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Generic Task No. A-42

~~8/1779~~
NUREG-~~0XXX~~ Rev. 1

TECHNICAL REPORT ON

MATERIAL SELECTION AND PROCESSING~~, INSERVICE INSPECTION,~~

COOLANT PRESSURE BOUNDARY

~~AND LEAK DETECTION~~ GUIDELINES FOR BWR PIPING

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A. [illegible]

B. [illegible]

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ABSTRACT

This report updates and supersedes the NRC technical positions established in NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping", published in July 1977.

This report sets forth the NRC staff's revised acceptable methods to reduce the intergranular stress corrosion cracking susceptibility of BWR piping and safe ends. For plants that cannot fully comply with the material selection, testing and processing guidelines of this document, varying degrees of augmented inservice inspection and leak detection requirements are presented.

ASME Code Class 1 & 2 pressure boundary

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I. INTRODUCTION

This report ~~updates~~ the NRC technical positions defined in NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," July 1977. This NUREG report ~~represents work accomplished under~~ Generic Task No. A-42, "Pipe Cracks in Boiling Water Reactors."

Leaks and cracks in the heat-affected zones (HAZs) of welds that join austenitic stainless steel piping and associated components in BWRs have been observed since the mid-1960's. Prior to September 1974, the affected piping was mainly Type 304 stainless steel with diameters of 8 inches or less. All the cracks were attributed to ~~intergranular stress~~ corrosion crack~~ing~~ (IGSCC) due to the combination of high local stress, sensitization of material, and high oxygen content in the water. In each case, it was believed that the problem had been corrected or substantially reduced by better control of welding, contaminants, and design.

During the last quarter of 1974, a number of incidents of IGSCC in weld HAZs of 4-inch diameter recirculation bypass lines and in 10-inch diameter core spray lines were ~~again~~ observed. Following these occurrences, the Nuclear Regulatory Commission (NRC) formed a Pipe Cracking Study Group (PCSG) in January 1975 to (a) investigate the cause, extent, and safety implications of cracks, (b) make an interim recommendation for operating plants, and (c) recommend corrective actions to be taken by future plants. In October 1975, the Study Group published its report, NUREG-75/067, "Technical Report, Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water

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Intergranular stress corrosion cracking

- The guidelines for reducing the (IGSCC) susceptibility have been extended to cover ASME Code Class 2 piping.
- Inclusion of augmented inservice inspection requirements for nonconforming safe ends.
- Updating the inservice inspection sampling schemes to comply with the most recent NRC position.
- Identification of NUREG-0531 recommendations which cannot be implemented immediately without further NRC evaluation.

Reactor Plants." During the same general time span, the General Electric Company (GE) conducted an independent evaluation of cracking problems and submitted their findings and recommendations to the NRC (NEDO-21000, "Investigation of Cause of Cracking in Austenitic Stainless Steel Pipes"). Following staff review of the Study Group's and GE's recommendations, the staff issued an implementation document, NUREG-0313. This document, based on the information available at that time, set forth the NRC technical positions consistent with the recommendations of the Study Group.

Since 1975, IGSCC has continued to be detected in recirculation bypass and core spray lines. Incidence of IGSCC has also been observed in some stainless steel recirculation riser piping up to 12 inches in diameter in Japan and in large diameter (> 20 inches) recirculation piping in Germany. These incidents, together with the questions concerning the reliability of ultrasonic inspections, led to the ~~activation~~ of a new PCSG by NRC in September 1978.

The new Study Group was specifically chartered to ~~reexamine the~~ conclusions and

- the significance of the cracks discovered in large diameter pipes relative to the conclusions and recommendations set forth in the referenced report and in its implementation document, NUREG-0313;
- resolution of concerns raised over the ability of ultrasonic techniques to detect cracks in austenitic stainless steel;
- the significance of the cracks found in large diameter sensitized safe ends, and any recommendations regarding the current NRC program for dealing with this matter;
- the potential for stress corrosion cracking in PWRs; and
- the significance of the safe end cracking at Duane Arnold relative to similar material and design aspects at other facilities.

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In February 1979, the Study Group issued a report, NUREG-0531, "Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants." The new Study Group reaffirmed that the conclusions and recommendations reported in NUREG-75/067 by the previous group and the implementing document, NUREG-0313, are still valid. In addition, they presented some new ~~ideas~~ to reduce the potential for IGSCC and addressed IGSCC in safe ends. During the same general time span, the General Electric Company conducted an independent evaluation of the recent cracking in large diameter pipes and submitted their findings and recommendations to the NRC.[1] The GE main conclusions are: (a) IGSCC in Type 304 stainless steel weld HAZs remains to be a non-safety problem in spite of recent cracking in large diameter pipes, and (b) GE approach outlined in NEDO-21000 continues to be valid.

The IGSCC occurs in a small percentage of the welds in BWR piping which contains relatively stagnant, intermittent or low flow coolant. Historically, these cracks have been discovered either by volumetric examination, by visual inspection, or by leakage detection systems. The growth pattern of the cracks are such that it is unlikely that these cracks would go undetected before they grow to significant size where the pipe function might be compromised. Further, because of the inherent high material toughness of austenitic stainless steel piping, IGSCC is unlikely to cause a rapidly propagating failure resulting in a loss-of-coolant accident.

Although the likelihood is extremely low that these IGSCCs will propagate far enough to create a significant safety hazard to the public, the occurrence

[1] Letter from G. Sherwood to V. Stello, "General Electric Meeting with NRC on IGSCC," September 12, 1978.

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4 Following the issuance of the Study Group report in February 1979, the NRC noticed the availability of the report in the Federal Register and requested interested parties to provide any comments to the NRC by May 15, 1979. A copy of the Notice is attached as Appendix A. Comments were requested so that the staff would have the benefit of including any public comments prior to the development of the draft guidelines targeted for issuance in August 1979. In response to the staff's request, comments from six organizations and individuals were received. These comments are summarized in Appendix B.

of such cracks is undesirable. Measures should therefore be taken to minimize IGSCC in BWR piping systems and improve overall plant reliability.

It is the purpose of this document to set forth acceptable methods to reduce the IGSCC susceptibility of BWR piping and thus provide an increased level of reactor coolant pressure boundary and engineered safety features systems integrity. Recognizing that complete compliance with these guidelines may not be practical, or even possible, for all plants, varying degrees of conformance to our guidelines are provided in Part IV. For plants that cannot fully comply with the ~~provisions~~ specified in Part II of this document, varying degrees of augmented inservice inspection and leak detection requirements are ~~establshed~~ in Part III.

II. SUMMARY OF ACCEPTABLE METHODS TO MINIMIZE CRACK SUSCEPTIBILITY - MATERIAL SELECTION, TESTING, AND PROCESSING

A. Selection of Materials

Only those materials described in 1. and 2., below, are acceptable to the NRC for ~~use~~ in BWR piping systems. Other materials shall not be used without prior evaluation and acceptance by the NRC.

1. Corrosion Resistant Materials

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All pipe and fitting material including safe ends, thermal sleeves, and weld metal should be of a type and grade that has been demonstrated to be highly resistant to oxygen-assisted stress corrosion in the as-installed condition. Materials which have been so demonstrated include ferritic steels, "Nuclear Grade" austenitic stainless steels,[2] Types 304L and 316L austenitic stainless steels, Type CF-3 cast stainless steel, and Type 308L stainless steel weld metal with at least 5% ferrite content. Unstabilized wrought austenitic stainless steel without controlled low carbon does not meet this requirement unless all such piping including welds is in the solution-annealed condition. The use of such material (i.e., regular grades of Types 304 and 316 stainless steels) should be avoided except in the solution- annealed condition and then only for non-welded applications. Where regular grades of Types 304 and 316 are used and welding or heat treatment is required, special measures should be taken to ensure that IGSCC will not occur. Such measures may include (1) solution annealing subsequent to the welding or heat treatment, and (2) weld cladding of materials to be welded using techniques which have been demonstrated to eliminate sensitization and reduce residual stresses.

2. <u>Corrosion Resistant "Safe Ends" and Thermal Sleeves</u>

All unstabilized wrought austenitic stainless steel materials used for safe ends and thermal sleeves without controlled low carbon

[2] These materials have controlled low carbon (0.02% max) and nitrogen (0.1% max) contents and meet all requirements, including mechanical property requirements, of ASME specification for regular grades of Type 304 or 316 stainless steel pipe.

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contents (L-grades and Nuclear grade) should be in the solution-annealed condition. If as a consequence of fabrication, welds joining these materials are not solution-annealed, they should be made between cast (or weld overlaid) austenitic stainless steel surfaces (5% minimum ferrite) or other materials having high resistance to oxygen-assisted stress corrosion. The joint design must be such that any high stress areas in unstabilized wrought austenitic stainless steel without controlled low carbon content, which may become sensitized as a result of the welding process, is not exposed to the reactor coolant. Thermal sleeve attachment geometries that form crevices where impurities may accumulate should not be exposed to a BWR coolant environment.

B. Testing of Materials

For new installations,

Tests should be made on all regular grade stainless steels to demonstrate that the material was properly annealed and is not susceptible to IGSCC. Such tests may include Practices A[3] and E[4] of ASTM A-262, "Recommended Practices for Detecting Susceptibility to Intergranular Attack in Stainless Steels."

The Electrochemical Potentiokinetic Reactivation (EPR) test which is presently being developed and qualified has not yet been formally evaluated and accepted by the NRC.

[3] Practice A - Oxalic Acid Etch Test for Classification of Etch Structures of Stainless Steels.

[4] Practice E - Copper-Copper Sulfate-Sulfuric Acid Test for Detecting Susceptibility to Intergranular Attack in Stainless Steels.

C. Processing of Materials

For initial construction and for repair welds with Type 304 or 316 stainless steel pipes, a corrosion resistant clad with a duplex microstructure (5% minimum ferrite) can be weld deposited on the prewelded surfaces to (a) minimize the HAZ on the pipe inner surface, (b) move the HAZ away from the highly stressed region next to the attachment weld, and (c) isolate the weldment from the environment. For initial later construction, the piping including clad surfaces should be solution-annealed prior to making the attachment welds. The joint design of all welds must be such that any high stress areas in the unstabilized wrought austenitic stanless steel, which may become sensitized as a result of the welding process, is not exposed to the reactor coolant.

Other processes for minimizing stresses and IGSCC in austenitic stainless steel weldments such as Induction Heating Stress Improvement (IHSI) and Heat Sink Welding (HSW) are currently being developed and have not yet been formally evaluated and accepted by the NRC.

III. INSERVICE INSPECTION AND LEAK DETECTION REQUIREMENTS FOR BWRs WITH VARYING DEGREES OF CONFORMANCE TO MATERIAL SELECTION, TESTING, AND PROCESSING GUIDELINES

A. For plants whose ASME Code Class 1 and 2 pressure boundary piping meets the guidelines of Part II, no augmented inservice inspection or leak detection requirements beyond those specified in the ~~present plant's~~ 10 CFR 50.55a(g), "Inservice Inspection Requirements," and the latest NRC Standard Technical Specifications for [illegible] ~~technical specifications~~ are necessary.

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B. ASME Code Class 1 and 2 pressure boundary piping that does not meet guidelines of Part II is designated nonconforming and must have additional inservice inspection and more stringent leak detection requirements. The degree of augmented inspection of such piping depends on whether the specific nonconforming piping runs are classified as "Service Sensitive." "Service Sensitive" lines were and will be inspected by the NRC and are defined as those that have experienced cracking of a generic nature in service, or that are considered to be particularly susceptible to cracking because of a combination of high local stress, material condition, and high oxygen content in the relatively stagnant, intermittent, or low flow coolant.

Examples of piping runs considered to be "Service Sensitive" include but are not limited to: core spray lines, recirculation riser lines, recirculation bypass lines (or "stub tubes" on plants where the bypass lines have been removed), CRD hydraulic return lines, isolation condenser lines, recirculation inlet lines with crevices formed by thermal sleeve attachment, and shutdown heat exchanger lines. If cracking is found in a particular piping run [illegible] it will be considered [illegible] "Service Sensitive."

Leakage detection and augmented inservice inspection requirements for nonconforming lines and nonconforming, service sensitive lines are specified below:

1. Nonconforming Lines that are not Service Sensitive

   a. Leak Detection: The reactor coolant leakage detection systems should be upgraded to enhance the discovery of unidentified

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leakage that may include through-wall cracks developed in austenitic stainless steel piping.

(1) The leakage detection system provided should include sufficiently diverse leak detection methods with adequate sensitivity to detect and measure small leaks in a timely manner and to identify the leakage sources within the practical limits. Acceptable leakage detection and monitoring systems are described in Section C, Regulatory Position of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

Particular attention should be given to upgrading and calibrating those leak detection systems that will provide prompt indication of an increase in leakage rate.

Other equivalent leakage detection and collection systems will be reviewed on a case-by-case basis.

(2) Plant shutdown should be initiated for inspection and corrective action when any of the leakage detection systems indicates, within a period of 24 hours or less, an increase in the rate of unidentified leakage in excess of 2 gallons per minute or its equivalent, or when the total unidentified leakage attains a rate of 5 gallons per minute or its

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equivalent, whichever occurs first. For sump level monitoring systems with fixed measurement interval method, the level should be monitored at 4-hour intervals or less.

(3) Unidentified leakage should include all leakage other than:

(a) Leakage into closed systems, such as pump seal or valve packing leaks that are captured, flow metered, and conducted to a sump or collecting tank, or

(b) Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operations of unidentified leakage monitoring systems or not to be from a through-wall crack in the piping within the reactor coolant pressure boundary.

b. Augmented Inservice Inspection: Inservice inspection of the nonconforming, nonservice sensitive lines should be conducted in accordance with the following program:

(1) For ASME Code Class 1 components and piping, each pressure retaining dissimilar metal weld subject to inservice inspection requirements of Section XI should be examined at least once in no more than 80 months (two-thirds of the time prescribed in the ASME Boiler and Pressure Vessel Code

Section XI). Such examination should include all internal attachments that are not through-wall welds but are welded to or form part of the pressure boundary.

(2) The following ASME Code Class 1 pipe welds subject to inservice inspection requirements of Section XI should be examined at least once in no more than 80 months:

(a) all welds at terminal ends[5] of pipe at vessel nozzles;

(b) all welds having a designed combined primary plus secondary stress range of $2.4S_m$ or more;

(c) all welds having a design cumulative fatigue usage factor of 0.4 or more; and

(d) sufficient additional welds with high potential for cracking to make the total equal to 25% of the welds in each piping system.

(3) The following ASME Code Class 2 pipe welds which are subject to inservice inspection requirements of Section XI, excluding

[5] Terminal ends are the extremities of piping runs that connect to structures, components (such as vessels, pumps, valves) or pipe anchors, each of which acts as rigid restraints or provides at least two degrees of restraint to piping thermal expansion.

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those in Residual Heat Removal Systems, Emergency Core Cooling Systems and Containment Heat Removal Systems, should be inspected at least once in no more than 80 months:

(a) all welds at locations where the stresses under the loadings resulting from Normal and Upset plant conditions including the Operating Basis Earthquake (OBE) as calculated by the sum of Equations (9) and (10) in NC-3652 exceed 0.8 $(1.2S_h + S_A)$;

(b) all welds at terminal ends of piping, including branch runs;

(c) all dissimilar metal welds.

(d) additional welds with high potential for cracking at structural discontinuities[6] such that the total number of welds selected for examination includes the following percentages of circumferential piping welds:

[6] Structural discontinuities include pipe weld joints to vessel nozzles, valve bodies, pump casings, pipe fittings (such as elbows, tees, reducers, flanges, etc., conforming to ANSI Standard B 16.9) and pipe branch connections and fittings.

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(i) 50% of the main steam system welds, and

(ii) 25% of the welds in all other systems.

(4) The following ASME Code Class 2 pipe welds, subject to inservice inspection requirements of Section XI, in each Residual Heat Removal Systems, Emergency Core Cooling Systems, and Containment Heat Removal Systems should be examined at least once in no more than 80 months:

(a) all welds of the terminal ends of pipe at vessel nozzles, and

(b) at least 10% of the welds selected proportionately from the following categories:

(i) circumferential welds at locations where the stresses under the loadings resulting from <u>any</u> plant conditions as calculated by the sum of Equations (9) and (10) in NC-3652 exceed 0.8 $(1.2S_h + S_A)$;

(ii) welds at terminal ends of piping, including branch runs,

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(iii) dissimilar metal welds,

(iv) welds at structural discontinuities, and

(v) welds that cannot be pressure tested in accordance with IWC-5000.

The welds to be examined shall be distributed approximately equally among runs (or portions of runs) that are essentially similar in design, size, system function, and service conditions.

(5) If examinations of (1), (2), (3), and (4) above, conducted during the first 80 months reveal no incidence of stress corrosion cracking, the examination frequency thereafter can revert to 120 months as prescribed in Section XI of the ASME Boiler and Pressure Vessel Code.

Sampling schemes other than those described in (2), (3), and (4) above for the same total number of welds will be reviewed on a case-by-case basis.

2. Nonconforming Lines that are Service Sensitive

a. Leak Detection: The leakage detection requirements, described in IIIB1a above, should be implemented.

b. Augmented Inservice Inspection:

(1) The welds and adjoining areas of bypassing piping of the discharge valves in the main recirculation loops, and of the austenitic stainless steel reactor core spray piping up to and including the second isolation valve, should be examined at each reactor refueling outage or at other scheduled or unscheduled plant outages. Successive examinations need not be closer than 6 months, if outages occur more frequently than 6 months. This requirement applies to all welds in all bypass lines whether the 4-inch valve is kept open or closed during operation.

In the event these examinations find the piping free of unacceptable indications for three successive inspections, the examination may be extended to each 36-month period (plus or minus by as much as 12 months) coincident with a refueling outage. In these cases, the successive examination may be limited to all welds in one bypass pipe run and

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one reactor core spray piping run. If unacceptable flaw indications are detected, the remaining piping runs in each group should be examined.

In the event these 36-month period examinations reveal no unacceptable indications for three successive inspections, the welds and adjoining areas of these piping runs should be examined at least once on a sampling basis as described in III.B.1.b(1) for dissimilar metal welds and (2) for other welds above at a frequency of an 80-month period.

(2) The welds and adjoining areas of other ASME Code Class 1 service sensitive piping should be examined on a sampling basis as described in III.B.1.b(1) for dissimilar metal welds and (2) for other welds above except that the frequency of such examinations should be at each reactor refueling outage or at other scheduled or unscheduled plant outages. Successive examinations need not be closer than 6 months, if outages occur more frequently than 6 months.

In the event these examinations find the piping free of unacceptable indications for three successive inspections, the examination may be extended to each 36-month period (plus or minus by as much as 12 months) coinciding with a refueling outage.

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In the event these 36-month period examinations reveal no unacceptable indications for three successive inspections, the frequency of examination may revert to 80-month periods (two-thirds the time prescribed in the ASME Code Section XI).

(3) All pressure retaining welds and adjoining areas of one half of the creviced safe ends including the internal attachment welds in recirculation inlet lines should be examined during the next refueling outage. The welds and adjoining areas of the other half of the creviced safe ends and internal attachment welds should be examined during the subsequent refueling outage. This sequence of inspections shall be maintained through the subsequent alternate refueling outages.

In the event the examinations of each safe end and internal attachment weld reveal no unacceptable indications for three successive inspections, the frequency of examination of each safe end and internal attachment weld may revert to 80-month periods.

(4) The area, extent, and frequency of examination of the augmented inservice inspection for ASME Code Class 2 service sensitive lines will be determined on a case-by-case basis.

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IMAGE EVALUATION TEST TARGET (MT-3)

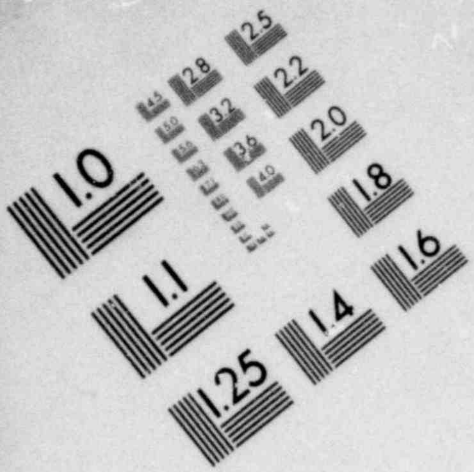
6″

IMAGE EVALUATION
TEST TARGET (MT-3)

6"

IMAGE EVALUATION TEST TARGET (MT-3)

6″



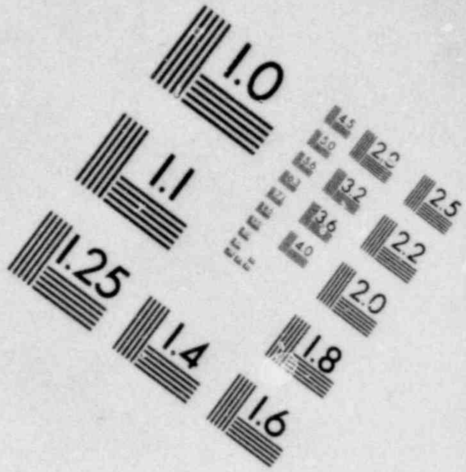
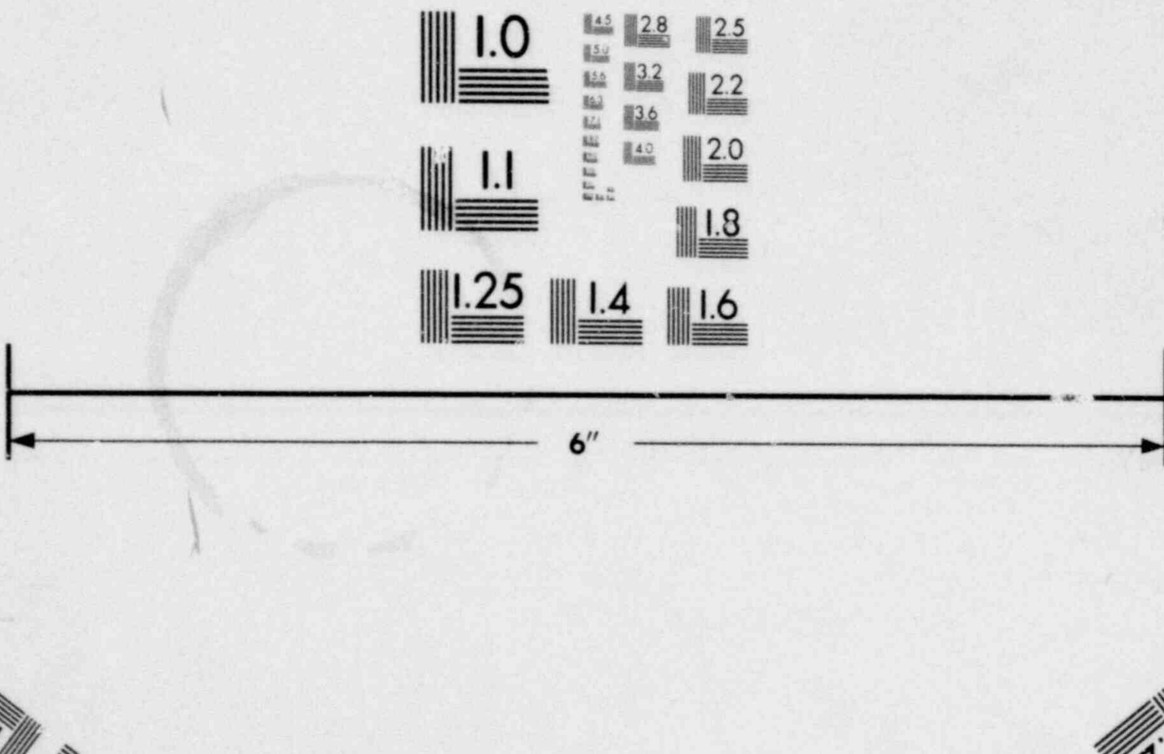
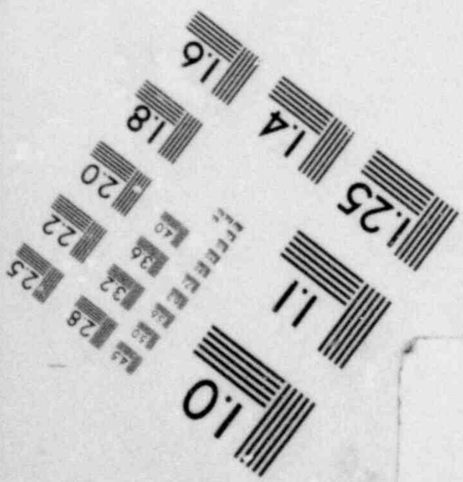
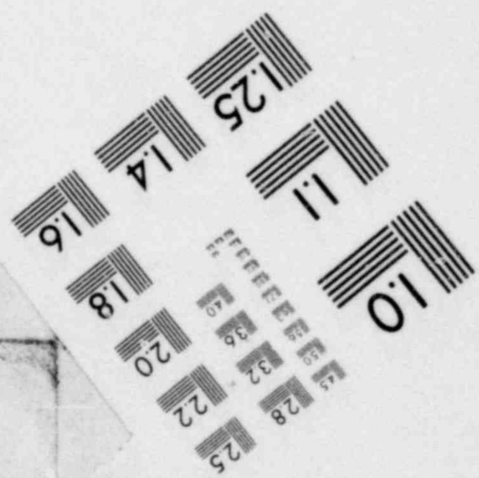


IMAGE EVALUATION TEST TARGET (MT-3)







3. Nondestructive Examination (NDE) Requirements

The method of examination and volume of material to be examined, the allowable indication standards, and examination procedures should comply with the requirements set forth in the applicable Edition and Addenda of the ASME Code, Section XI, specified in paragraph (g), "Inservice Inspection Requirements," of 10 CFR 50.55a, "Codes and Standards." In addition, as a minimum, all reflectors should be mapped with respect to geometry and a procedure should be implemented by which amplitude and metal path are recorded automatically.

In some cases, the code examination procedures may not be effective for detecting or evaluating IGSCC and other ultrasonic (UT) procedures or advanced nondestructive examination techniques may be required to detect and evaluate stress corrosion cracking in austenitic stainless steel piping. Improved UT procedures have been developed by certain organizations. Specific guidance for implementation of these improved techniques cannot be provided at this time. Recommendations for the development and eventual implementation of these improved techniques are included in Part V.

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IV. IMPLEMENTATION OF MATERIAL SELECTION, TESTING, AND PROCESSING GUIDELINES

A. For plants under review, but for which a construction permit has not been issued, all lines should conform to the guidelines stated in Part II.

B. For plants that have been issued a construction permit, all lines should conform to the guidelines stated in Part II unless it can be demonstrated to the staff that implementing the guidelines of Part II would result in undue hardship. Where the guidelines of Part II are not complied with, additional measures should be taken in accordance with the guidelines stated in Part III of this document.

C. For plants that have been issued an operating license, NRC designated service sensitive lines should be modified to conform to the guidelines stated in Part II, to the extent practicable. Lines that ~~have~~ experienced cracking ~~in plant~~ during service and are required to be replaced should be replaced with piping that conforms to the guidelines stated in Part II. Where the guidelines of Part II are not complied with, additional measures should be taken in accordance with the guidelines stated in Part III of this document.

V. GENERAL RECOMMENDATIONS

The measures outlined in Part II of this document provide for positive actions that are consistent with current technology. The implementation of these actions should markedly reduce the susceptibility of stainless steel piping to stress corrosion cracking in BWRs. It is recognized that additional means could be used to limit the extent of corrosion of BWR pressure boundary piping materials and to improve the overall system integrity. These include plant design and operational procedure considerations to reduce system exposure to

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potentially aggressive environment, improved material selection, special fabrication and welding techniques, and provisions for volumetric inspection capability in the design of weld joints. The use of such means to limit IGSCC will be reviewed on a case-by-case basis.

Although ~~The~~ the items identified below are not required for the present plant safety, they may be expected to lead to means of limiting the extent of IGSCC and improving the chance ~~changes~~ of detecting such IGSCC. These items have not yet been fully developed and accepted by the NRC.

Specifically, areas that ~~need~~ require further ~~NRC and/or industry~~ consideration are:

A. Improved ultrasonic inspection methods. Such methods should be included in the ASME ~~codified~~ Code or included in a Regulatory Guide.

B. Development and implementation of an improved focused inservice inspection program based on stress rule index, material of construction, history of cracking, etc.

C. Improved weld joint design to assure that required examinations can be performed effectively. ~~for better inspectability.~~

D. Reduction of oxygen content in reactor coolant during all phases of reactor operation by water chemistry control, de-aeration of systems, etc.

E. Minimization of stagnant or low flow coolant pressure boundary piping.

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F. Evaluation of newly developed alternate corrosion resistant materials in BWR environment.

G. Improvement of material corrosion resistance by cladding, heat sink welding, induction heating stress improvement, mechanical working of the inside surface of pipe welds, etc.

H. Evaluation of the Electrochemical protentiokinetic reactivation technique for detecting and quanitifying the degree of sensitization in stainless steel piping.

Continued evaluation and verification

I. ~~Reevaluation~~ of leak before break ~~postulation~~, concept.

and implementation

J. Evaluation of leakage detection capability to improve early d tection of small leaks.

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APPENDIX A

NRC NOTICE IN THE FEDERAL REGISTER
REQUESTING PUBLIC COMMENT ON NUREG-0531

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1737 005

[7590-01-M]

DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

Investigation and Evaluation of Stress Corrosion Cracking in Piping of Light Water Reactor Plants

AGENCY: U.S. Nuclear Regulatory Commission.

ACTION: Request for public comment on NUREG-0531 "Investigation and Evaluation of Stress Corrosion Cracking in Piping of Light Water Reactor Plants" February 1979.

SUMMARY: On September 14, 1978, the Nuclear Regulatory Commission established a new Pipe Crack Study Group. The Group was to evaluate recent pipe and safe end cracking experience relative to previous staff conclusions and recommendations. The NRC seeks public comment on the report which summarizes the Group's review and conclusions.

DATES: The public comment period expires May 15, 1979.

FOR FURTHER INFORMATION CONTACT:

Darrell G. Eisenhut, Deputy Director for Operating Reactors, Division of Operating Reactors, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. (Phone: 301-492-7221)

SUPPLEMENTARY INFORMATION: In 1975, a Pipe Cracking Study Group was established by the United States Nuclear Regulatory Commission (USNRC) to review intergranular stress-corrosion cracking (IGSCC) in Boiling Water Reactors (BWRs). The Group reported its findings concerning stress-corrosion cracking in by-pass lines and core spray piping of austentic stainless steel in a report, *Technical Report—Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants* (NUREG-75/067).

During 1978, IGSCC was reported for the first time in large-diameter piping in a BWR. This discovery, together with questions concerning the capability of ultrasonic detection methods to detect small cracks, led to the formation of a new Pipe Crack Study Group (PCSG) by USNRC on September 14, 1978.

The charter of the new PCSG was to specifically address the five following questions:

"1. The significance of the cracks discovered in large-diameter pipes relative to the conclusions and recommendations set forth in the referenced report (NUREG-75/067) and its implementation document, NUREG-0313;

2. Resolution of the concerns raised over the ability to use ultrasonic techniques to detect cracks in austenitic stainless steel;

3. The significance of cracks found in large-diameter sensitized safe ends and any recommendations regarding the current NRC program for dealing with this matter;

4. The potential for stress corrosion cracking in PWRs;

5. Examine the significance of cracking in the Inconel safe ends that has been experienced at the Duane Arnold Operating Facility, and develop any recommendations regarding NRC actions taken or to be taken."

The PCSG limited the scope of the study to BWR and PWR piping and safe ends attached to the reactor pressure vessel. The PCSG reviewed existing information—either that contained in written records or that collected through meetings in this country and in foreign countries. The specific areas considered are presented in the chapters of this report:

- BWR Cracking Experience and Corrective Actions
- PWR Cracking Experience and Corrective Actions
- Metallurgy Associated with Pipe Cracking
- Reactor Coolant Chemistry
- Pipe Configuration and Stress Levels
- Duane Arnold Safe-End Cracking
- Methods of Detecting Cracks
- Significance of Cracks
- Recent Development Relevant to Control and Detection of IGSCC

The review of these topics in the context of changes occurring since the preparation of NUREG-75/067 led to the preparation of specific conclusions and recommendations relevant to the current status of IGSCC, the significance of the problem, and the reliability of detection and measures available to correct or minimize IGSCC in existing and future plants. These conclusions and recommendations are presented in the newly issued PCSG report.

The NRC staff will review the Study Group report and its conclusions/recommendations and the public comments received during this comment period. Following this review, the staff will decide what further actions, if any, are required for the licensing and operation of reactors.

Requests for a single copy of the report should be made in writing to U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Technical Information and Document Control.

Comments on this report should be sent to the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Duputy Director, Division of Operating Reactors. The comment period expires May 15, 1979. Copies of all comments received will be available for examination in the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C.

Dated at Bethesda, Md., this 6th day of March, 1979.

For the Nuclear Regulatory Commission.

VICTOR STELLO, Jr.,
Director, Division of Operating Reactors, Office of Nuclear Reactor Regulation.

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APPENDIX B

SUMMARY OF PUBLIC COMMENTS ON NUREG-0531

In response to NRC's request, comments on NUREG-0531, "Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants" from the following ~~seven~~ six organizations and individuals were received. Their substantive comments are summarized below:

A. General Electric Company

1. The use of regular grades of Type 304 and 316 stainless steel in BWR piping systems should be avoided unless carbon content is restricted to 0.35% or less. If regular grades without special carbon restrictions are used, steps should be taken to ensure that IGSCC cannot occur. Such measures may include non welded applications, solution annealing, weld cladding, or other measures that have been adequately tested to provide reasonable assurance of reliable performance during the life of the plant. (paragraph 4.8)

2. The use of IHSI on existing plant welds raises some areas for further investigation. The effect of the treatment on existing cracks should be determined, as well as any effects of the thermomechanical IHSI cycle.

   It is recognized that IHSI provides residual stress reduction in piping of all diameters. GE-EPRI has planned tests to determine the extent of benefit to be derived from IHSI. (paragraph 10.5.1)

3. It is recommended that the recommendations contained in NUREG-0313 continue to be considered for operating plants and plants under review for an operating license or construction permit. On a case by case basis plans should be developed for in-service inspection which would improve the probability of early crack identification. These plans should consider differences in stress, carbon content, degree of material sensitization and the frequency of past cracking incidents in other plants as well as other factors related to plant operation and inspection history. (paragraph 2.11)

4. Based on the incidence of IGSCC in recirculation-riser piping in the offshore plants, it is recommended that an augmented in-service inspection program considering the above factors be developed for these lines. (paragraph 2.11)

5. Further clarification is requested on the second recommendation relative to safe ends on reactor pressure vessels. General Electric considers that special inspections of uncreviced safe-ends with tuning fork designs are not warranted. (p. 7.4)

6. To ensure that General Electric is aware of the complete list of NRC identified field cracking incidents in piping, it is requested that a detailed list be provided of these incidents by plant and line type. (p. 2.1)

B. Washington Public Power Supply System (WPPSS)

1. WPPS questions whether there is sufficient experience to warrant placing the riser lines in the service-sensitive category. It only requires one minor extension of this logic to place the whole system in this category. (paragraph 2.11)

2. Fabrication of Materials - WPPSS feels that more discussion is warranted on the merits and adequacy of ASTM A-262 for acceptance of materials used in environments conducive to stress corrosion cracking. By using the techniques in ASTM A-262, are we possibly accepting material which is partially sensitized prior to welding? (paragraph 4.2.3)

C. Combustion Engineering - Power Systems

1. There appears to be an error in the specification for Boron concentrations in Table 5.2, "Summary of PWR Reactor Coolant Chemistry Specifications". The correct refueling boron concentration should be $<$ 4400 ppm.

D. Carolina Power and Light Company

1. There is not sufficient justification for reclassifying the recirculation - riser piping as nonconforming, service sensitive lines. (Recommendation 2.11.2)

2. It is not practical to require utilities to reclassify their welded attachments as nonconforming, service sensitive lines. The welds, in most cases, do not have configurations that will allow ultrasonic inspections. (Recommendation 7.4.1)

E. Lawrence Livermore Laboratory

1. Long term effects of redistribution of stress must be considered when any heating and cooling cycle is superimposed on an existing welding process. There is the possibility that cracks that occur could propagate without arrest because of the new stress redistribution created by IHSI. (p.10.4)

2. The report does not state how the results of A262 A and E compare with the lots of stainless steel which have experienced IGSCC in the BWR environment. There should be more discussion of electrochemical potentiokinetic reactivation technique (p. 4.3).

3. Since the critical level of sensitization is probably a critical level of chromium depletion around the carbides, measurements which emphasize these critical parameters should be the basis of regulatory requirements. (p. 4.4)

4. Further study is needed on the role of residual stress distribution on crack growth. The residual stress distribution in circumferential welds tends to promote cracking all around the circumference. As the crack extends around a significant portion of the inside wall, the residual stresses in the axial direction should increase and accelerate the crack growth. (p. 6.4)

5. The tearing modulus concept appears to be still at a research level. A major comprehensive study of this topic appears to be justified. (p. 9.1)

6. The relation of leakage rate to crack size should be studied relative to its usage as a criterion for crack detection. (p. 8.5)

7. The 3-D presentation of internal defects by acoustical holography certainly warrants consideration as a complementary technique to ultrasonic testing.

F. Southwest Research Insititue (SWRI)

1. SWRI agrees with Conclusions 1, 2, 3, and 4, Chapter 8. However, it should be noted that sizing is not as important as the detection of IGSCC. Once a reflector is identified as IGSCC, the area must be repaired under present requirements. Therefore, there is no need to size the cracks at this time. (p.8.7)

2. It is not SWRI's opinion that all reflectors observed within the HAZ should be classified as IGSCC. However, all crack-like indications within the HAZ of suspect austenitic welds should be classified as IGSCC. (p. 8.7)

3. The technique mentioned in Recommendation 2 of Chapter 8 may serve to reduce radiation exposure. However, it does nothing to improve the technical adequacy or credibility of the examination. In fact, this approach may, at times, reduce the adequacy of the examination. It should be noted that automatic recording and analysis of signal response and positional data for manual examinations will, in the near future, provide improved examinations while reducing radiation exposure to examination personnel. (p.8.7)