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PIPE BREAK TESTING OF PRIMARY LOOP PIPING SIMILAR TO DEPARTMENT OF ENERGY'S NEW PRODUCTION REACTOR-HEAVY WATER REACTOR

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INTRODUCTION

The United States Department of Energy (DOE) was designing a new generation of production reactors to replace the aging production reactors at the Savannah River Site (SRS) in Aiken, South Carolina. A heavy water reactor was one of the options being pursued during the Title I design phase. This program was continued through the design phase; however, a decision was made not to construct a new production reactor.

The increases in understanding and the vast experience gained from existing production reactors and commercial nuclear power plants led to proposed improvements in several areas of the proposed design for the New Production Reactor-Heavy Water Reactor (NPR-HWR). One such area of improvement was to show that the tough austenitic piping of the NPR-HWR at the relatively moderate temperatures and pressures could provide a design where an instantaneous DEGB was highly improbable.

The Pipe Impact Test Facility (PITF) at ORNL was designed to load pipes in bending. The maximum bending stress is located at the center of the pipe. The pipe is loaded in four-point bending. This loading condition provides a constant bending stress across the three-ft (914.4 mm) long center section of the pipe. The flaw being studied is at a weld in the exact center of the constant bending section of the pipe. Static slow oscillary cyclic (SOC) and dynamic loadings have been used to produce bending in the pipe. This bending load tends to open the flaw in the pipe. Therefore, the PITF can be used to show crack stability in tough austenitic piping when being loaded at safe-shutdown earthquake (SSE) load levels and above.

BACKGROUND

The testing concept and discussion of initial testing on a 316LN stainless steel pipe were provided in a previous ORNL report [1], and at the American Society of Mechanical Engineers (ASME) 1992 pressure vessels and piping conference [2]. Results of that testing were reviewed by a special Piping Integrity Review Group (PIRG) established by DOE. The initial meeting of this group was held in August 1992 and the last meeting in December 1992.

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DEGRADED PIPING MATERIAL TESTS

The subject of this paper is to review the recent failure testing of the Savannah River C-reactor piping weldment, which will be referred to as the C-pipe in the remainder of the paper. The intent of this paper is to further familiarize the technical community with Oak Ridge National Laboratory's (ORNL) pipe test program and associated activities surrounding the C-pipe test (as conducted on behalf of the DOE-NPR Program).

The C-pipe is a 16 in. schedule-40 type-304 stainless steel (SS304) pipe which saw service in the Savannah River Plant (SRP) C-reactor for 27 years (1957-1984). The two-ft pipe piece which contains the subject butt weld was selected because of its availability, because it could be decontaminated to levels that allow its handling under U.S. Department of Energy (DOE) regulations, and because it appeared from outside-surface in-service inspection (ISI) by ultrasonic methods (UT) to suffer from inside-surface stress corrosion cracking (SCC) in sensitized material near the weld.

After decontamination, additional detailed inspections were performed in July 1992. These inspections included dye penetrant examination of the pipe weld inside surface and a full volumetric ultrasonic (UT) examination of the weld completed by Westinghouse Savannah River Company (WSRC) K-reactor ISI team members. The dye penetrant examination showed that a lack-of-penetration ran circumferentially in a dashed/broken pattern at the inside center of the weld for much of the pipe's circumference. This lack of penetration was estimated to be approximately .030 in. (0.76 mm) deep. The UT examination revealed that the called IGSCC crack locations were most likely the UT equipment sensing grinding marks on the inside surface of the pipe.

The C-pipe test was commenced on August 18, 1992, 118K cycles of fatigue (at $1/2 S_y$) were applied to break through-wall and create a leak before break (LBB) crack. The surface crack length was two-in. (50.8 mm).

Large Load Cycling Test. At normal operating conditions and stress ($1/5 S_y$) leakage was 2 gpm (0.126 l/s). The referenced two-in. (50.8 mm) crack was then loaded slowly in fully reversed bending cycles (SOC) at the ram maximum load of 100 kips (445 kN) as previously discussed. This loading provided a bending moment in the pipe center section of 400 ft-kips (542 kN-m) which is a factor of 3.6 times larger than the previous testing [1] completed on the SS316L piping. For this load, three strain gages located 180° away from the crack developed on the average over 0.2 percent strain (with peak strains approaching 0.3%).

After ten cycles of loading, the surface crack length was measured at 3.0 in (76 mm). The crack length at the end of 20 cycles was established at 3.7 in. (94 mm); and at 30 cycles was 4.8 in. (122 mm). At the end of 30 cycles the top of the crack had twisted over to the west weld fusion line, while the bottom of the crack tip had twisted over to the east weld fusion line. From this position, the remaining 10 cycles caused the crack to tear essentially along the weld fusion line. Tearing was essentially evenly incremental at 0.3 in./cycle (7.6 mm/cycle) through cycle 33. The incremental tearing then increased to 0.7 in./cycle (18 mm/cycle) through cycle 37. On the remaining load cycles, tearing was as listed below:

<u>Cycle</u>	<u>Crack Growth</u>
38	1.2 in. (30.5 mm)
39	1.5 in. (38.1 mm)
40	5.0 in. (127.0 mm)

It is important to note that the crack was between 10 and 11 inches (254-279 mm) when, in cycle 40, the 100 kip (445 kN) load could no longer be supported. Crack extension rates during

the tearing instability were of modest value on the order of two-in./sec (51 mm/sec). Further, it is noted that the frequency of loading programmed into the controller was 1/60 Hz. Sketches of the crack length dimensions are on Fig. 1.

Prior to completion of the ORNL high load cycling of the C-pipe, Battelle completed sensitivity studies of the effect of crack length on the pipe loading [3]. This was done for the ORNL configuration (circumferentially through-wall cracked pipes under four-point bending) using NRC-Pipe Code version 1.4 and using average material properties of SRP SS304 at 125°C, as provided by SR [4]. The LBB.ENG2 J-estimation scheme option was used to obtain predictions of load versus load-line displacement. The base metal stress-strain curve was represented by WSRC and Battelle as a Ramberg-Osgood curve with yield = 29 ksi (200 MPa), $\epsilon_0 = 0.001$, $N = 5$, and $\alpha = 2$. The Battelle estimate of load versus load-line displacement for weld metal [3] correlates well with the ORNL measured values.

The hydraulic ram loading fixture used in the C-pipe test is shown in Fig. 2. The C-pipe crack after 40 load cycles to high loads is shown in Fig. 3. The crack length at the end of 40 cycles of large load was 16.5 in. (419 mm), or the crack grew a total of 14.5 in. (368 mm) during the test. At this crack length the pipe can support a load of approximately 85 kips (378 kN) without additional tearing. That load represents a factor of 3.1 times the loads anticipated for the New Production Reactor's (NPR's) safe-shutdown earthquake (SSE).

PIRG Comments

Forward. The Piping Integrity Review Group (PIRG) was established to provide a technical review of tests on wrought austenitic stainless steel piping subjected to repetitive loads simulating sixty years of operation of the heavy-water New Production Reactor. This end-of-life condition was followed by severe loads simulating greater than SSE seismic loading. The following is taken from the second and last PIRG report, with discussion limited to one experiment (C-Pipe) and supporting analytic studies. The following PIRG members participated actively in the development of this report:

Spencer H. Bush, Chairman; Robert J. Bosnak; Macintyre R. Louthan; Everett C. Rodabaugh; Gery M. Wilkowski. In addition, Robert P. Kennedy provided written comments, but did not attend the review and discussion sessions.

Introduction. The second PIRG report addresses the comments, conclusions and recommendations of PIRG related to the C-pipe test and the analytic studies supporting or expanding on this test. The first report addressed scope issues in the PIRG Charter.

Analytic Programs. PIRG compliments Battelle Memorial Institute (BMI) on an excellent, albeit preliminary, analytic study simulating both a modeling of crack growth and realistic dynamic fracture assessment of a typical NPR-HWR primary pipe system. Other experiments and accompanying analyses have confirmed that the analytic technique used can predict observed experimental behavior with an acceptable degree of accuracy. These studies confirmed which welds would see the highest loading conditions. As expected, they were in the welds attaching elbows to the reactor pressure vessel nozzles and the heat exchanger nozzles.

Restrictions Related to Conclusion One Below. Conclusion is based on a case where fatigue is the mechanism leading to cracking and the following restrictions also apply:

- Material is wrought austenitic stainless steel;
- Pipe is schedule 40 or no less than 0.5 in. (13 mm) wall for larger diameter piping;

- New materials selected essentially are immune to IGSCC;
- Reactors considered are essentially free from severe water hammer;
- Primary system temperatures not exceeding ($\sim 100^{\circ}\text{C}$) and pressures (<1.72 MPa); therefore, thermal fatigue should not be an issue;
- Massive components such as pumps or heat exchangers are anchored so that a severe seismic event will not lead to gross movement of the component and overload failure of the pipe;
- For newer reactors ASME Section III Division 1 is used; for planned reactors the 1992 or later edition of the Code is used, provided that later editions are comparable to the 1992 edition;
- An Inservice Inspection program basically complying with IWB of Section XI (1992 edition) is in place;
- Technical Specifications define a leak detection system; this may be more difficult for low pressure light-water reactors;

PIRG Conclusions/Recommendations

Provided the restriction cited in the RESTRICTIONS RELATED TO CONCLUSION ONE section are in force, then the following conclusions are applicable:

- (1) For DOE reactors at low pressure (≤ 1.72 MPa) and temperatures not exceeding ($\sim 100^{\circ}\text{C}$) with austenitic stainless steel piping, the DEGB should not be a DBE;
 - (2) For DOE low pressure low temperature reactors subject to IGSCC the analytic results with a 360° deep crack indicate that an instantaneous DEGB is highly improbable; PIRG believes further analyses would confirm that the instantaneous DEGB need not be a DBE used to evaluate core flow instability;
- For ALWRs, information is inadequate to justify relaxation of the current DEGB requirements other than that provided by 10 CFR 50 GDC-4 and 10 CFR 50.46 regarding leak-before-break;

With regard to recommendations PIRG suggests that further analytic studies could lead to relaxation of the instantaneous DEGB requirement as a DBE when the crack mechanism is IGSCC.

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Fig. 1. - SKETCHES OF THE OPENING CRACK WERE MADE AT ZERO LOAD AT COMPLETION OF VARIOUS CYCLES.

Fig. 2. - THE ORNL TEST FIXTURE IS SHOWN HERE.

Fig. 3. - THE "C" PIPE IS SHOWN HERE AFTER 40 CYCLES OF BENDING AT +/- 400 ft-kips MOMENTS.