



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS  
WASHINGTON, D. C. 20555

February 8, 1975

Honorable William A. Anders  
Chairman  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555

Subject: REPORT ON CRACKING IN BOILING WATER REACTOR PIPING

Dear Mr. Anders:

At its 178th meeting, February 6-8, 1975, the Advisory Committee on Reactor Safeguards reviewed both the problem of cracks and leakage in stainless steel piping in several Boiling Water Reactors (BWRs) and the suggested programs to mitigate or to resolve the problem. During its review, the Committee had the benefit of discussions with representatives of the General Electric Company, the Regulatory Staff and their consultants and of the documents listed. On the basis of this review, the Committee offers the following comments:

1. Stress corrosion cracking (SCC) in sensitized stainless steel (SSS) is not a new problem

Austenitic stainless steels have a history of cracking or leaking due to stress-corrosion cracking (SCC), corrosion-fatigue, or to a combination of both static stress and fatigue when used in certain environments (1, 2, 3). The critical parameters are believed to be combined stresses; degrees of sensitization resulting from prior heating; specific materials with types 304 and 316 stainless steels particularly susceptible; and contaminants in the coolant, with levels of oxygen common to BWRs apparently being sufficient to cause intergranular stress-corrosion cracking (ISCC), providing stresses and levels of sensitization are sufficiently high. The Committee reported on a previous case of ISCC in a Report on the Nine Mile Point Nuclear Station dated June 16, 1970 (4).

The first incident in the current problem of cracking and leakage occurred during September 1974 at Dresden 2, where a leaking crack was detected in the heat-affected zone (HAZ) immediately adjacent to a weld in a 4-inch bypass loop connected to the 28-inch main recirculating line. Since the first observed leak, additional cracks or leaks have been detected in Type 304 stainless steel bypass lines at Quad Cities 2 (September and December), Dresden 2 (December), Quad Cities 1 (January 1975), Peach Bottom 3 (January), Monticello (January) and Millstone 1 (January). Additionally, leaks were detected in the 10-inch core spray lines adjacent to the safe ends in Dresden 2 during January (5 - 11).

The factors common to the cracks and leaks examined to date in the bypass loops appear to be high but unknown residual stress values resulting from welding, some sensitization of the HAZ due to the welding process, a level of oxygen equal to or greater than 0.2 ppm in the coolant and a low-amplitude medium-frequency vibration in the bypass loop. All bypass loop cracks were oriented circumferentially, initiated at the inner surface in the HAZ, and propagated intergranularly. Rates of propagation appear to be relatively high, based on the fact that nondestructive examinations (NDE) by ultrasonics revealed either no signals or signals within permissible limits at examinations in the September-November period, yet cracks or leaks were detected in the examined weldments two to three months later. Such behavior is typical of ISCC where the initiation period is usually long (months or years) followed by a rapid rate of propagation.

2. Actions taken by the Nuclear Regulatory Commission were both necessary and appropriate

The Nuclear Regulatory Commission in its Inspection and Enforcement Bulletin Number 75-01, dated January 30, 1975, required licensees with operating BWRs to complete within 20 days a volumetric inspection of all circumferential weldments in core spray loops through the second isolation valve. It also required them to conduct a system functional hydrostatic test on the core spray piping beyond the second isolation valve (downstream of the reactor vessel). A representative sampling of other pressure retaining welds in austenitic piping should be examined.

In earlier Regulatory Operations Bulletins 74-10, 74-10A, and 74-10B dated September 18, 1974, December 17, 1974 and January 24, 1975, the Commission requested that all accessible welds in bypass piping be examined by volumetric NDE and results evaluated. In addition, an expanded leak detection program should be initiated (12 - 15).

The Committee believes the NRC positions were appropriately conservative to establish the magnitude of the problem. However, establishment of the number and location of cracks is only the first stage.

3. Acceptable interim measures to repair the cracked pipes are available and should be implemented

An acceptable interim second stage consists of the repair of cracks or leaks by welding; by replacement with loop components of the same or similar alloys; or by the removal, if suitable, of the bypass loops.

4. Although cracks such as have occurred and such as may occur in the future in similar systems, or even in the modified systems, clearly increase the probability of a pipe failure, the consequent increase in risk to the public does not now appear to be substantial and is no cause for immediate alarm

During its review of SCC and possible solutions the Committee considered the relative safety aspects of failures in the bypass lines or in the core spray lines. The sudden failure of these lines, resulting in either a double-ended or single-ended pipe break, was reviewed in the context of LOCA-ECCS, applying Appendix K of 10 CFR Part 50. The Committee agrees that such breaks should not cause unacceptable high temperatures in the fuel. However, mechanisms such as SCC leading to an unpredictable degradation of components should be minimized so that the overall level of safety is not decreased significantly.

The possibility that cracks can be initiated in larger lines such as the main recirculating lines, and that these cracks could propagate rapidly, and result in sudden rupture, requires consideration. While it is not possible to quantify the probability of pipe break in large lines due to SCC, there is a substantial body of data relevant to nuclear and non-nuclear piping that indicates the probability of failure is markedly lower in large thick-walled piping compared to smaller piping. Although results from the recently required inspection of larger pipes are not yet available, there is evidence that levels of residual stress, degree of sensitization, and level of cyclic train amplitude all are less in large diameter thick-walled piping than in smaller piping for similar welding and design parameters.

5. Inspection and leak detection procedures are available and can be used to reduce the probability of a pipe rupture while a reactor is in operation

Prudence dictates that enhanced inservice inspection (12-16) be continued on larger pipes until there is a sufficiently complete understanding of stress corrosion cracking, and experience demonstrates a lack of SCC susceptibility of the large pipes. The ACRS believes it is appropriate to conduct the examinations required by NRC and to make appropriate repairs with currently used materials on an interim basis. The Committee suggests a continuation at an appropriate level of the expanded program of NDE on the austenitic stainless steel piping in the BWR system. The fuller utilization of leak detection and location methods should be considered, and the procedures to be followed in the event of unidentified leakage should be re-evaluated by the Regulatory Staff.

6. The interim measures now being proposed cannot be expected to eliminate the problem of SCC in SSS

The Committee urges that serious consideration be given to the third stage, namely, the minimization of SCC through a combination of modifications in design, material, fabrication, and possibly, coolant composition. While it is desirable to review and modify design and fabrication aspects, the Committee believes that particular attention should be given to materials not susceptible to SCC in the spectrum of environments encountered in LWRs. While the current problem is limited to BWRs, there has been SCC in PWRs in the steam generators and secondary systems. The programs suggested below can also contribute significantly to PWRs.

7. A longer range solution should be sought by programs of research and development

Specific examples where additional information would be desirable, several of which are being reviewed or where programs are underway, include such items as: (1) development of quantitative techniques for detecting and measuring residual stresses; (2) determination of the combined role of cyclic and static stresses on SCC; (3) response of piping to SCC as a function of degrees of sensitization; (4) assessment of the reliability and sensitivity of NDE techniques such as ultrasonic testing when applied to piping systems; (5) better methods for measuring and controlling both oxygen and other impurities in the coolant immediately adjacent to regions of potential failure; (6) role of internal surface finish on the initiation phase of SCC; and (7) selection of alloys not susceptible to SCC, which can be used to replace limited components in operating reactors (e.g., bypass loops), or which might be used for the piping systems in future reactors. Such alloys must meet ASME Section III Code requirements of design, mechanical properties, fabricability and weldability, and have a minimal effect on plant operability. Examples of the last requirement would include coupling of dissimilar metals, relative degree of "crud" formation of replacement alloys compared to current alloys, and possible mechanisms for controlling "crud" formation, to establish tradeoffs. Examples of alloys with substantially promise include the low carbon stainless steels (304L), stabilized stainless steels (347), low alloy steel (2-5 chromium), and austeno-ferritic alloys (17).

The Committee wishes to review the programs in the preceding paragraph when the ongoing studies reach a suitable stage.

Sincerely,

/s/ W. Kerr

W. Kerr  
Chairman

References attached

References

- 1) S. H. Bush and R. L. Dillon, "Stress Corrosion in Nuclear Systems", USAEC, Advisory Committee on Reactor Safeguards (March 8, 1973)
- 2) W. E. Berry, et al, "Stress Corrosion Cracking of Sensitized Stainless Steel," CORROSION - NACE, 29, No. 12 (December 1973)
- 3) C. F. Cheng, "Intergranular Stress-Assisted Corrosion Cracking of Austenitic Alloys in Water-Cooled Nuclear Reactors", J. Nucl. Mater., 5th Cong. on Corrosion (September 1973)
- 4) Letter, J. M. Hendrie (ACRS) to G. T. Seaborg (USAEC), "Report on Nine Mile Point Nuclear Station", (June 16, 1973)
- 5) "Statement of the Chairman of the Nuclear Regulatory Commission Before the Joint Committee on Atomic Energy" (February 5, 1975)
- 6) C. F. Cheng, "Failure Analysis for Cracked Type 304 Stainless Steel Piping in the 4-Inch Recirculation Bypass Lines of the Dresden-II and Quad Cities-II BWR Systems", Argonne National Laboratory (December 1974)
- 7) D. E. Delwiche, et al, "Recirculation Loop Four-Inch Bypass Line Cracks (Quad Cities 2 and Dresden 2)", NEDO-20669 (January 1975)
- 8) Letter, W. E. Berry (Battelle-Columbus) to D. D. Galle (Commonwealth Edison), On subject on constant strain rate stress-corrosion studies (November 26, 1974)
- 9) Office of Inspection and Enforcement (NRC-IE) Notification of An Incident or Occurrence No. 143, "Cracks in Core Spray Lines" (January 30, 1975)
- 10) NRC-IE Notification No. 142, "Cracks in 4" Recirculation Piping" (January 23, 1975)
- 11) USAEC-R0, Operating Experience Bulletin, "Cracks in Pipes at BWR Facilities" (January 15, 1975)
- 12) NRC-IE, Bulletin No. 75-01, "Through Wall Cracks in Core Spray Piping at Dresden 2" (January 30, 1975)
- 13) AEC-R0, Bulletin No. 74-10, "Failures in 4-Inch Bypass Piping at Dresden-2" (September 18, 1974)

References (continued)

- 14) AEC-RO, Bulletin No. 74-10A, "Failures in 4-Inch Bypass Piping at Dresden-2" (December 17, 1974)
- 15) NRC-IE, Bulletin No. 74-10B, "Failures in 4-Inch Bypass Piping at Dresden-2" (January 24, 1975)
- 16) NRC, "Program for Investigation of Cracks in the Bypass Line of the Primary Cooling System of Boiling Water Reactors" (January 9, 1975)
- 17) Letter, E. L. Zebroski (EPRI) to R. F. Fraley (NRC-ACRS), "Information Submitted by the Electric Power Research Institute for ACRS meeting on BWR Pipe Cracking, February 6, 1975" (February 5, 1975)