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NUCLEAR REGULATORY COMMISSION
REGION IV
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ARLINGTON, TEXAS 76012

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July 26, 1979

Docket No. 50-285

Omaha Public Power District
ATTN: T. E. Short,
Assistant General Manager
1623 Harney Street
Omaha, Nebraska 68102

Gentlemen:

The enclosed IE Bulletin 79-17 is forwarded to you for action. A written response is required. If you desire additional information regarding this matter, please contact this office.

Sincerely,

James H. Sneyzik / acting
Karl V. Seyfrit
Director

Enclosures:

1. IE Bulletin No. 79-17
2. Listing of IE Bulletins
Issued in Last 12
Months

cc: R. L. Andrews, Manager
Fort Calhoun Station
Post Office Box 98
Fort Calhoun, Nebraska 68102

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

IE Bulletin No. 79-17
Date: July 26, 1979
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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 IE 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 i. 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 inasmuch 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.

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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 (HPSI) 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 79-111.

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⁻, O₂.

- (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 on a representative number of circumferential welds in normally accessible* portions of systems identified by Item 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 UT 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 be provided to the Director, Division of Operating Reactors, Office of Inspection and Enforcement, Washington, D. C. 20555.

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

November 24, 1976

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 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 No. 1 Containment Recirculation Spray Piping revealed cracking similar to Unit No. 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 lines 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 source 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 nondestructive 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).

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS

| Bulletin No. | Subject | Date Issued | Issued To |
|--------------|--|-------------|--|
| 78-11 | Examination of Mark I Containment Torus Welds | 7/21/78 | BWR Power Reactor Facilities for action: Peach Bottom 2 and 3, Quad Cities 1 and 2, Hatch 1, Monticello and Vermont Yankee |
| 78-12 | Atypical Weld Material in Reactor Pressure Vessel Welds | 9/26/78 | All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP) |
| 78-12A | Atypical Weld Material in Reactor Pressure Vessel Welds | 11/24/78 | All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP) |
| 78-12B | Atypical Weld Material in Reactor Pressure Vessel Welds | 3/19/79 | All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP) |
| 78-13 | Failures in Source Heads of Kay-Ray, Inc., Gauges Models 7050, 7050B, 7051, 7051B, 7060, 7060B, 7061 and 7061B | 10/27/78 | All General and Specific Licensees with Kay-Ray Gauges |

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| 78-14 | Deterioration of Buna-N Components in ASCO Solenoids | 12/19/78 | All GE BWR facilities with an Operating License (OL) or Construction Permit (CP) |
| 79-01 | Environmental Quali- fication of Class IE Equipment | 2/8/79 | All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP) |
| 79-01A | Environmental Qualification of Class IE Equipment | 6/6/79 | All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP) |
| 79-02 | Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts | 3/8/79 | All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP) |
| 79-02 (Rev. 1) | Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts | 6/21/79 | All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP) |
| 79-03 | Longitudinal Weld Defects In ASME SA-312 Type 304 Stainless Steel Pipe Spools Manufactured by Youngstown Welding and Engineering Company | 3/12/79 | All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP) |
| 79-04 | Incorrect Weights for Swing Check Valves Manufactured by Velan Engineering Corporation | 3/30/79 | All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP) |

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| 79-05 | Nuclear Incident at Three Mile Island | 4/1/79 | All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP) |
| 79-05A | Nuclear Incident at Three Mile Island | 4/5/79 | All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP) |
| 79-05B | Nuclear Incident at Three Mile Island | 4/21/79 | All B&W Power Reactor Facilities with an Operating License (OL) |
| 79-06 | Review of Operational Errors and System Misalignments Identified During The Three Mile Island Incident | 4/11/79 | All Pressurized Water Power Reactor Facilities Except B&W Facilities |
| 79-06A | Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident | 4/14/79 | All Westinghouse PWR Facilities with an Operating License (OL) |
| 79-06A (Rev. 1) | Review of Operational Errors and System Mis- alignments Identified During the Three Mile Island Incident | 4/18/79 | All Pressurized Water Power Reactor Facilities of Westinghouse Design with an Operating License (OL) |
| 79-06B | Review of Operational Errors and System Misalignments Identified During The Three Mile Island | 4/14/79 | All Combustion Engineer- ing PWR Facilities with an Operating License (OL) |
| 79-07 | Seismic Stress Analysis of Safety-Related Piping | 4/14/79 | All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP) |

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| 79-08 | Events Relevant to BWR Reactors Identified During Three Mile Island Incident | 4/14/79 | All BWR Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP) |
| 79-09 | Failures of GE Type AK-2 Circuit Breaker in Safety Related Systems | 5/11/79 | All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP) |
| 79-10 | Requalification Training Program Statistics | 5/11/79 | All Power Reactor Facilities with an Operating License (OL) |
| 79-11 | Faulty Overcurrent Trip Device in Circuit Breakers for Engineered Safety Systems | 5/22/79 | All Power Reactor Facilities with an Operating License (OL) or a Construction Permit (CP) |
| 79-12 | Short Period Scrams at BWR Facilities | 5/31/79 | All Power Reactor Facilities with an Operating License (OL) or a Construction Permit (CP) |
| 79-13 | Cracking In Feedwater System Piping | | All PWRs with an Operating License (OL) for action. All BWR with a Construction Permit (CP) for information |
| 79-14 | Seismic Analyses for As-Built Safety-Related Piping System | 7/2/79 | All Power Reactor facilities with an Operating License (OL) or a Construction Permit (CP) |
| 79-15 | Deep Drain Pump Deficiencies | 7/11/79 | All Power Reactor Facilities with a Construction Permit and/or Operating License (OL) |
| 79-16 | Vital Area Access Controls | 7/26/79 | All Power Reactors with an Operating License (OL) or anticipating fuel loading prior to January 1981. |