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

July 17, 1979

IE Information Notice No. 79-19

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 is 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.

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

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 INFORMATION NOTICES
ISSUED IN 1979

INFORMATION Notice No.	Subject	Date Issued	Issued To
79-01	Bergen-Paterson Hydraulic Shock and Sway Arrestor	2/2/79	All power reactor facilities with an OL or a CP
79-02	Attempted Extortion - Low Enriched Uranium	2/2/79	All Fuel Facilities
79-03	Limitorque Valve Geared Limit Switch Lubricant	2/9/79	All power reactor facilities with an OL or a CP
79-04	Degradation of Engineered Safety Features	2/16/79	All power reactor facilities with an OL or a CP
79-05	Use of Improper Materials in Safety-Related Components	3/21/79	All power reactor facilities with an OL or CP
79-06	Stress Analysis of Safety-Related Piping	3/23/79	All Holders of Reactor OL or CP
79-07	Rupture of Radwaste Tanks	3/26/79	All power reactor facilities with an OL or CP
79-08	Interconnection of Contaminated Systems with Service Air Systems Used As the Source of Breathing Air	3/28/79	All power reactor facilities with an OL and Pu Processing fuel facilities
79-09	Spill of Radioactively Contaminated Resin	3/30/79	All power reactor facilities with an OL
79-10	Nonconforming Pipe Support Struts	4/16/79	All power reactor facilities with a CP
79-11	Lower Reactor Vessel Head Insulation Support Problem	5/7/79	All holders of Reactor OLs and CPs

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ISSUED IN 1979

INFORMATION Notice No.	Subject	Date Issued	Issued To
79-12	Attempted Damage to New Fuel Assemblies	5/11/79	All Fuel Facilities Research Reactors, and Power Reactors with an OL or CP
79-13	Indication of Low Water Level in the Oyster Creek Reactor	5/29/79	All Holders of Reactor OLs and CPs
79-14	NRC Position of Electrical Cable Support Systems	6/11/79	All Power Reactor Facilities with a CP
79-15	Deficient Procedures	6/7/79	All Holders of Reactor OLs and CPs
79-16	Nuclear Incident at Three Mile Island	6/22/79	All Research Reactors and Test Reactors with OLs
79-17	Source Holder Assembly Damage From Misfit Between Assembly and Reactor Upper Grid Plate	6/20/79	All Holders of Reactor OLs and CPs
79-18	Skylab Reentry	7/5/79	All Holders of Reactor OLs