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Closeout of IE Bulletin 79–17: Pipe Cracks in Stagnant Borated Water Systems at PWR Plants

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ABSTRACT

Documentation is provided in this report for the closeout of IE Bulletin 79-17 and its revision on the safety-related subject of pipe cracks in stagnant borated water systems at operating plants with pressurized water reactors (PWRs). Closeout is based on the implementation and verification of actions required by the bulletin. Evaluation of utility responses and NRC/Region inspection reports indicates that the bulletin is closed for all of the 41 operating PWRs to which it was issued for action. It is concluded that the concerns of the bulletin have been resolved. Background information is supplied in the Introduction and Appendix A.

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CLOSEOUT OF IE BULLETIN 79-17: PIPE CRACKS IN STAGNANT BORATED WATER SYSTEMS AT PWR PLANTS

INTRODUCTION

This report provides documentation for the closeout of IE Bulletin 79-17 in accordance with the Statement of Work in Task Order 37 under NRC Contract 05-85-157-02. Documentation is based on the records obtained from the NRC Document Control System.

IE Bulletin 79-17 and its Revision 1 were issued on July 26 and October 29, 1979, respectively. Operators of pressurized water reactors (PWRs) were required to take specific actions. The bulletin was forwarded for information only to all other licensees and holders of construction permits of nuclear power reactors. The concern was cracking of piping in safety-related systems containing stagnant borated water. As indicated in the bulletin (see page A-1), the cracking problem had been identified at enough plants to be considered generic.

IE Circular 76-06 (see page A-10 of this report) was issued on November 26, 1976, to describe several incidents of through-wall cracking in low-pressure safety-related piping systems. The systems of concern were not frequently flushed, or contained nonflowing liquids. Licensees were required to take five specific actions, including reporting.

IE Bulletin 79-17 Revision 1 (see page A-5) was issued three months later than the initial bulletin, in order to clarify certain terms and requirements and to allow more time because of the shortage of qualified personnel for ultrasonic examinations. The term "stagnant oxygenated borated water systems" was clarified to refer "to those systems serving as engineered safeguards having no normal operating functions and containing essentially air saturated borated water where flow conditions do not exist on a <u>continuous basis</u>".

Copies of IE Bulletin 79-17, Revision 1 of the bulletin, and IE Circular 76-06 are included in Appendix A of this report for presentation of background information and required actions. Also included in Appendix A is a summary of the NRC/IE review of the bulletin results. Evaluation of utility responses and NRC/Region inspection reports is documented in Appendix B as the basis for bulletin closeout. Abbreviations used in this report and associated documents are listed in Appendix C.

SUMMARY

1. The bulletin is closed for the following 41 facilities per the Criterion (see page B-6):

Arkansas 1,2	Indian Point 2,3	Robinson 2
Beaver Valley 1	Kewaunee	Salem 1
Calvert Cliffs 1,2	Maine Yankee	San Onofre 1
Cook 1,2	Millstone 2	St. Lucie 1
Crystal River 3	North Anna 1	Surry 1,2
Davis-Besse 1	Oconee 1,2,3	TMI 1
Farley 1	Palisades	Trojan
Fort Calhoun 1	Point Beach 1,2	Turkey Point 3,4
Ginna	Prairie Island 1,2	Yankee-Rowe 1
Haddam Neck	Rancho Seco 1	Zion 1,2

2. The bulletin is open for no facilities.

3. Because they are shut down indefinitely or permanently, the following PWR facilities are excluded from Table B.1:

Indian Point 1	TMI 2

CONCLUSIONS

1. Evaluation of utility responses and NRC/Region inspection reports indicates that the concerns of the bulletin have been resolved.

2. For backup of Conclusion 1 above, refer to the summary of results on page A-13.

APPENDIX A

Background Information and Required Actions

Notes:

1. IE Circular 76-06 (see page A-10) was enclosed with both the initial bulletin and Revision 1 of the bulletin.

2. For actions required by the initial issue of the bulletin, see pages A-2 through A-4; for actions required by Revision 1 of the bulletin, see pages A-7 through A-9.

3. Revision 1 of the bulletin need not be addressed by a utility if the following requirements are met in the initial response(s):

   (a) The definition used for "stagnant" is correct, as clarified in Revision 1.

   (b) "Facilities having previously experienced cracking in identified systems, Item 1, are requested to identify (list) the new materials utilized in repair or replacement on a system-by-system basis. If a report of this information and that requested above has been previously submitted to the NRC, please reference the specified report(s) in response to this bulletin." (See Item 1.d, page A-7).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D. C. 20555

July 26, 1979

IE Bulletin No. 79-17

PIPE CRACKS IN STAGNANT BORATED WATER SYSTEMS AT PWR PLANTS

Description of Circumstances:

During the period of November 1974 to February 1977 a number of cracking incidents have been experienced in safety-related stainless steel piping systems and portions of systems which contain oxygenated, stagnant or essentially stagnant borated water. Metallurgical investigations revealed these cracks occurred in the weld heat affected zone of 8-inch to 10-inch type 304 material (schedule 10 and 40), initiating on the piping I.D. surface and propagating in either an intergranular or transgranular mode typical of Stress Corrosion Cracking. Analysis indicated the probable corrodents to be chloride and oxygen contamination in the affected systems. Plants affected up to this time were Arkansas Nuclear Unit 1, R. E. Ginna, H.B.Robinson Unit 2, Crystal River Unit 3, San Onofre Unit 1, and Surry Units 1 and 2. The NRC issued Circular 76-06 (copy attached) in view of the apparent generic nature of the problem.

During the refueling outage of Three Mile Island Unit 1 which began in February of this year, visual inspections disclosed five (5) through-wall cracks at welds in the spent fuel cooling system piping and one (1) at a weld in the decay heat removal system. These cracks were found as a result of local boric acid build-up and later confirmed by liquid penetrant tests. This initial identification of cracking was reported to the NRC in a Licensee Event Report (LER) dated May 16, 1979. A preliminary metallurgical analysis was performed by the licensee on a section of cracked and leaking weld joint from the spent fuel cooling system. The conclusion of this analysis was that cracking was due to Intergranular Stress Corrosion Cracking (IGSCC) originating on the pipe I.D. The cracking was localized to the heat affected zone where the type 304 stainless steel is sensitized (precipitated carbides) during welding. In addition to the main through-wall crack, incipient cracks were observed at several locations in the weld heat affected zone including the weld root fusion area where a miniscule lack of fusion had occurred. The stresses responsible for cracking are believed to be primarily residual welding stresses in as much as the calculated applied stresses were found to be less than code design limits. There is no conclusive evidence at this time to identify those aggressive chemical species which promoted this IGSCC attack. Further analytical efforts in this area and on other system welds are being pursued.

Based on the above analysis and visual leaks, the licensee initiated a broad based ultrasonic examination of potentially affected systems utilizing special techniques. The systems examined included the spent fuel, decay heat removal, makeup and purification, and reactor building spray systems which contain stagnant or intermittently stagnant, oxygenated boric acid environments. These systems range from 2 1/2-inch (HPCI) to 24-inch (borated water storage tank suction), are type 304 stainless steel, schedule 160 to schedule 40 thickness respectively. Results of these examinations were reported to the NRC on June 30, 1979 as an update to the May 16, 1979 LER. The ultrasonic inspection as of July 10, 1979 has identified 206 welds out of 946 inspected having UT indications characteristic of cracking randomly distributed throughout the aforementioned sizes (24"-14"-12"-10"-8"-2" etc.) of the above systems. It is important to note that six of the crack indications were found in 2 1/2-inch diameter pipe of the high pressure injection lines inside containment. These lines are attached to the main coolant pipe and are nonisolable from the main coolant system except for check valves. All of the six cracks were found in two high pressure injection lines containing stagnated borated water. No cracks were found in the high pressure injection lines which were occasionally flushed during makeup operations. The ultrasonic examination is continuing in order to delineate the extent of the problem.

The above information was previously provided in Information Notice 79-19.

For All Pressurized Water Reactor Facilities with an Operating License:

1. Conduct a review of safety related stainless steel piping systems within 30 days of the date of this Bulletin to identify systems and portions of systems which contain stagnant oxygenated borated water. These systems typically include ECCS, decay/residual heat removal, spent fuel pool cooling, containment spray and borated water storage tank (BWST-RWST) piping.

    (a) Provide the extent and dates of the hydrotests, visual and volumetric examinations performed per 10 CFR 50.55a(g) (Re: IE Circular 76-06 enclosed) of identified systems. Include a description of the non-destructive examination procedures, procedure qualifications and acceptance criteria, the sampling plan, results of the examinations and any related corrective actions taken.

    (b) Provide a description of water chemistry controls, summary of chemistry data, any design changes and/or actions taken, such as periodic flushing of recirculation procedures to maintain required water chemistry with respect to pH, B, CL^-, F^-, O_2.

(c) Describe the preservice NDE performed on the weld joints of identified systems. The description is to include the applicable ASME Code sections and supplements (addenda) that were followed, and the acceptance criterion.

(d) Facilities having previously experienced cracking in identified systems, Item 1, are requested to identify (list) the new materials utilized in repair or replacement on a system-by-system basis. If a report of this information and that requested above has been previously submitted to the NRC, please reference the specific report(s) in response to this Bulletin.

2. Facilities at which ISI examinations have not been performed (i.e., visual and volumetric UT) on stagnant portions of systems identified in Item 1 above, shall complete the following actions at the earliest practical date but not later than 90 days after the date of the Bulletin.

(a) Perform ASME Section XI visual examination (IWA 2210) of normally accessible* welds of all engineered safety systems at service pressure to verify system integrity.

(b) Conduct ultrasonic examination and liquid penetrant surface examination or a representative number of circumferential welds in normally accessible* portions of systems identified by 1 above. It is intended that the sample number of welds include all pipe diameters in the 2-1/2 inch to 24-inch range with no less than a 10 percent sample by system and pipe wall thickness. It is also intended that the U.T. examination cover the weld fusion zone and a minimum of 1/2-inch on each side of the weld at the pipe I.D. The examination shall be in accordance with the provisions of ASME Code Section XI-Appendix III and Supplements of the 1975 Winter Addenda except all signal responses shall be evaluated as to the nature of the indications. These code methods or alternative examination methods, combination of methods, or newly developed techniques may be used provided the procedures yield a demonstrated effectiveness in detecting stress corrosion cracking in austenitic stainless steel piping.

(c) If cracking is identified during Item (a) and (b) examinations, all welds of safety-related piping systems and associated subsystems where dynamic flow conditions do not exist during normal operations (Item 1) shall be subject to volumetric examination and repair including piping in areas which are normally inaccessible.

* Normally accessible refers to those areas of the plant which can be entered during reactor operation.

3. Identification of cracking in one unit of a multi-unit facility which causes safety-related systems to be inoperable shall require immediate examination of accessible portions of other similar units which have not been inspected under the ISI provisions of 10 CFR 50.55a(g) unless justification for continued operation is provided.

4. Any cracking identified shall be reported to the Director of the appropriate NRC Regional Office within 24 hours of identification followed by a 14 day written report.

5. Provide a written report to the Director of the appropriate NRC Regional Office within 30 days of the date of this Bulletin addressing the results of your review required by Item 1.

6. Complete the examination required by Item 2 within 90 days of the date of this Bulletin and provide a written report to the Director of the appropriate NRC Regional Office within 120 days of the date of this Bulletin describing the results of the inspections required by Item 2 and any corrective measures taken.

7. Copies of the reports required by Items 4, 5 and 6 above shall also be provided to the Director, Division of Operating Reactors, Office of Inspection and Enforcement, Washington, D.C. 20555.

Approved by GAO, B180225 (R0072), clearance expires 7/31/80. Approval was given under a blanket clearance specifically for identified generic problems.

Enclosures:
1. IE Circular 76-06
2. List of IE Bulletins
   Issued in 1979

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

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October 29 , 1979

IE Bulletin No. 79-17
Revision 1

PIPE CRACKS IN STAGNANT BORATED WATER SYSTEMS AT PWR PLANTS

Description of Circumstances:

IE Bulletin No. 79-17, issued July 26, 1979, provided information on the cracking R1
experienced to date in safety-related stainless steel piping systems at PWR R1
plants. Certain actions were required of all PWR facilities with an operating R1
license within a specified 90-day time frame. R1

After several discussions with licensee owner group representatives and inspection R1
agencies it has been determined that the requirements of Item 2, particularly R1
the ultrasonic examination, may be impractical because of unavailability of R1
qualified personnel in certain cases to complete the inspections within the time R1
specified by the Bulletin. To alleviate this situation and allow licensees the R1
resources of improved ultrasonic inspection capabilities, a time extension and R1
clarifications to the bulletin have been made. These are referenced to the R1
affected items of the original bulletin. R1

During the period of November 1974 to February 1977 a number of cracking incidents have been experienced in safety-related stainless steel piping systems and portions of systems which contain oxygenated, stagnant or essentially stagnant borated water. Metallurgical investigations revealed these cracks occurred in the weld heat affected zone of 8-inch to 10-inch type 304 material (schedule 10 and 40), initiating on the piping I.D. surface and propagating in either an intergranular or transgranular mode typical of Stress Corrosion Cracking. Analysis indicated the probable corrodents to be chloride and oxygen contamination in the affected systems. Plants affected up to this time were Arkansas Nuclear Unit 1, R. E. Ginna, H. B, Robinson Unit 2, Crystal River Unit 3, San Onofre Unit 1, and Surry Units 1 and 2. The NRC issued Circular No. 76-06 (copy enclosed) in view of the apparent generic nature of the problem.

During the refueling outage of Three Mile Island Unit 1 which began in February of this year, visual inspections disclosed five (5) through-wall cracks at welds in the spent fuel cooling system piping and one (1) at a weld in the decay heat removal system. These cracks were found as a result of local boric acid buildup and later confirmed by liquid penetrant tests. This initial identification of cracking was reported to the NRC in a Licensee Event Report (LER) dated May 16, 1979. A preliminary metallurgical analysis was performed by the licensee on a section of cracked and leaking weld joint from the spent fuel cooling system.

R1 - Identifies those additions or revision to IE Bulletin No. 79-17

The conclusion of this analysis was that cracking was due to Intergranular Stress Corrosion Cracking (IGSCC) originating on the pipe I.D. The cracking was localized to the heat affected zone where the type 304 stainless steel is sensitized (precipitated carbides) during welding. In addition to the main through-wall crack, incipient cracks were observed at several locations in the weld heat affected zone including the weld root fusion area where a miniscule lack of fusion had occurred. The stresses responsible for cracking are believed to be primarily residual welding stresses in as much as the calculated applied stresses were found to be less than code design limits. There is no conclusive evidence at this time to identify those aggressive chemical species which promoted this IGSCC attack. Further analytical efforts in this area and on other system welds are being pursued.

Based on the above analysis and visual leaks, the licensee initiated a broad based ultrasonic examination of potentially affected systems utilizing special techniques. The systems examined included the spent fuel, decay heat removal, makeup and purification, and reactor building spray systems which contain stagnant or intermittently stagnant, oxygenated boric acid environments. These systems range from 2 1/2-inch (HPCI) to 24-inch (borated water storage tank suction), are type 304 stainless steel, schedule 160 to schedule 40 thickness respectively. Results of these examinations were reported to the NRC on June 30, 1979 as an update to the May 16, 1979 LER. The ultrasonic inspection as of July 10, 1979 has identified 206 welds out of 946 inspected having UT indications characteristic of cracking randomly distributed throughout the aforementioned sizes (24"-14"-12"-10"-8"-2" etc.) of the above systems. It is important to note
that six of the crack indications were reportedly found in 2 1/2-inch diameter R1
pipe of the high pressure injection lines inside containment. These lines are
attached to the main coolant pipe and are nonisolable from the main coolant system
except for check valves. All of the six crack indications were found in two R1
high pressure injection lines containing stagnated borated water. No crack R1
indications were found in high pressure injection lines which were utilized for R1
makeup operations.

Recent data reported from Three Mile Island Unit 1 indicates that the extent R1
of IGSCC experienced in stainless steel piping at that facility may be more R1
limited than originally stated above. Of the 1902 total welds originally R1
inspected 350 contained U.T. indications which required further evaluation. R1
These 350 welds have been reinspected with a second U.T. procedure which pur- R1
portedly provides better discrimination between actual cracks and geometrical R1
reflectors. Hence, the Licensee now estimates that approximately 38 of the R1
350 welds contain IGSCC and the remaining welds, including those in high pressure R1
injection and decay heat lines, contain only geometrical reflectors. Further R1
metallurgical analysis of these welds is required to verify the adequacy of the R1
U.T. procedures and to determine the nature of the cracking. R1

For All Pressurized Water Reactor Facilities with an Operating License:

1. Conduct a review of safety related stainless steel piping systems within
30 days of the date of this Bulletin (July 26, 1979) to identify systems R1
and portions of systems which contain stagnant oxygenated borated water. These systems typically include ECCS, decay/residual heat removal, spent fuel pool cooling, containment spray and borated water storage tank (BWST-RWST) piping.

   For this review, the term "stagnant, oxygenated borated water systems" refers R1
to those systems serving as engineered safeguards having no normal operating R1
functions and contain essentially air saturated borated water where dynamic R1
flow conditions do not exist on a <u>continuous basis</u>. However, these systems R1
must be maintained ready for actuation during normal power operations. Where R1
your definition for stagnant differed from the one given above please supple- R1
ment your previous response within 30 days of this Bulletin revision. R1

   (a) Provide the extent and dates of the hydrotests, visual and volumetric examinations performed per 10 CFR 50.55a(g) (Re: IE Circular No. 76-06 enclosed) of identified systems. Include a description of the non-destructive examination procedures, procedure qualifications and acceptance criteria, the sampling plan, results of the examinations and any related corrective actions taken.

   (b) Provide a description of water chemistry controls, summary of chemistry data, any design changes and/or actions taken, such as periodic flushing or recirculation procedures to maintain required water chemistry with respect to pH, B, Cl^-, F^-, O_2.

   (c) Describe the preservice NDE performed on the weld joints of identified systems. The description is to include the applicable ASME Code sections and supplements (addenda) that were followed, and the acceptance criterion.

   (d) Facilities having previously experienced cracking in identified systems, Item 1, are requested to identify (list) the new materials utilized in repair or replacement on a system-by-system basis. If a report of this information and that requested above has been previously submitted to the NRC, please reference the specific report(s) in response to this Bulletin.

2. All operating PWR facilities shall complete the following inspection on the R1
stagnant piping systems identified in Item 1 at the earliest practical date R1
but not later than twelve months from the date of this bulletin revision. R1
Facilities which have been inspected in accordance with the original R1
Bulletin, Sections 2(a) and 2(b) satisfy the requirements of this Revision. R1

(a) Until the examination required by 2(b) is completed a visual examination R1
shall be made of all normally accessible welds of the engineered safety R1
systems at least monthly to verify continued systems integrity. Sim- R1
ilarly, the normally inaccessible welds, shall be visually examined R1
during each cold shutdown. R1

The relevant provisions of Article IWA 2000 of ASME Code Section XI R1
and Article 9 of Section V are considered appropriate and an acceptable R1
basis for this examination. For insulated piping, the examination may R1
be conducted without the removal of insulation. During the examination R1
particular attention shall be given to both insulated and noninsulated R1
piping for evidence of leakage and/or boric acid residues which may R1
have accumulated during the service period preceding the examination. R1
Where evidence of leakage and/or boric acid residues are detected at. R1
locations, other than those normally expected, (such as valve stems, R1
pump seals, etc.) the piping shall be cleaned (including insulation R1
removal) to the extent necessary to permit further evaluation of the R1
piping condition. In cases where piping conditions observed are not R1
sufficiently definitive, additional inspections (i.e., surface and/or R1
volumetric) shall be conducted in accordance with Item 2.(b). R1

(b) An ultrasonic examination shall be performed on* a representative sample R1
of circumferential welds in normally accessible portions of systems R1
identified by 1 above. It is intended that the sample number of welds R1
selected for examination include all pipe diameters within the 2 1/2- R1
inch to 24-inch range with no less than a 10 percent sampling being R1
taken. The approach to selection of the sample shall be based on the R1
following criteria: R1

(1) Pipe Material Chemistry - As a first consideration, those welds R1
in austenitic stainless steel piping (Types 304 and 316 ss) R1
having 0.05 to 0.08 wt. % carbon content based on available R1
material certification reports. R1

(2) Pipe Size and Thickness - An unbiased mixture of pipe diameters R1
and actual wall thickness distributed among both horizontal and R1
vertical piping runs shall be included in the sample. R1

(3) System Importance - The sample welds shall focus the examination R1
primarily on those systems required to function in the emergency R1
core cooling mode and secondly, on the containment spray system. R1

The U.T. examination sample may be focused on noninsulated piping R1
runs. The evaluation shall cover the weld root fusion zone and a R1
minimum of 1/2 inch on the pipe I.D. (counterbore area) on each side R1
of the weld. The procedure(s) for this examination shall be essentially R1

*Normally accessible refers to those areas of the plant which can be entered R1
during reactor operation. R1

in accordance with ASME Code Section XI, Appendix III and Supplements R1
of the 1975 Winter Addenda, except all signal responses shall be eval- R1
uated as to the nature of the reflectors. Other alternative examination R1
methods, combination of methods, or newly developed techniques may be R1
used provided the procedure(s) have a proven capability of detecting R1
stress corrosion cracking in austenitic stainless steel piping. R1

For welds of systems included in the sample having pipe wall thickness R1
of 0.250 inches and below, visual and liquid penetrant surface examina- R1
tion may be used in lieu of ultrasonic examination. R1

(c) If cracking is identified during Item 2(a) and 2(b) examinations, all R1
welds in the affected system, shall be subject to examination and repair R1
considerations. In addition, the sample welds to be examined on the. R1
remaining normally accessible noninsulated piping shall be increased to R1
25 percent using the criteria outlined in paragraph 2(b). In the event R1
that cracking is identified in other systems at this sampling level, R1
all accessible and inaccessible welds of the systems identified in R1
item 1 shall be subject to examination. R1

3. Identification of cracking in one unit of a multi-unit facility which causes safety-related systems to be inoperable shall require immediate examination of accessible portions of other similar units which have not been inspected under the ISI provisions of 10 CFR 50.55a(g) unless justification for continued operation is provided.

4. Any cracking identified shall be reported to the Director of the apppropriate NRC Regional Office within 24 hours of identification followed by a 14 day written report.

5. Provide a written report to the Director of the appropriate NRC Regional R1
Office within 30 days of the date of this bulletin revision addressing the R1
results of your review if required by Item 1. Provide a schedule of your R1
inspection plans in response to Item 2(b) in those cases in which the R1
inspections have not been completed. R1

6. Provide a written report to the Director of the appropriate NRC Regional R1
Office within 30 days of the date of completion of the examinations required R1
by Items 2(a), 2(b), or 2(c) describing the inspection results and any cor- R1
rective actions taken. R1

7. Copies of the reports required by Items above shall also be provided to the Director, Division of Operating Reactors, Office of Inspection and Enforcement, Washington, D.C. 20555.

Approved by GAO, B180225 (R0072), clearance expires 7/31/80. Approval was given under a blanket clearance specifically for identified generic problems.

Enclosures:
1. IE Circular No. 76-06
2. List of IE Bulletins Issued
in the Last Six Months

November 26, 1976
IE Circular No. 76-06

STRESS CORROSION CRACKS IN STAGNANT, LOW PRESSURE STAINLESS PIPING CONTAINING BORIC ACID SOLUTION AT PWR's

DESCRIPTION OF CIRCUMSTANCES:

During the period November 7, 1974 to November 1, 1975, several incidents of through-wall cracking have occurred in the 10-inch, schedule 10 type 304 stainless steel piping of the Reactor Building Spray and Decay Heat Removal Systems at Arkansas Nuclear Plant No. 1.

On October 7, 1976, Virginia Electric and Power also reported through-wall cracking in the 10-inch schedule 40 type 304 stainless discharge piping of the "A" recirculation spray heat exchanger at Surry Unit No. 2. A recent inspection of Unit 1 Containment Recirculation Spray Piping revealed cracking similar to Unit 2.

On October 8, 1976, another incident of similar cracking in 8-inch schedule 10 type 304 stainless piping of the Safety Injection Pump Suction Line at the Ginna facility was reported by the licensee.

Information received on the metallurgical analysis conducted to date indicates that the failures were the result of intergranular stress corrosion cracking that initiated on the inside of the piping. A commonality of factors observed associated with the corrosion mechanism were:

1. The cracks were adjacent to and propagated along weld zones of the thin-walled low pressure piping, not part of the reactor coolant system.

2. Cracking occurred in piping containing relatively stagnant boric acid solution not required for normal operating conditions.

3. Analysis of surface products at this time indicate a chloride ion interaction with oxide formation in the relatively stagnant boric acid solution as the probable corrodant, with the state of stress probably due to welding and/or fabrication.

The source of the chloride ion is not definitely known. However, at ANO-1 the chlorides and sulfide level observed in the surface tarnish film near welds is believed to have been introduced into the piping during testing of the sodium thiosulfate discharge valves, or valve leakage. Similarly, at Ginna the chlorides and potential oxygen availability were assumed to have been present since original construction of the borated water storage tank which is vented to atmosphere. Corrosion attack at Surry is attributed to in-leakage of chlorides through recirculation spray heat exchange tubing, allowing buildup of contaminated water in an otherwise normally dry spray piping.

ACTION TO BE TAKEN BY LICENSEE:

1. Provide a description of your program for assuring continued integrity of those safety-related piping systems which are not frequently flushed, or which contain nonflowing liquids. This program should include consideration of hydrostatic testing in accordance with ASME Code Section XI rules (1974 Edition) for all active systems required for safety injection and containment spray, including their recirculation modes, from source of water supply up to the second isolation valve of the primary system. Similar tests should be considered for other safety-related piping systems.

2. Your program should also consider volumetric examination of a representative number of circumferential pipe welds by non-destructive examination techniques. Such examinations should be performed generally in accordance with Appendix I of Section XI of the ASME Code, except that the examined area should cover a distance of approximately six (6) times the pipe wall thickness (but not less than 2 inches and need not exceed 8 inches) on each side of the weld. Supplementary examination techniques, such as radiography, should be used where necessary for evaluation or confirmation of ultrasonic indications resulting from such examination.

3. A report describing your program and schedule for these inspections should be submitted within 30 days after receipt of this Circular.

4. The NRC Regional Office should be informed within 24 hours, of any adverse findings resulting during nondestructive evaluation of the accessible piping welds identified above.

5. A summary report of the examinations and evaluation of results should be submitted within 60 days from the date of completion of proposed testing and examinations.

This summary report should also include a brief description of plant conditions, operating procedures or other activities which provide assurance that the effluent chemistry will maintain low levels of potential corrodants in such relatively stagnant regions within the piping.

Your responses should be submitted to the Director of this office, with a copy to the NRC Office of Inspection and Enforcement, Division of Reactor Inspection Programs, Washington, D.C. 20555.

Approval of NRC requirements for reports concerning possible generic problems has been obtained under 44 U.S.C 3152 from the U.S. General Accounting Office. (GAO Approval B-180255 (R0062), expires 7/31/77.)

SUMMARY OF NRC/IE REVIEW OF BULLETIN RESULTS

The following summary is based on the memorandum of April 1, 1981, for E. L. Jordan from W. J. Collins thru R. W. Woodruff, "Results of IE Bulletin 79-17, Revision 1". Background material already included in Appendix A is omitted from this summary.

1. "Except for the cracking found at San Onofre Unit 1 and ANO-1, which appears to be of a localized nature, no further evidence of a SCC problem was reportedly observed in the PWR plants beyond those previously identified in Table 1."

    There were two cracks at San Onofre Unit 1 and three cracks at Arkansas Unit 1. Table 1 is omitted from this summary.

2. "The chemistry of the contained fluid is periodically sampled on a daily, weekly or monthly basis depending on system function to assure that specific acceptance limits on element concentrations established by the NSSS and technical specifications for operation is maintained."

3. "In addition, the required ISI pump testing does serve to provide flushing and recirculation through portions of these systems at system pressure on a regular basis."

    The phrase "In addition" refers to Item 2 above.

4. "Based on the results of the Bulletin, the SCC experience in several operating PWR plants to date does not appear to be generically widespread. Despite this reassuring fact, a sufficient number of instances have occurred to emphasize the need for (1) continued control of secondary water chemistry with respect to trace impurities, particularly caustics, halogens and sulfide contaminants of fluids to minimize the potential for additional incidents in conjunction with (2) development of specific inservice inspection criteria to ensure that further cracking, should it occur, does not go undetected."

5. "In recognition of the latter need, the main ASME Code Committee on Section XI ISI rules has assigned a working group to develop ISI requirements for both BWR and PWR Class 2 piping systems including ECCS, RHR and CHRS, consistent with current 10 CFR 50.55(a) provisions. A final draft of the working group's proposed ISI rules is expected to be

presented to the respective code committees for Section XI rules consideration later this year. In view of the above measures being taken, no further action with respect to subject bulletin appears warranted and it is proposed it be closed out."

The phrase "the latter need" refers to Item 4 above.

APPENDIX B

Documentation of Bulletin Closeout

TABLE B.1 BULLETIN CLOSEOUT STATUS

Facility	Utility	Docket	Facility Status 07-26-79(1)	NRC Region	NSSS	Utility Response Date	Inspection Report and Date	Closeout Status(2)
Arkansas 1	AP&L	50-313	OL	IV	B&W	08-24-79 11-26-79(3) 01-21-80 02-04-80(4) 03-07-80(3) 10-16-80(3) 11-11-80	81-30(12-07-81)	Closed
Arkansas 2	AP&L	50-368	OL	IV	C-E	08-24-79 11-26-79(3) 03-07-80(3) 10-16-80(3) 11-11-80	81-29(12-07-81)	Closed
Beaver Valley 1	DLC	50-334	OL	I	W	08-28-79 10-29-79 08-18-80(3) 01-03-80	80-25(12-05-80) 82-08(05-04-82)	Closed
Calvert Cliffs 1	BG&E	50-317	OL	I	C-E	08-24-79 11-27-79(3)	82-03(02-24-82)	Closed
Calvert Cliffs 2	BG&E	50-318	OL	I	C-E	08-24-79 11-27-79(3)	82-03(02-24-82)	Closed
Cook 1	IMECO	50-315	OL	III	W	09-06-79 09-27-79 10-23-79 11-30-79(3)	82-22(02-14-83)	Closed
Cook 2	IMECO	50-316	OL	III	W	09-06-79 09-27-79 10-23-79 11-30-79(3)	82-22(02-14-83)	Closed
Crystal River 3	FPC	50-302	OL	II	B&W	08-27-79 10-03-79 11-29-79(3) 04-30-80	81-10(07-08-81)	Closed

Notes indicated by numbers within the parentheses are located at the end of the table.

TABLE B.1 BULLETIN CLOSEOUT STATUS (contd)

Facility	Utility	Docket	Facility Status 07-26-79(1)	NRC Region	NSSS	Utility Response Date	Inspection Report and Date	Closeout Status(2)
Davis-Besse 1	TECO	50-346	OL	III	B&W	08-24-79 11-28-79(3) 01-15-80(3) 04-29-80 06-27-80(3) 02-16-82	80-21(07-25-80)	Closed
Farley 1	APCO	50-348	OL	II	W	08-27-79 11-27-79(3) 02-27-80(3)	80-19(07-28-80)	Closed
Fort Calhoun 1	OPPD	50-285	OL	IV	C-E	08-24-79 08-31-79 11-28-79(3) 03-26-80(3)	79-17(11-15-79) (5)	Closed
Ginna	RG&E	50-244	OL	I	W	08-24-79 10-24-79 11-26-79(3) 04-25-80(3)	89-06(07-28-89)	Closed
Haddam Neck	CYAPCO	50-213	OL	I	W	08-24-79 11-28-79(3)	86-27(11-25-86)	Closed
Indian Point 2	ConEd	50-247	OL	I	W	08-24-79 11-30-79(3) 01-24-80(3) 04-21-80(3)	89-19(10-18-89)	Closed
Indian Point 3	PASNY	50-286	OL	I	W	08-23-79 10-24-79 02-06-80 06-01-84	80-13(03-10-81)	Closed
Kewaunee	WPS	50-305	OL	III	W	08-29-79 11-30-79(3) 07-25-80(3)	81-08(05-19-81)	Closed
Maine Yankee	MYAPCO	50-309	OL	I	C-E	08-22-79 10-05-79	80-19(02-08-81)	Closed

Notes indicated by numbers within the parentheses are located at the end of the table.

TABLE B.1 BULLETIN CLOSEOUT STATUS (contd)

Facility	Utility	Docket	Facility Status 07-26-79(1)	NRC Region	NSSS	Utility Response Date	Inspection Report and Date	Closeout Status(2)
Millstone 2	NNECO	50-336	OL	I	C-E	08-24-79 11-28-79(3)	80-19(10-27-80)	Closed
North Anna 1	VEPCO	50-338	OL	II	W	08-28-79 09-14-79 10-08-79 11-29-79(3) 05-15-80(3)	80-17(05-07-80)	Closed
Oconee 1	DUPCO	50-269	OL	II	B&W	08-30-79 11-28-79(3) 01-24-80 05-15-80	80-15(05-15-80) (6)	Closed
Oconee 2	DUPCO	50-270	OL	II	B&W	08-30-79 11-28-79(3) 02-18-80	80-02(02-25-80) (6)	Closed
Oconee 3	DUPCO	50-287	OL	II	B&W	08-30-79 11-28-79(3) 02-13-80 05-15-80	80-02(02-25-80) (6)	Closed
Palisades	CPC	50-255	OL	III	C-E	08-24-79 02-01-80 05-09-80	80-15(09-23-80)	Closed
Point Beach 1	WEPCO	50-266	OL	III	W	08-27-79 11-30-79(3)	80-17(12-04-80)	Closed
Point Beach 2	WEPCO	50-301	OL	III	W	08-27-79 11-30-79(3)	80-17(12-04-80)	Closed
Prairie Island 1	NSP	50-282	OL	III	W	(7)	81-07(04-15-81)	Closed
Prairie Island 2	NSP	50-306	OL	III	W	08-24-79 10-09-79 11-28-79(3) 01-11-80(3) (2 responses)	81-08(04-15-81)	Closed

Notes indicated by numbers within the parentheses are located at the end of the table.

TABLE B.1 BULLETIN CLOSEOUT STATUS (contd)

Facility	Utility	Docket	Facility Status 07-26-79(1)	NRC Region	NSSS	Utility Response Date	Inspection Report and Date	Closeout Status(2)
Rancho Seco 1	SMUD	50-312	OL	V	B&W	08-21-79 11-28-79(3) 03-28-80(3)	80-24(09-11-80)	Closed
Robinson 2	CP&L	50-261	OL	II	W	08-24-79 09-06-79 12-06-79(3)	80-13(07-14-80)	Closed
Salem 1	PSE&G	50-272	OL	I	W	08-24-79 11-26-79(3) 12-27-79(3) 07-14-83 12-04-84	87-37(01-21-88)	Closed
San Onofre 1	SCE	50-206	OL	V	W	08-24-79 09-11-79 11-21-79	80-31(11-26-80)	Closed
St. Lucie 1	FPL	50-335	OL	II	C-E	08-27-79 10-25-79 08-14-80	81-08(04-27-81)	Closed
Surry 1	VEPCO	50-280	OL	II	W	08-30-79 10-08-79 11-27-79(3) 12-27-79(3) 02-06-80 03-27-80(3)	80-35(10-15-80)	Closed (8)
Surry 2	VEPCO	50-281	OL	II	W	08-30-79 09-14-79 10-08-79 11-27-79(3) 12-27-79(3) 02-06-80 03-27-80(3)	80-38(10-15-80)	Closed (8)

Notes indicated by numbers within the parentheses are located at the end of the table.

TABLE B.1 BULLETIN CLOSEOUT STATUS (contd)

Facility	Utility	Docket	Facility Status 07-26-79(1)	NRC Region	NSSS	Utility Response Date	Inspection Report and Date	Closeout Status(2)
TMI 1	Met-Ed	50-289	OL	I	B&W	08-15-79 09-05-79 12-12-80	81-26(11-02-81)	Closed
Trojan	PGE	50-344	OL	V	W	08-24-79 11-21-79(3)	80-25(11-03-80)	Closed
Turkey Point 3	FPL	50-250	OL	II	W	08-27-79 10-25-79 11-30-79(3) 08-20-80	80-20(06-27-80)	Closed (9)
Turkey Point 4	FPL	50-251	OL	II	W	08-27-79 10-25-79 11-30-79(3) 08-20-80	80-12(06-27-80)	Closed (9)
Yankee-Rowe 1	YAECO	50-029	OL	I	W	08-24-79 11-27-79(3) 06-27-80 07-02-80 08-08-80(4) 08-26-80(3)	81-11(07-15-81)	Closed
Zion 1	CECO	50-295	OL	III	W	08-24-79 12-19-79(3)	80-01(03-03-80)	Closed
Zion 2	CECO	50-304	OL	III	W	08-24-79 12-19-79(3)	80-01(03-03-80)	Closed

Notes for Table B.1:

1. Facility status is based on Reference 1 (see page B-6). The following abbreviation applies to facility status: OL, Operating License.

2. See page B-6 for the Bulletin Closeout Criterion.

Notes for Table B.1 (contd)

3. Response to Revision 1 of the bulletin.

4. Licensee Event Report (LER).

5. Per the telephone calls of 4-26-89 and 4-27-89 between Thomas Westerman (Region IV) and NRC Headquarters, Inspection Report 79-17 is intended to close the bulletin for Fort Calhoun.

6. Per the telephone call of 08-01-89 between Kerry Landis (Region II) and NRC Headquarters, Inspection Report 80-15 closes IEB 79-17 for Oconee 1, and Inspection Report 80-02/02 closes the bulletin for Oconee 2,3.

7. Utility response dates for Prairie Island 1 are as follows: 08-24-79, 10-09-79, 11-28-79(3), 01-11-80(3), 02-04-80(3), 03-07-80(3), 04-11-80(3), 05-06-80(3), 06-09-80(3), 07-10-80(3), 08-14-80(3), and 10-30-80.

8. See the memorandum of March 30, 1981 on Surry 1,2 for M. Shymlock (Project Inspector) from N. Economos (Engineering Inspection Branch).

9. See the memorandum of October 27, 1980 on Turkey Point 3,4 for M. Shymlock (Project Inspector) from N. Economos (Engineering Inspection Branch).

CRITERION FOR CLOSEOUT OF BULLETIN

The utility response and an NRC/Region inspection report for the facility indicate that actions required by the bulletin and its revision (see pages A-2 and A-7) have been completed satisfactorily.

REFERENCES

1. United States Nuclear Regulatory Commission, Licensed Operating Reactors, Status Summary Report, Data as of 07-31-89, NUREG-0020, Volume 13, Number 8, August, 1989.

2. United States Nuclear Regulatory Commission, Code of Federal Regulations, Energy, Title 10, Chapter 1, January 1, 1987, cited as 10CFR 0.735-1.

APPENDIX C

Abbreviations

AEPSCO	American Electric Power Services Corporation
APCO	Alabama Power Company
AP&L	Arkansas Power and Light Company
ASME	American Society of Mechanical Engineers, The
BG&E	Baltimore Gas and Electric Company
B&W	Babcock and Wilcox Company
BWST	Borated Water Storage Tank
C-E	Combustion Engineering Incorporated
CECO	Commonwealth Edison Company
CFR	Code of Federal Regulations
CHRS	Containment Heat Removal System
ConEd	Consolidated Edison Company of New York, Inc.
CPC	Consumers Power Company
CP&L	Carolina Power and Light Company
CR	Contractor Report
CYAPCO	Connecticut Yankee Atomic Power Company
DLC	Duquesne Light Company
DUPCO	Duke Power Company
ECCS	Emergency Core Cooling System
FPC	Florida Power Corporation
FPL	Florida Power & Light Company
GAO	Government Accounting Office
GPUN	GPU Nuclear Corporation
HPCI	High Pressure Coolant Injection
HPSI	High Pressure Safety Injection
IE	(See NRC/IE)
IEB	Inspection and Enforcement Bulletin (NRC)
IGSCC	Intergranular Stress Corrosion Cracking
IMECO	Indiana and Michigan Electric Company
IR	Inspection Report (NRC/IE)
ISI	Inservice Inspection
LER	Licensee Event Report
LPSI	Low Pressure Safety Injection
Met-Ed	Metropolitan Edison Company
MYAPCO	Maine Yankee Atomic Power Company
NNECO	Northeast Nuclear Energy Company
NRC/IE	Nuclear Regulatory Commission/ Office of Inspection & Enforcement
NSP	Northern States Power Company
NSSS	Nuclear Steam Supply System
NU	Northeast Utilities
NYPA (PASNY)	New York Power Authority

OL	Operating License
OPPD	Omaha Public Power District
PASNY (NYPA)	Power Authority of the State of New York
PGE	Portland General Electric Company
PSE&G	Public Service Electric and Gas Company
PWR	Pressurized Water Reactor
R	Region (NRC)
RCS	Reactor Coolant System
RG&E	Rochester Gas and Electric Corporation
RHR	Residual Heat Removal
RWST	Refueling Water Storage Tank
SCC	Stress Corrosion Cracking
SCE	Southern California Edison Company
SI	Safety Injection
SMUD	Sacramento Municipal Utility District
TECO	Toledo Edison Company
TMI	Three Mile Island
TVA	Tennessee Valley Authority
UT	Ultrasonic Test
VEPCO	Virginia Electric and Power Company
W	Westinghouse Electric Corporation
WEPCO	Wisconsin Electric Power Company
WPS	Wisconsin Public Service Corporation
YAECO	Yankee Atomic Electric Company