low alloy steel (AISI 4140)(SA-193-B7)		(1) Corroded studs replaced
Calvert Cliffs Unit 1	1980	Reactor coolant pump closure studs	Low alloy steel	Possible gasket leakage of borated water	(1) 27 studs replaced
	1980	Steam generator manway studs	Low alloy steel	Gasket leakage of borated water	(1) 11 studs replaced
Calvert Cliffs Unit 2	1981	Reactor coolant pump closure studs	Low alloy steel	Possible gasket leakage of borated water	(1) 12 studs replaced
	1981	Pressurizer manway studs	Low alloy steel	Seal leakage of borated water	(1) 2 studs replaced
	1981	Safety injection check valve studs	Low alloy steel	Gasket leakage of borated water	(1) 16 studs replaced

TABLE 3. BORATED WATER CORROSION OF THREADED FASTENERS

Plants	Year Reported	Components and Parts	Materials of Parts	Contributing Factors	Corrective Action
Oconee 3	1981	Reactor coolant pump closure studs	Low alloy steel	Closure gasket leakage of borated water	(1) 1 stud replaced
Oconee 2	1981	Reactor coolant pump closure studs	Low alloy steel	Closure gasket leakage of borated water	(1) 1 stud replaced
Arkansas 1	1981	Steam generator manway closure studs	Low alloy steel	Closure gasket leakage of borated water	(1) Corroded studs replaced
Kewaunee	1981	8-inch motor operated valve-body to bonnet studs	Low alloy steel	Valve body to bonnet gasket leakage of concentrated (12%) borated water	(1) Corroded studs replaced
D.C. Cook 2	1981	Check valve bonnet bolts	Low alloy steel (AISI 4140)(A-193-87)	Valve body to bonnet gasket leakage of borated water	(1) All 12 studs replaced

TABLE 4. EROSION-CORROSION OF THREADED FASTENERS

'lants	Year Reported	Components and Parts	Materials of Parts	Contributing Factors	Corrective Action
ion 1	1979	Chemical & volume control system valve bolts	Low alloy steel (AISI 4140) (A193-B7)	Valve gasket leakage of borated water	(1) Degraded bolts replaced (2) Valve bonnet reassembled

TABLE 5. OTHER DEGRADATION OF THREADED FASTENERS

Plants	Year Reported	Components and Parts	Materials of Parts	Contributing Factors	Corrective Action
Pilgrim 1 (BWR)	1981	Valve limit-torque operator motor hold down bolts			Replace bolts
Surry 2	1981	Service water pump impeller capscrew	Carbon steel		(1) Broken capscrew replaced (2) All impeller capscrews to be replaced with stainless steel capscrews

TABLE 6. SUMMARY OF DEGRADED THREADED FASTENERS IN REACTOR COOLANT PRESSURE BOUNDARY

Degraded Reactor Coolant Pressure Boundary Threaded Fasteners	No. of Reported Incidents	Plants (Year Incident Reported & Reactor Vendor)	Mode of Failure
Reactor vessel closure studs	1	LaCrosse (1970) AC	Stress Corrosion (SC)
Pressurizer manway closure studs	2	Calvert Cliffs 2 (1981) CE St. Lucie 1 (1978) CE	Borated Water Corrosion (BC)
Steam generator manway closure studs	8	Maine Yankee (1982) CE Oconee 3 (1980) B&W Arkansas 1 (1981) B&W Arkansas 1 (1978)(1980) B&W Calvert Cliffs 1 (1980) CE St. Lucie 1 (1977) CE San Onofre 1 (1977) W	SC SC BC SC BC BC SC
Reactor coolant pump closure studs	5	Ft. Calhoun (1980) CE Calvert Cliffs 1 (1980) CE Calvert Cliffs 2 (1981) CE Oconee 3 (1981) B&W Oconee 2 (1981) B&W	BC BC BC BC BC
Safety injection check valve studs	1	Calvert Cliffs 2 (1981) CE	BC

TABLE 6. SUMMARY OF DEGRADED THREADED FASTENERS IN REACTOR COOLANT PRESSURE BOUNDARY

Degraded Reactor Coolant Pressure Boundary Threaded Fasteners	No. of Reported Incidents	Plants (Year Incident Reported & Reactor Vendor)	Mode of Failure
CVCS isolation valve bolts	1	Zion 1 (1979) W	Erosion-Corrosion
Check valve studs	1	D.C. Cook 2 (1981) W	BC

TOTAL NO. OF PLANTS: 13
TOTAL NO. OF EVENTS: 19

TABLE 7. SUMMARY OF DEGRADED THREADED FASTENERS IN SUPPORTS

Degraded Threaded Fasteners In Component Supports	No. of Plants	Plant (year Incident Reported)
Steam generator support bolts	6	Surry 1 (1974) Surry 2 (1974) Sequoyah 1 (1977) Sequoyah 2 (1977) Prairie Island 1 (1980) Prairie Island 2 (1980)
Steam generator support embed anchor studs	2	Ginna (1970) Haddam Neck (1973)
Reactor coolant pump support bolts	1	Waterford (1981)
Reactor vessel embed anchor studs	1	Midland (1979)
Piping restraint embed anchor bolts	1	Palo Verde (1981)

TOTAL NO. OF PLANTS: 11

TABLE 8. SUMMARY OF DEGRADED THREADED FASTENERS IN INTERNALS

Degraded Threaded Fasteners In Component Internals	No. of Plants	Plants (year incidents reported)
Reactor vessel internals - thermal shield bolts	4	Oconee 1 (1981) Oconee 2 (1982) Yankee Rowe (1968) Big Rock Point (1968)
Reactor vessel internals - Hold down bolts for ring shim	1	Palisades (1972)
Main steam isolation valve internals-studs	1	D.C. Cook 1 (1981)
Service water pump internals-impeller capscrew	1	Surry 2 (1981)

TOTAL NO. OF PLANTS: 7

TABLE 5. OTHER DEGRADATION OF THREADED FASTENERS

Plants	Year Reported	Components and Parts	Materials of Parts	Contributing Factors	Corrective Action
Sequoyah 1	1977	Steam generator support bolts (1-1/2 inch diameter)		(1) Quench cracks	Replace bolts
Sequoyah 2	1977	Steam generator support bolts (1-1/2 inch diameter)		(1) Quench cracks	Replace bolts
Arkansas 1	1980	Emergency feedwater turbine steam inlet bolts	Carbon steel (C-1117)	(1) Wrong material (2) Water hammer	(1) All replaced with low alloy steel (AISI 4140) bolts
Waterford	1981	Reactor coolant pump support bolts	A-490	(1) Improper torque (2) Some bolts too short	(1) Replaced failed bolts and short bolts (2) Retorque bolts with calibrated torque equipment (3) Improve QA plan for bolting
Vermont Yankee	1981	Valve limit-torque operator motor mounting bolts			(1) 4 mounting bolts replaced

BNL-NUREG -31743
INFORMAL REPORT
LIMITED DISTRIBUTION

EVALUATION OF FAILED REACTOR COOLANT PUMP INTERNAL BOLTS FROM THE H. B. ROBINSON NUCLEAR POWER STATION

CARL J. CZAJKOWSKI

DATE PUBLISHED - OCTOBER 1982

CORROSION SCIENCE GROUP
DEPARTMENT OF NUCLEAR ENERGY BROOKHAVEN NATIONAL LABORATORY
UPTON, NEW YORK 11973

16pp

Prepared for the U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Contract No. DE-AC02-76CH00016

~~8301110160~~

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Nuclear Regulatory Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

BNL-NUREG-31743
INFORMAL REPORT
LIMITED DISTRIBUTION

EVALUATION OF FAILED REACTOR COOLANT PUMP INTERNAL BOLTS FROM THE H. B. ROBINSON NUCLEAR POWER STATION*

Carl Czajkowski

August 1982

Brookhaven National Laboratory
Corrosion Science Group
Department of Nuclear Energy
Upton, New York 11973

*Research carried out under the auspices of the U. S. Nuclear Regulatory Commission.

TABLE OF CONTENTS

LIST OF FIGURES

Abstract

A metallographic failure analysis was performed on two cracked bolts on the diffuser adapter to casing adapter plate from Reactor Coolant Pumps of the H. B. Robinson Unit 2 nuclear power plant. The observations included transgranular cracking and pits on the shank of one bolt and the head of another. SEM evaluation of the fracture surface disclosed areas of probable fatigue interaction. The report concludes that the bolt failure was due to a stress corrosion cracking (SCC) mechanism caused by probable chloride contamination and fatigue enhancement.

1.0 INTRODUCTION

In April 1982, during the 10 year inservice inspection of reactor coolant pumps at the H. B. Robinson plant, the utility discovered that various diffuser adaptor to casing adaptor bolts had failed in service. In May 1982, a bolt which had cracked in the thread area was sent to Brookhaven National Laboratory (BNL) for examination(1). The report concluded that the bolt failed by a stress corrosion cracking mechanism probably caused by high tensile stresses in the bolt and probable chloride contamination.

In June 1982, three additional bolts were sent to BNL from the H. B. Robinson plant for examination. The work was done by a subcontract issued to BNL from the Franklin Institute under P.O. #C-67718. This report describes the results of this examination.

The analysis of the bolts was to be a metallurgical investigation of the failure mechanism and encompassed the following methods of evaluation;

a) Visual/Photography

b) Scanning Electron Microscopy (SEM) and Energy Dispersive Spectroscopy (EDS)

c) Optical Microscopy/Metallography.

2.0 VISUAL/PHOTOGRAPHY

The first bolt examined (Figure 1) had no visual cracks and was later determined by discussion with A. Herth (USNRC-Region 2) to be an archive specimen if additional testing was needed. It was identified by the packing designation as Pump: Capscrew 21. The bolt was 4 inches long and was .0625 inch in diameter.

The second specimen received (Figure 2) was just the shank portion of a bolt with a fracture on one end and the appearance of mechanical severing on the opposing end. This specimen was identified as PUMP CAPSCREW "B".

The third bag examined had two specimens inside. The first specimen (Figure 3) was a portion of the threaded area of a bolt with the same appearance of mechanical severing on one end. The second specimen was a bolt head with a thin metallic sheath surrounding it (Figure 4). This bolt head had fractured at the bolt head/shank juncture. This bag was identified as PUMP: CAPSCREW 13/14.

The outside of the three plastic bags were surveyed by BNL Health Physics personnel with the following results: (at contact)

Specimen I.D.	21	"B"?	13/14
Beta-Gamma	600 mR/hr	1.1 R/hr	2 R/hr
Gamma	200 mR/hr	250 mR/hr	800 mR/hr

3.0 SEM/EDS

The fracture face of the specimen identified as PUMP CAPSCREW "B"? was cut from the bolt shank and viewed in the SEM. Figure 5 is a low magnification photograph of the fracture face. The fracture was relatively flat with only one area of ridging and was transgranular in appearance. One area of pitting was noticed during the examination (Figure 6) with the pit lying on the plane of the fracture surface. The ridge area was examined (Figure 7) and had the appearance of some intergranularity present on the shank side of the fracture face. The examination also disclosed one area of possible fatigue contribution (Figure 8) which was seen in an area almost diametrically opposite the fracture ridge.

The fracture face of the bolt head (identified as PUMP CAPSCREW 13/14) was also examined by SEM. It also had a very flat appearing fracture surface (Figure 9) and was generally transgranular in nature (Figures 10, 11 and 12). No pitting was evident in the evaluation of the fracture surface. Figure 13 is an SEM photograph of the apparent area of fracture initiation which had definite fatigue wavefront characteristics evident.

Since decontamination had been performed on these bolts prior to SEM evaluation, EDS examination for contaminants was omitted. An analytical SEM technique for stress corrosion cracking environment determination(2) was performed on the fracture surfaces of both bolts examined. This technique produced data generated by examination of Type 316 stainless steel fracture surfaces. The technique indicates whether the stress corrosion cracking occurred by environmental interaction of the stainless steel with either chloride or hydroxide solutions. The examination entails the measurement of Cr/Fe ratios after EDS analysis. Classification of these ratios is then based on the observations that in chloride solutions, the corrosion product is chromium rich (Cr/Fe 0.6-0.9) while in hydroxide solutions, the corrosion produce is iron rich (Cr/Fe 0.4-0.6).

Figure 14 is an EDS scan of the base material at 35 kV. The Cr/Fe/Ni ratios are in reasonable agreement with those of a Type 304 austenitic stainless steel. (Note: the use of a higher kV accelerating voltage provides a more accurate determination of the materials intensive properties.)

Figures 15, 16, 17, 18, and 19 are EDS scans across the fracture face of PUMP: CAPSCREW 13/14 (the bolt head fracture). The Cr/Fe ratio for these scans ranged from a low value of .957 to a high value of 2.10. Figures 20, 21, 22, 23, and 24 are EDS scans across the fracture face of PUMP CAPSCREW B7. This fracture face had Cr/Fe ratios ranging from 0.839 to a maximum of 2.027.

The ratios observed on these two specimens are significantly higher than those previously reported(1). These greater values in Cr/Fe ratios are probably due to the fracture faces having a more prolonged contact with the corrodent solution, than the previously examined bolt which cracked in the thread area.

These high Cr/Fe ratios indicate the possibility of chloride induced stress corrosion cracking as the mode of failure.

4.0 OPTICAL MICROSCOPY/METALLOGRAPHY

Additional characterization of the cracking phenomenon was accomplished by cutting cross sections from the two fractured bolts and examining them metallographically. These samples were made by cutting transversely to the fracture face approximately mid-cross section of the bolt. The first section examined (Figure 25) was from the shank fracture area after ASTM A-262 was performed. It can be seen in the photomicrograph that the structure is that commonly seen in austenitic stainless steel with no evidence of sensitization. There was virtually no evidence of significant cold work on the specimen.

The second specimen examined (Figure 26) was the area adjacent to the bolt head failure fracture face. The structure of the material after ASTM A-262 etching also shows no evidence of sensitization. This specimen showed some areas of significant cold work (Figures 27, 28, and 29). Various areas of pitting were also found on the bolt head cross section (Figures 30, 31, and 32) which is indicative of chloride contamination.

5.0 DISCUSSION/CONCLUSIONS

It has been observed that chlorides will cause pitting as well as transgranular cracking in austenitic stainless steels(3,4) at higher temperatures. Since these bolts were from the same pumps which supplied the first bolt for examination(1); the conclusion that chloride contamination is contributing to the cracking is consistent.

The following conclusions can then be drawn from the failure analysis observations:

1. The bolting material is of the austenitic stainless variety (probably Type 304).

2. The bolting material is in the non-sensitized condition.

3. The transgranular characteristics of the cracking, with pitting in evidence, indicates chloride contamination. This is also substantiated by the very high Cr/Fe ratios encountered on the fracture surfaces.

4. Since evidence of fatigue was observed on both fracture faces; it is reasonable to assume that fatigue interaction contributed to the bolt failures by a fatigue assisted stress corrosion cracking mechanism caused by probable chloride contamination.

6.0 ACKNOWLEDGEMENTS

The author wishes to thank the Franklin Institute for funding the analysis. In addition thanks are due L. Gerlach, R. Sabatini, P. Knisely, and especially Dr. J. R. Weeks for his constant support.

7.0 REFERENCES

1. Czajkowski, C. J., BNL-31530, July 1982.

2. Poulson, B., Gartside, H., Br. Corr. J., 1978, Vol. 13, No. 2.

3. Anderson, P. A., Duquette, D. J., Corrosion, Vol. 36, No. 8, August 1980.

4. Leckie, H. P., Uhlig, H. H., Jour. of the Electrochem. Soc., Vol. 173, No. 12, December 1966.

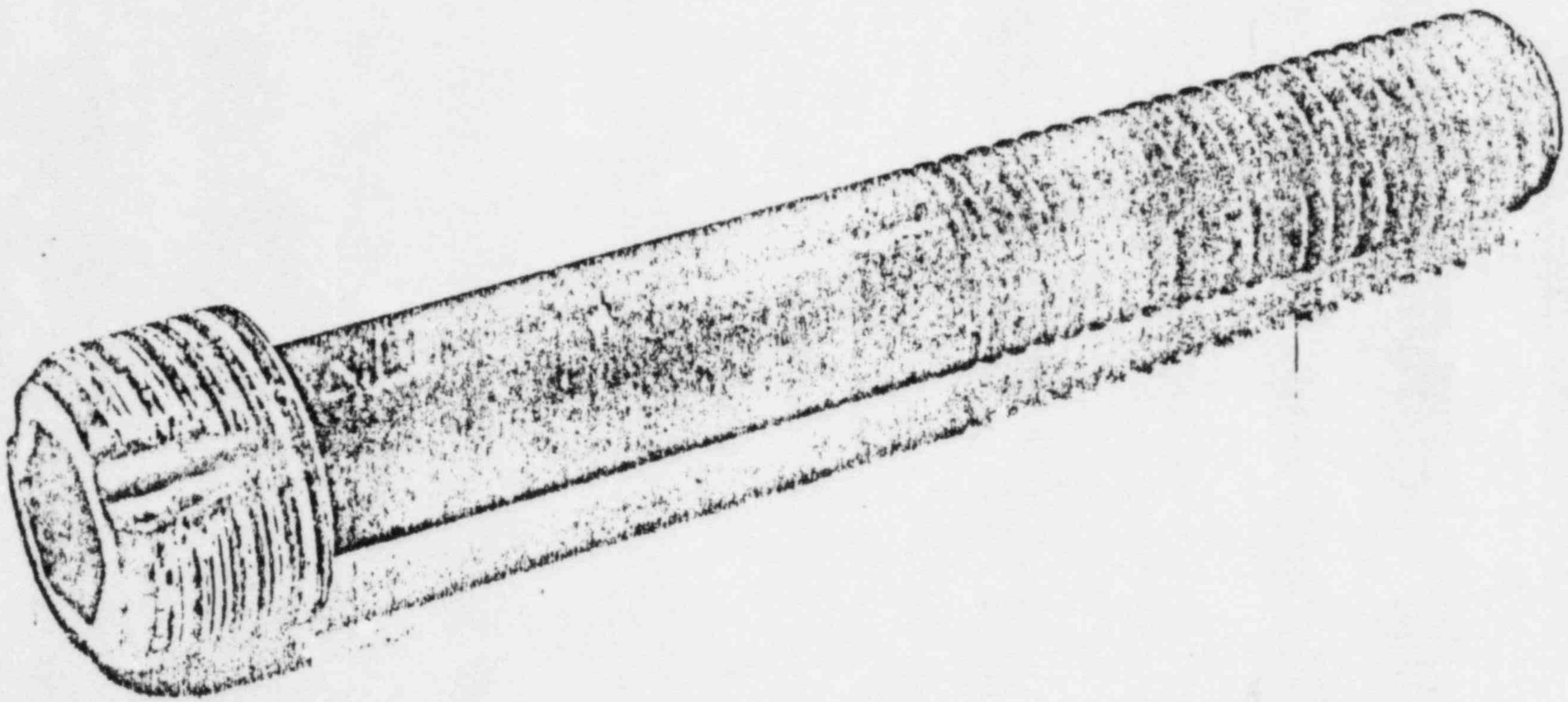


Figure 1. Photograph of archive bolt from H.B.Robinson.
(identified as PUMP:CAPSCREW 21)

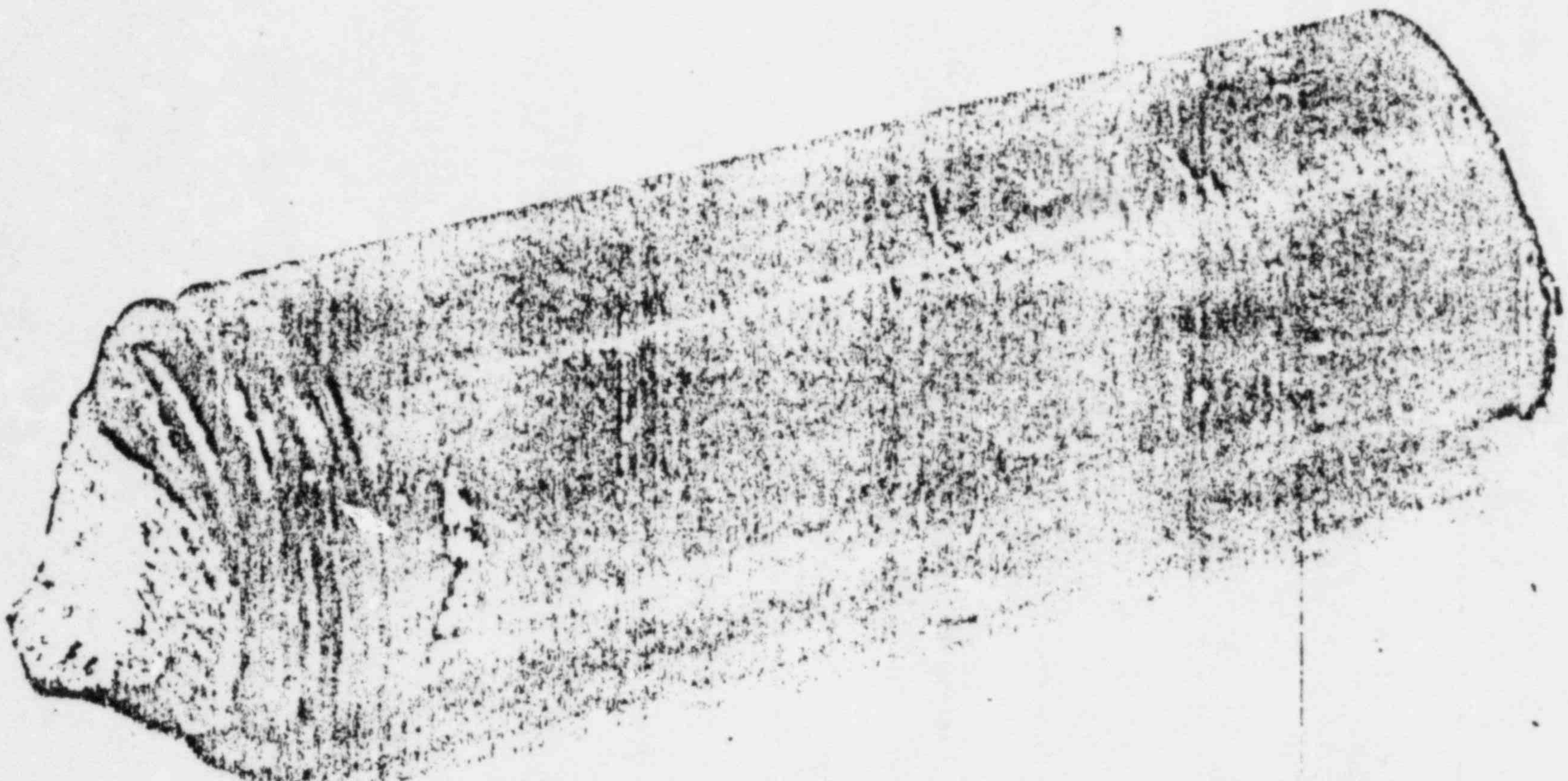


Figure 2. Photograph of shank portion of a bolt from H.B.Robinson.
(identified as PUMP:CAPSCREW " B "?)

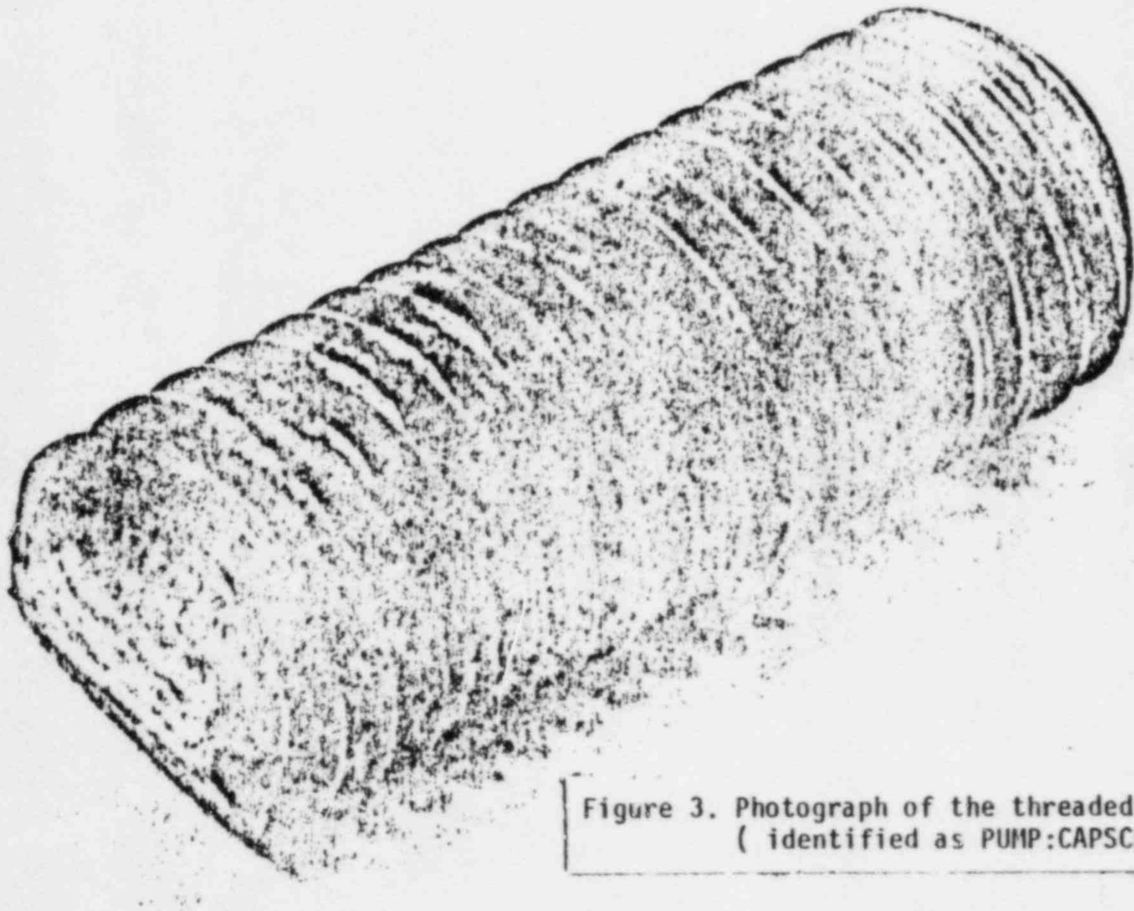


Figure 3. Photograph of the threaded portion of a bolt.
(identified as PUMP:CAPSCREW 13/14)

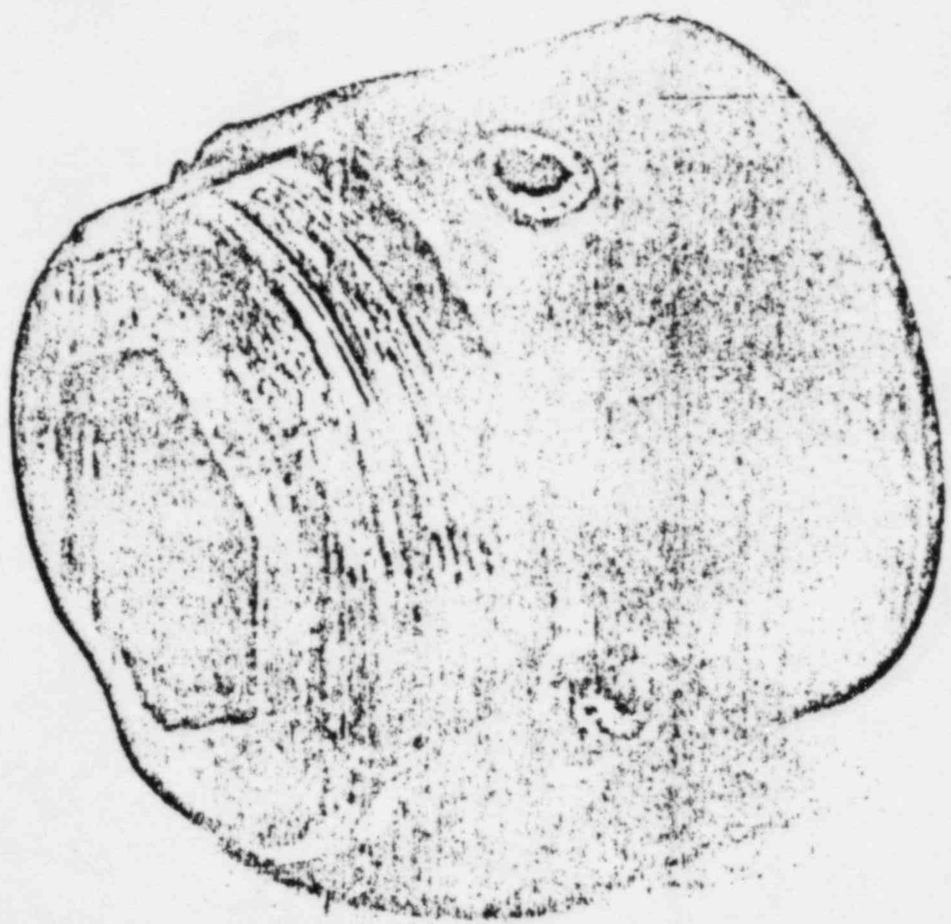
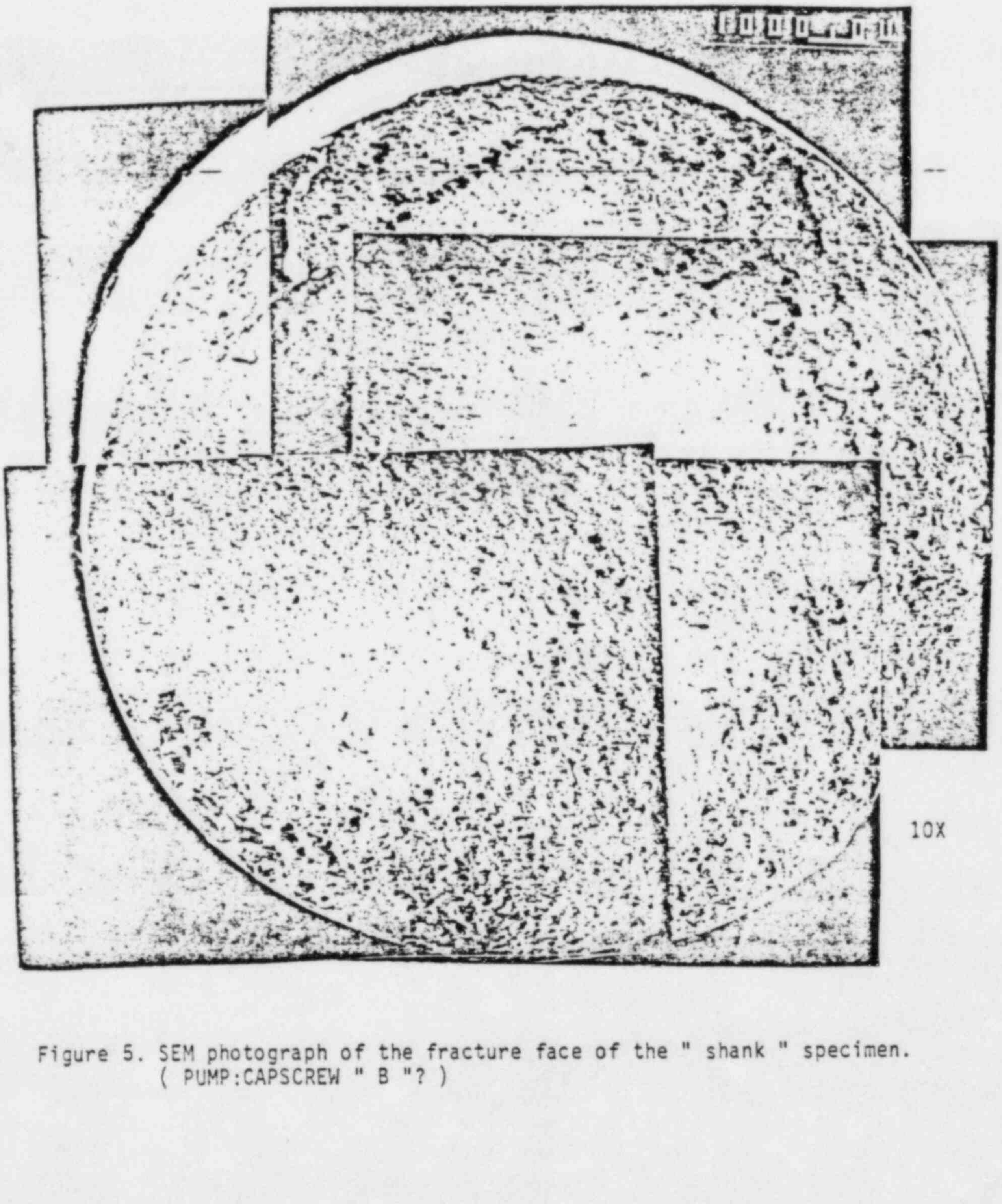
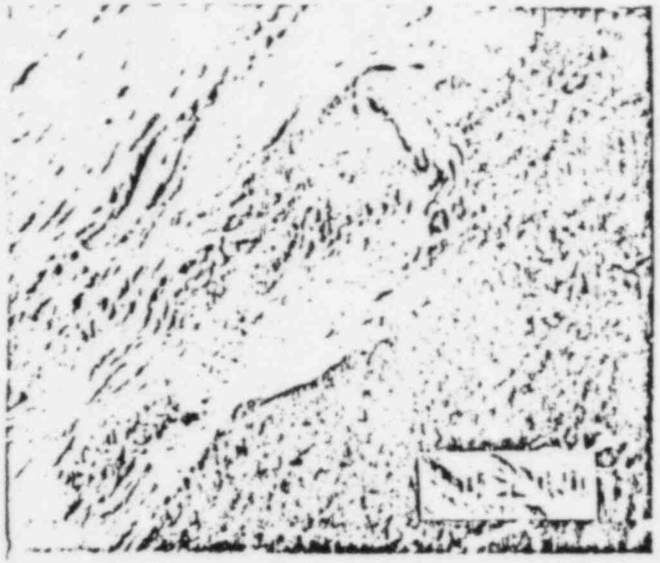


Figure 4. Photograph of the bolt head from H.B.Robinson.
(identified as PUMP:CAPSCREW 13/14)



10X

Figure 5. SEM photograph of the fracture face of the " shank " specimen.
(PUMP:CAPSCREW " B "?)



1000X

Figure 6. SEM photo of a pit on the fracture face of PUMP:CAPSCREW " B "?.

100X

Figure 7. Probable initiation site of the fracture on PUMP:CAPSCREW " B "?.

2000X

Figure 8. Possible area of fatigue striation on PUMP:CAPSCREW " B "?.

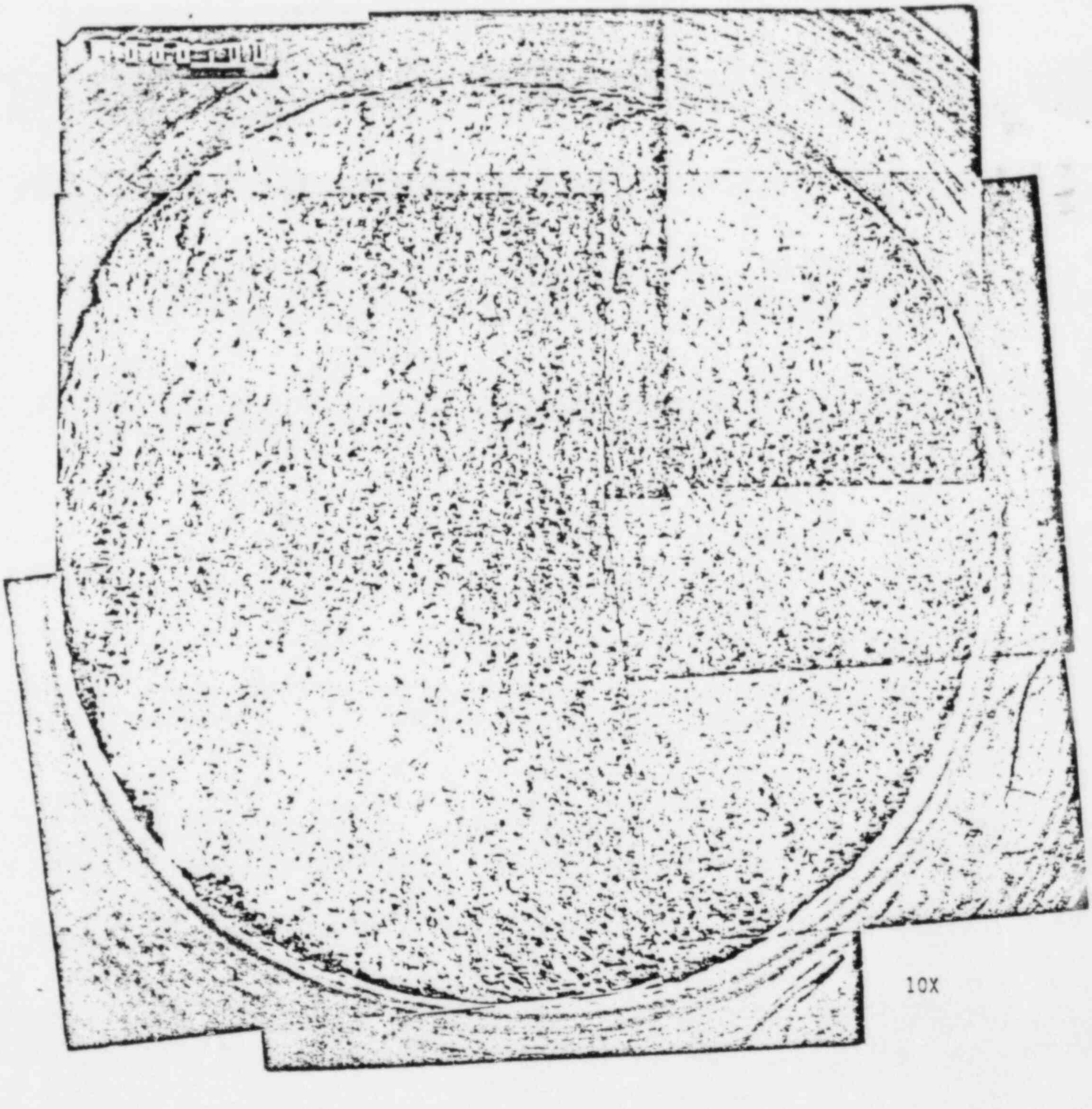
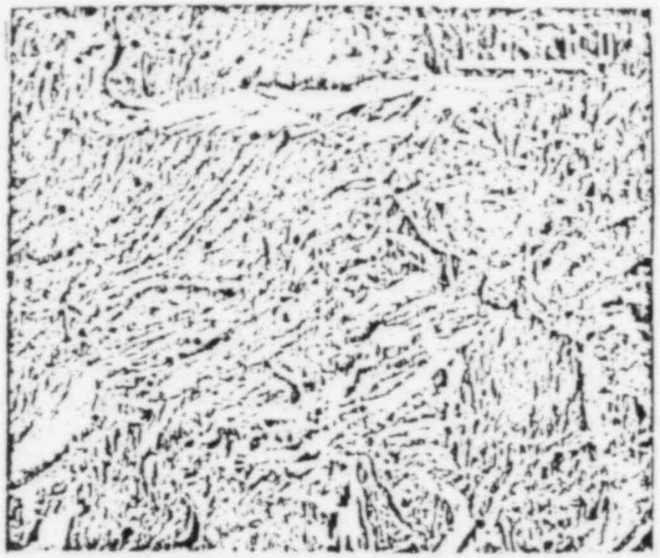


Figure 9. SEM photograph of the fracture face of the bolt head specimen.
(PUMP:CAPSCREW 13/14)

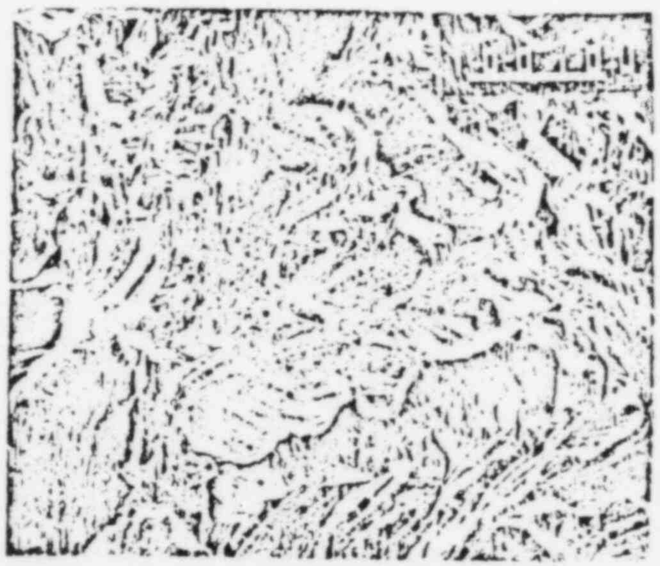


200X

Figure 10. SEM photo of a transgranular area of the fracture face on PUMP:CAPSCREW 13/14.

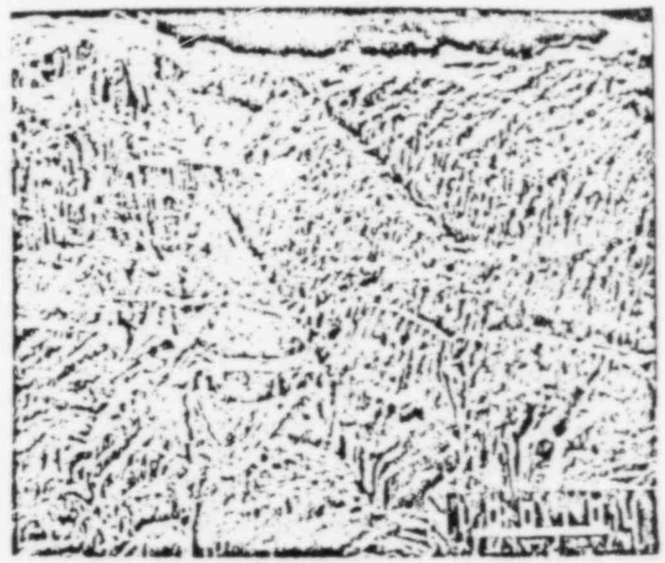
200X

Figure 11. SEM photo depicting the transgranularity near the edge of the fracture face.



200X

Figure 12. SEM photo showing additional areas of transgranularity.



200X

Figure 13. SEM photo of fatigue striations on PUMP:CAPSCREW 13/14.

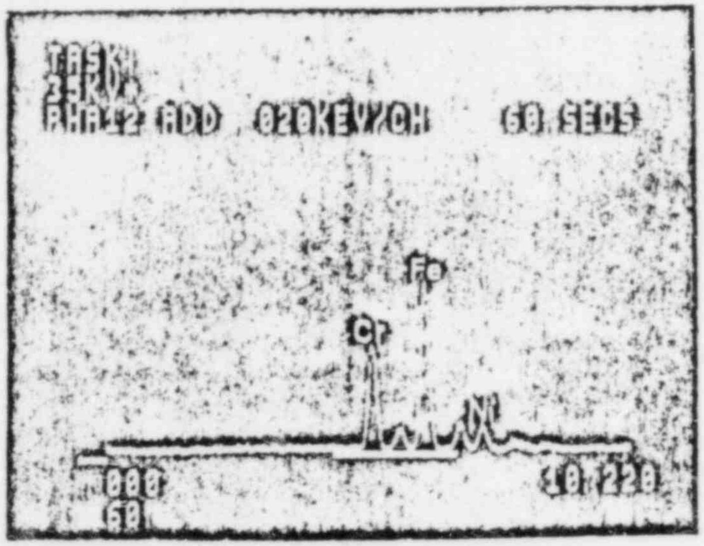


Figure 14. EDS scan of base metal for Cr/Fe ratio (Cr/Fe - .638)

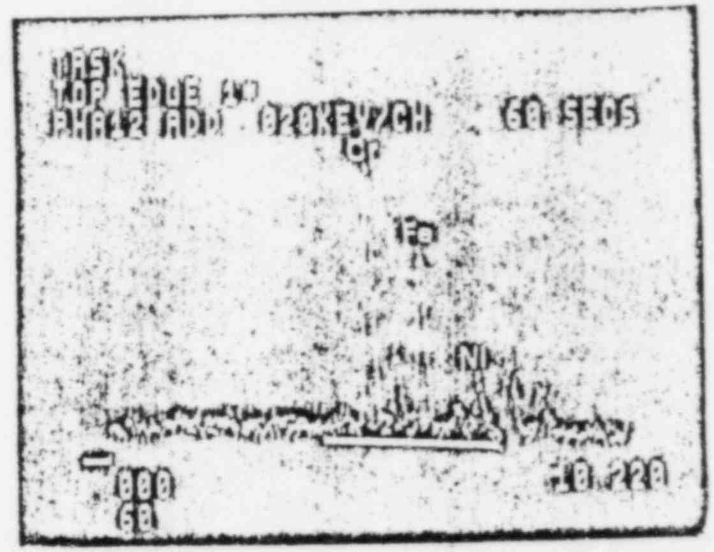


Figure 15. EDS scan of fracture face for Cr/Fe ratio (Cr/Fe - 1.426)

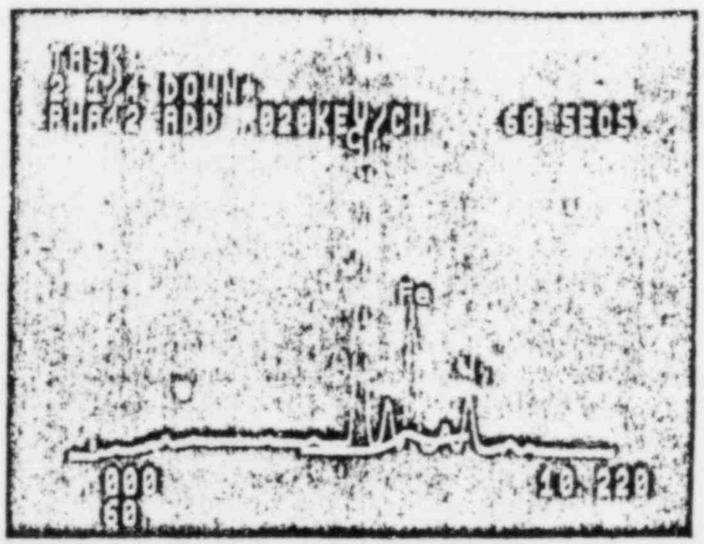


Figure 16. EDS scan of fracture face for Cr/Fe ratio (Cr/Fe - 2.10)

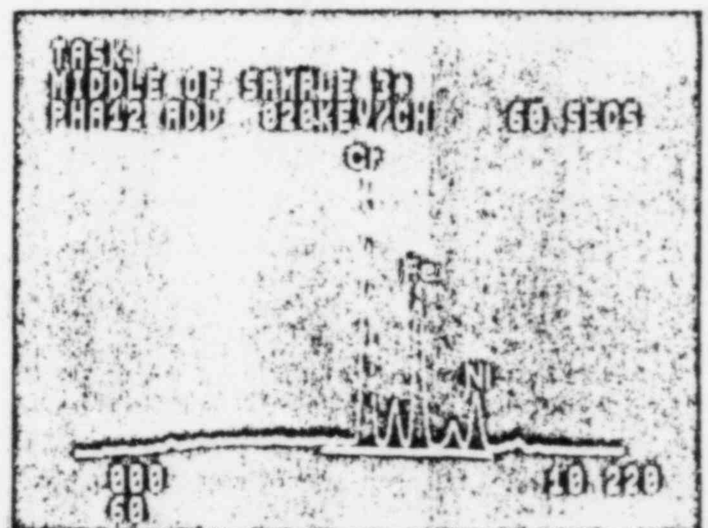


Figure 17. EDS scan of fracture face for Cr/Fe ratio (Cr/Fe - 1.61)

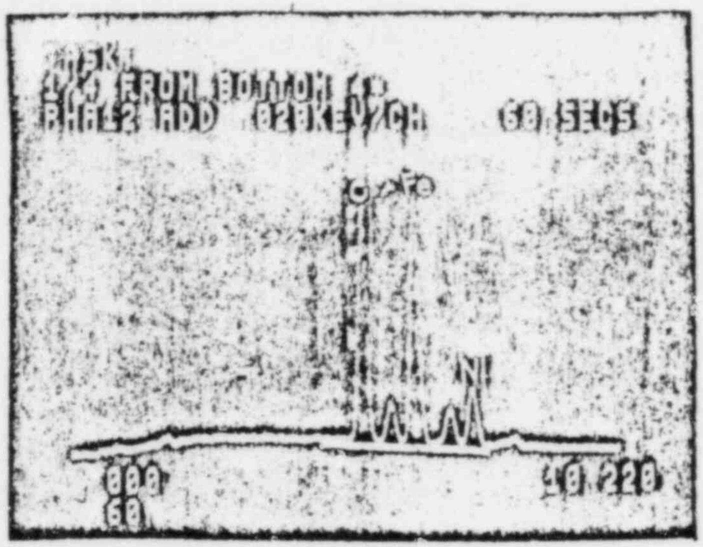


Figure 18. EDS scan of fracture face for Cr/Fe ratio (Cr/Fe - .957)

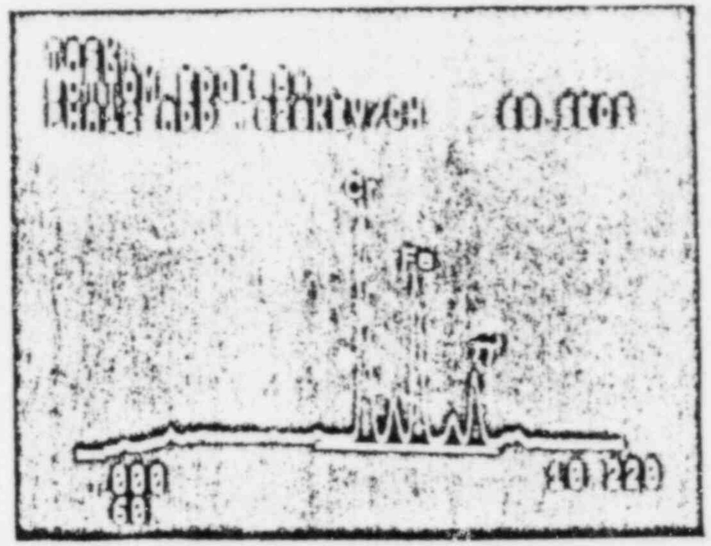


Figure 19. EDS scan of fracture face for Cr/Fe ratio (Cr/Fe - 1.417)

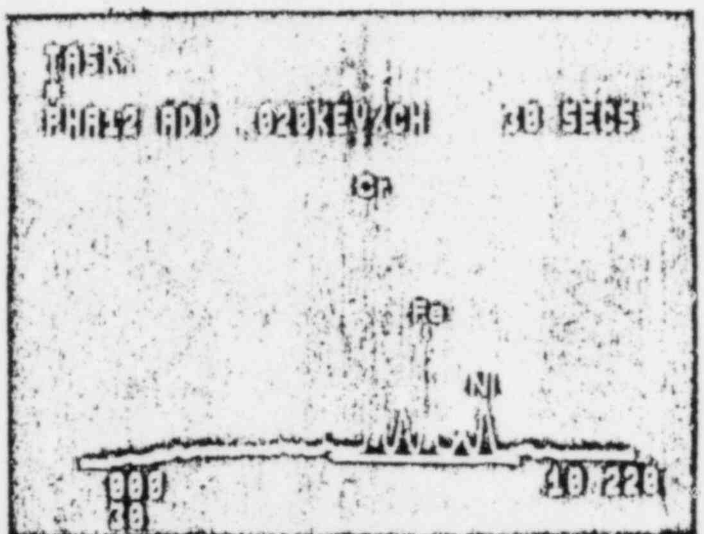


Figure 20. EDS scan of shank fracture face for Cr/Fe ratio (Cr/Fe - 2.0)

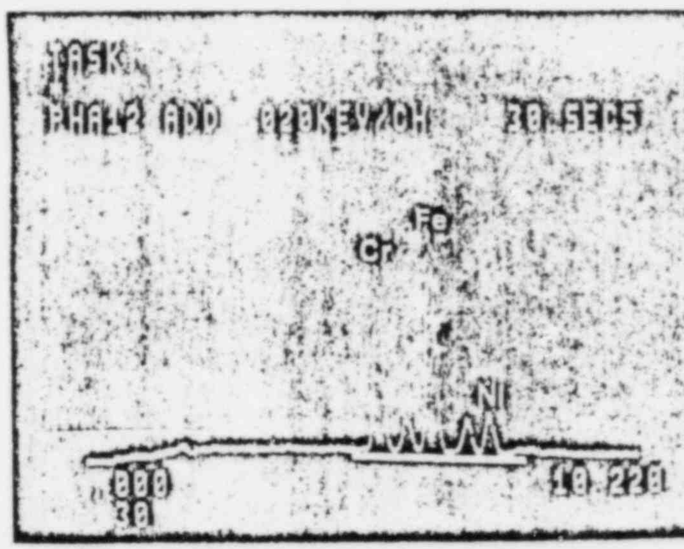


Figure 21. EDS scan of shank fracture face for Cr/Fe ratio (Cr/Fe - .839)

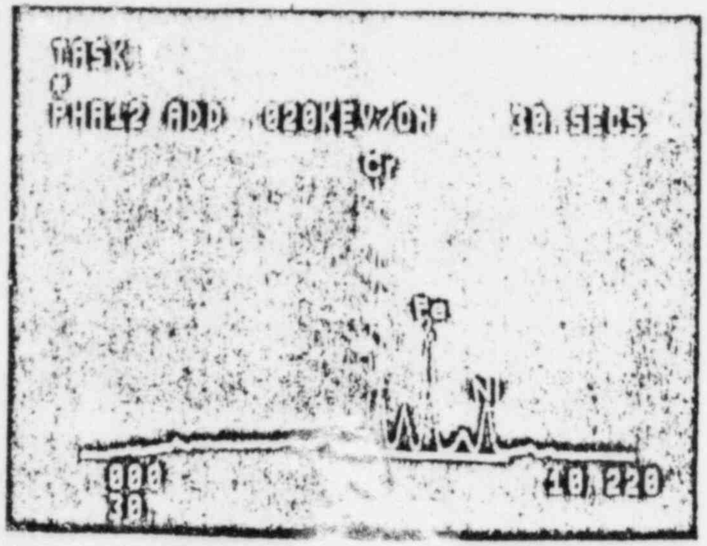


Figure 22. EDS scan of shank fracture face for Cr/Fe ratio (Cr/Fe - 2.027)

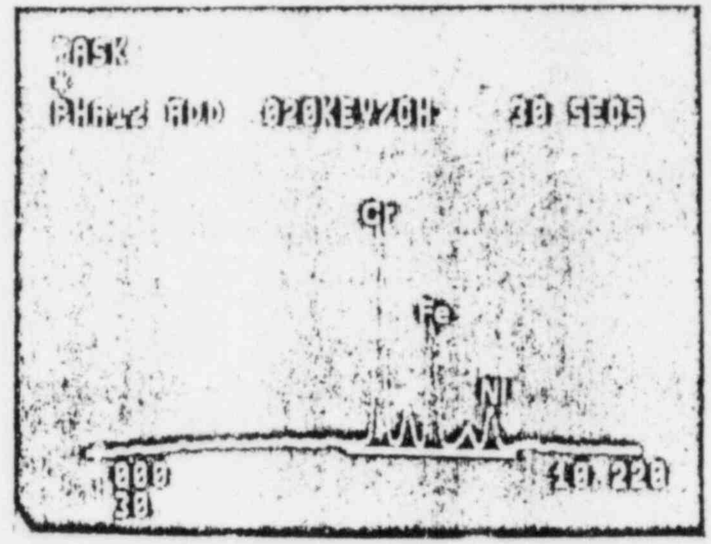


Figure 23. EDS scan of shank fracture face for Cr/Fe ratio (Cr/Fe - 2.025)

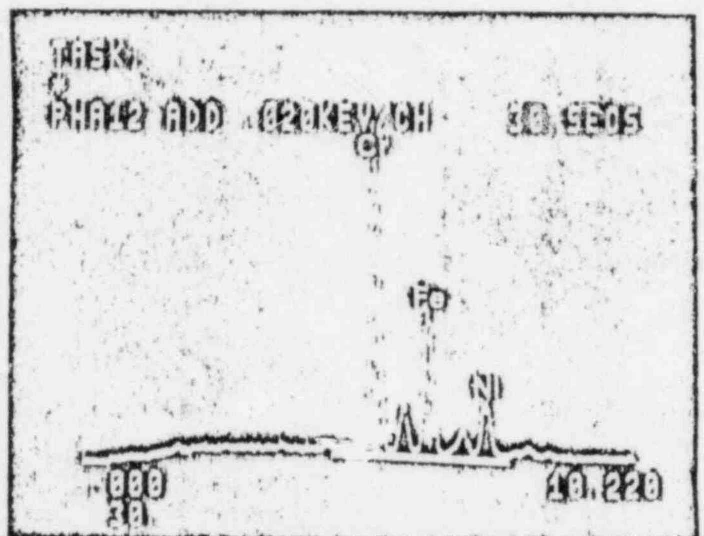
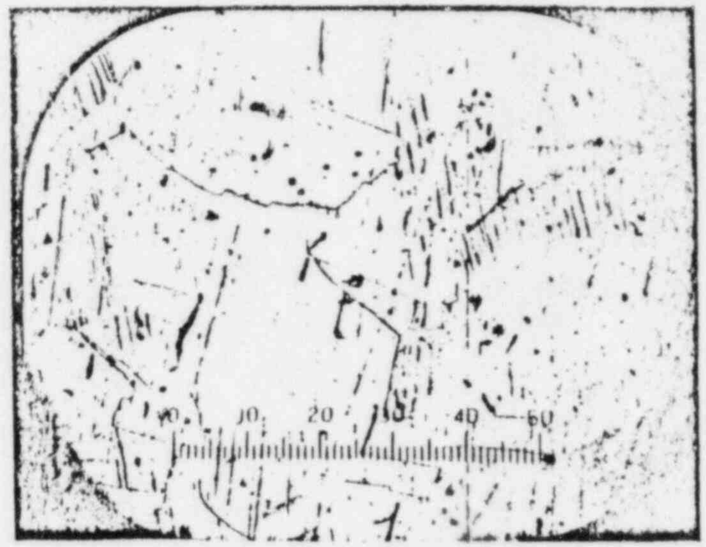


Figure 24. EDS scan of shank fracture face for Cr/Fe ratio (Cr/Fe - 1.743)

50X

Figure 25. Optical photomicrograph of shank bolt after ASTM A-262 test (Div.= 0.001mm)



50X

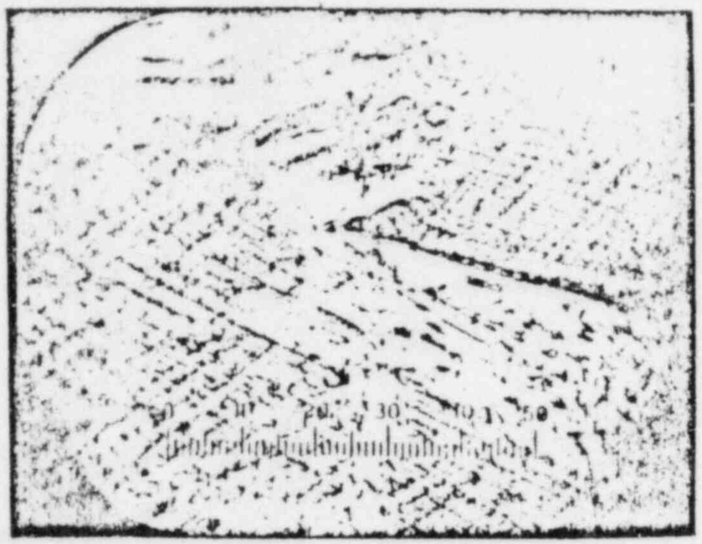
Figure 26. Optical photomicrograph of bolt head structure after ASTM A-262 (Div.= 0.001mm)

50X

Figure 27. Low magnification photo of a cold work area of bolt head (Div.=0.001mm)

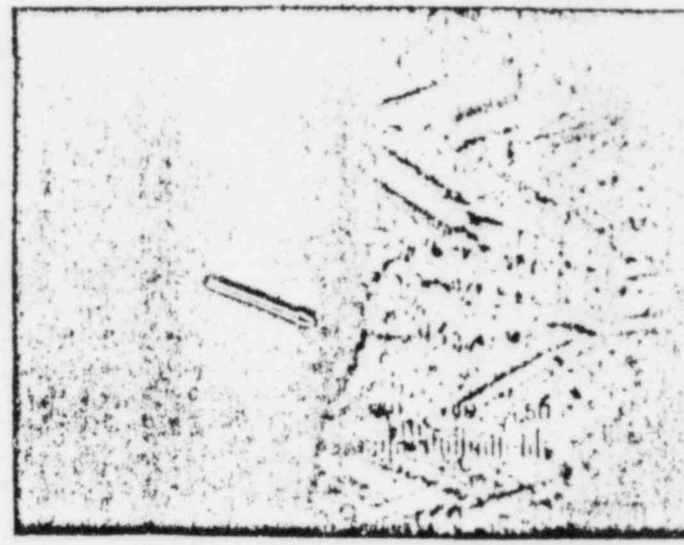
100X

Figure 28. Higher magnification photo of the cold work area (Div.=0.0005mm)



200X

Figure 29. Optical photomicrograph of a cold work area on bolt head (Div.=0.00025mm)

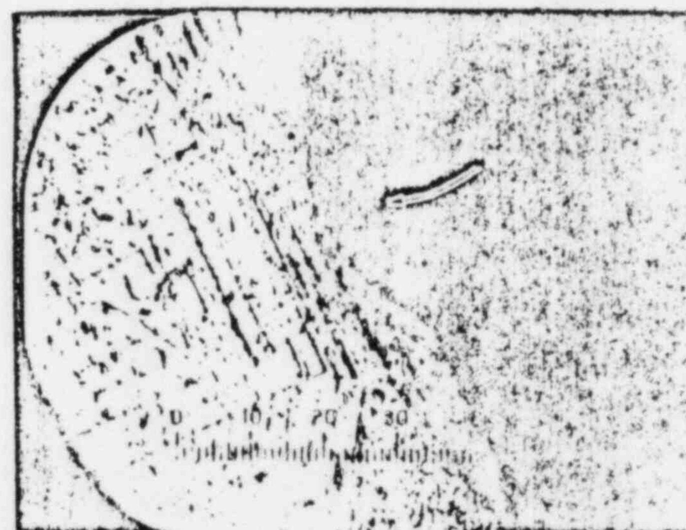


50X

Figure 30. Photomicrograph of pit on the edge of the bolt head (Div.=0.001mm)

50X

Figure 31. Photomicrograph of another pit on the edge of the bolt head (Div.=0.001mm)



100X

Figure 32. Higher magnification photo of Figure 29 (Div.=0.0005mm)