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UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
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ATLANTA, GEORGIA 30303

OCT 29 1979

In Reply Refer To:
RII:JPO
50-261

Carolina Power and Light Company
Attn: J. A. Jones, Senior Executive Vice President
and Chief Operating Officer
411 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

The enclosed Bulletin No. 79-17, Revision 1 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 P. O'Reilly
Director

Enclosure:
IE Bulletin No. 79-17,
Revision 1 w/encls.

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cc w/encl:
R. B. Starkey, Jr., Plant Manager
Post Office Box 790
Hartsville, South Carolina 29550

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 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. R1

   For this review, the term "stagnant, oxygenated borated water systems" refers to those systems serving as engineered safeguards having no normal operating functions and contain essentially air saturated borated water where dynamic flow conditions do not exist on a continuous basis. However, these systems must be maintained ready for actuation during normal power operations. Where your definition for stagnant differed from the one given above please supplement 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 stagnant piping systems identified in Item 1 at the earliest practical date but not later than twelve months from the date of this bulletin revision. Facilities which have been inspected in accordance with the original 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

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*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.)

LISTING OF IE BULLETINS
ISSUED IN LAST SIX MONTHS

Bulletin No.	Subject	Date Issued	Issued To
79-24	Frozen Lines	9/27/79	All Power Reactor Facilities with an OL or a CP
79-23	Potential Failure of Emergency Diesel Generator Field Exciter Transformer	9/12/79	All Power Reactor Facilities with an OL or a CP
79-22	Possible Leakage of Tubes of Tritium Gas Used in Timepieces for Luminosity	9/5/79	Each Licensee who Receives Tubes of Tritium Gas in Timepieces for Luminosity
79-21	Temperature Effects on Level Measurements	8/13/79	All PWR's with an Operating License
79-20	Packaging Low-Level Radioactive Waste for Transport and Burial	8/10/79	All Materials Licensees who did not receive Bulletin No. 79-19
79-19	Packaging Low-Level Radioactive Waste for Transport and Burial	8/10/79	All Power and Research Reactors with OLs, Fuel Facilities except uranium mills, and certain materials licensees
79-18	Audibility Problems Encountered on Evacuation of Personnel from High-Noise Areas	8/7/79	All OLs for Action All CPs for Information
79-17 (Rev. 1)	Pipe Cracks in Stagnant Borated Water Systems at PWR Plants	10/29/79	All PWRs with Operating License
79-17	Pipe Cracks in Stagnant Borated Water Systems at PWR Plants	7/26/79	All PWRs with Operating License
79-16	Vital Area Access Controls	7/26/79	All Holders of and applicants for Power Reactor Operating Licenses who Anticipage loading fuel prior to 1981
79-15 (Supp. 1)	Deep Draft Pump Deficiencies	7/18/79	All Power Reactor Licensees with a CP and/or OL

LISTING OF IE BULLETINS
ISSUED IN LAST SIX MONTHS

Bulletin No.	Subject	Date Issued	Issued To
79-15	Deep Draft Pump Deficiencies	7/11/79	All Power Reactor Licensees with a CP and/or OL
79-14 (Correction)	Seismic Analyses for As-Built Safety-Related Piping System	7/27/79	All Power Reactor Facilities with an OL or a CP
79-14 (Supp. 2)	Seismic Analyses for As-Built Safety-Related Piping System	9/7/79	All Power Reactor Facilities with an OL or a CP
79-14 (Rev. 1)	Seismic Analyses for As-Built Safety-Related Piping System	7/18/79	All Power Reactor Facilities with an OL or a CP
79-14	Seismic Analyses for As-Built Safety-Related Piping System	7/2/79	All Power Reactor Facilities with an OL or a CP
79-13 (Rev. 2)	Cracking in Feedwater System Piping	10/17/79	All PWR's with an Operating License
79-13 (Rev. 1)	Cracking in Feedwater System Piping	8/30/79	All PWR's with an Operating License
79-13	Cracking in Feedwater System Piping	6/25/79	All PWR's with an OL for action. All BWRs with a CP for information
79-12	Short Period Scrams at BWR Facilities	5/31/79	All GE BWR Facilities with an OL
79-11	Faulty Overcurrent Trip Device in Circuit Breakers for Engineered Safety Systems	5/22/79	All Power Reactor Facilities with an OL or a CP
79-10	Requalification Training Program Statistics	5/11/79	All Power Reactor Facilities with an OL
79-06C	Nuclear Incident at Three Mile Island - Supplement	7/26/79	To all PWR Power Reactor Facilities with an OL

LISTING OF IE BULLETINS
ISSUED IN LAST SIX MONTHS

Bulletin No.	Subject	Date Issued	Issued To
79-05C	Nuclear Incident at Three Mile Island - Supplement	7/26/79	To all PWR Power Reactor Facilities with an OL
79-02 (Rev. 1) (Supp. 1)	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	8/20/79	All Power Reactor Facilities with an OL or a CP
79-02 (Rev. 1)	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	6/21/79	All Power Reactor Facilities with an OL or a CP
79-01A	Environmental Qualification of Class 1E Equipment (Deficiencies in the Environmental Qualification of ASCO Solenoid Valves)	6/6/79	All Power Reactor Facilities with an OL or a CP