# BWR OWNERS' GROUP

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Project No. 691

BWROG-02042
April 25, 2002

Mr. Joseph Donoghue
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U.S. Nuclear Regulatory Commission
Washington DC 20555-0001.

SUBJECT: **A REVIEW OF NUREG/CR-5750 IGSCC IMPROVEMENT FACTOR AND PROBABILITY OF RUPTURE GIVEN A THROUGH-WALL CRACK**

The NRC Staff is pursuing Commission directive to risk informed 10CFR50 via option 3 rulemaking. In this process, the Staff has been gathering information relative to loss-of-coolant accident (LOCA) frequencies and conducting an internal review of Appendix J of NUREG/CR-5750 provides estimated frequencies for LOCAs in U.S. commercial nuclear power plants.

The subject review was conducted by the BWROG to provide current information on IGSCC improvement factors and probability of piping rupture. This review is documented in the attached report.

Appendix J of NUREG/CR-5750 LOCAs frequency estimates were made for boiling water reactors (BWRs) using available operating experience data at the time of publication. The scope of data included information on corrosion mechanisms acting on primary pressure boundary piping, and information on fracture mechanics analyses. Frequency estimates covered small break LOCA (SBLOCA), medium break LOCA (MBLOCA), and large break LOCA (LBLOCA). Appendix J identified a factor of 20 improvement to account for the IGSCC mitigation measures implemented in the BWR plants during the 1990s. Based on current field data, this factor is judged to be conservative. A more current improvement factor of 33 is implied by the field data.

This report being submitted to you for information in the rulemaking efforts. If desired by the Staff, the BWROG will provide a presentation of this information at our mutual convenience. For more information, please contact Rick Hill (GE) at (408)-925-5388 or at Richard.hill@gene.ge.com.

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Sincerely,

J. A. Gray, Jr.

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*April 2002*

# A REVIEW OF NUREG/CR-5750 IGSCC IMPROVEMENT FACTOR AND PROBABILITY OF RUPTURE GIVEN A THROUGH-WALL CRACK

**April 2002**

**Prepared for**

**BWR Owners' Group**

**Prepared by**

**GE Nuclear Energy**

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# A REVIEW OF NUREG/CR-5750 IGSCC IMPROVEMENT FACTOR AND PROBABILITY OF RUPTURE GIVEN A THROUGH-WALL CRACK

**April 2002**

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## Important Notice Regarding the Contents of this Report
## Please Read Carefully

## Background

Appendix J of NUREG/CR-5750 (Reference 1) provides estimated frequencies for loss-of-coolant accidents (LOCAs) in U.S. commercial nuclear power plants. Estimates are made for both pressurized water reactors (PWRs) and boiling water reactors (BWRs) using available operating experience data, information on corrosion mechanisms acting on primary pressure boundary piping, and information on fracture mechanics analyses. Frequency estimates covered small break LOCA (SBLOCA), medium break LOCA (MBLOCA), and large break LOCA (LBLOCA). For the BWRs, medium break was defined as the break in liquid piping with inside diameter in the range of 1-inch to 5-inches. For the BWR steam piping, the medium range was from 4-inches to 5-inches. Break in larger than 5-inch diameter BWR piping was defined as large break. The estimates for LBLOCA and MBLOCA frequencies were calculated using the following equation:

$$\text{LOCA Frequency} = (F_{TW})(P_{R/TW})(IGSCC_{BWR\text{-}only})$$

where,

$F_{TW}$ = Frequency of through-wall (TW) cracks in primary (un-isolatable) Piping
= (Number of through-wall cracks/number of reactor calendar years of operating experience)

$P_{R/TW}$ = Mean probability of rupture given a through-wall crack
= 2.5/(nominal pipe diameter in mm), for pipe diameters from 1 to 10 inches (from 25 to 250 mm)
= 0.01, for pipe diameters greater than 10 inches (250 mm)

IGSCC = IGSCC improvement factor, for BWRs only = 1/20=0.05

The total U.S. BWR operating experience from 1969 through 1997 was used to estimate the MBLOCA and LBLOCA frequencies for through-wall cracking. A review of the IGSCC improvement factor and the assumed $P_{R/TW}$ for large diameter pipes is the subject of this interim report.

## IGSCC Improvement Factor

Reference 1 used a factor of 20 improvement to account for the IGSCC mitigation measures implemented in the BWR plants during the 1990s. It was judged to be conservative with respect to the improvement factor of 33 implied by the field data considered in Reference 1. The relevant statistical test results from several GE reports are reviewed here to provide qualitative technical basis to assume the improvement factor to be 33 or higher. It should be noted that the original factor of 20 was an improvement criterion, not the result of testing. Actual tests had to show FOI of > 20 to be considered a successful mitigation.

Figure 1 shows the test set-up used in the pipe test laboratory (PTL) to conduct the factor of improvement (FOI) tests at GE. All of the tests were conducted on 4-inch diameter pipes at a temperature of 550°F. The test conditions were quite aggressive (not typical of plant conditions) to shorten the test time to a reasonable value (less than 10,000 hours): high loads, cycling and 8 ppm oxygen water environment. A statistical approach was used to characterize the FOI. References 2 through 6 report on the factors of improvement obtained for various IGSCC mitigation remedies. The following summarizes the key FOI results pertaining to various IGSCC mitigation measures:

- Reference 2 reports on the FOI obtained for the alternate materials such as low carbon and nuclear grade stainless steels. The tests were conducted in 8 ppm oxygen environment. Figure 2 shows the FOI obtained from various heats of nuclear grade Type 304 and 316 stainless steels. It is seen that the FOI are in excess of 30 and reach a value up to 70 for some of the heats. It should be noted that the lower FOIs for some heats did not mean that there was IGSCC failure. It was simply test termination for specimen examination.
- FOI results for solution heat treatment (SHT) after welding, solution heat-treated corrosion resistant cladding (shop CRC), as-deposited corrosion resistant cladding (field CRC) and heat sink welding (HSW) are reported in Reference 3. The FOI ranged from a low of ≥ 7 to a high of ≥ 89. The ≥ sign is used to indicate that the test was stopped before failure due to time constraints. Some of the low FOI result from the fact that the accelerated loading (up to 136% of 550°F yield strength of the material) tended to eliminate the beneficial residual stresses such as from HSW.
- Reference 4 provides the test details and results for induction heating stress improvement (IHSI) treated pipes. The applied loading resulted in stresses greater than yield strength of the material. Although none of the IHSI treated welds had failed during the test, the statistical methodology predicted a FOI of ≥ 11.5. Had the test been continued, the FOI would have been higher.
- Reference 5 reported on the FOI tests conducted for hydrogen water chemistry (HWC) environment. The FOI were determined by comparing failure hours in HWC environment (20 to 50 ppb oxygen) versus the normal BWR water (200 ppb oxygen). None of the HWC environment pipes had failed during the test interval. FOI of ≥ 25 were reported.
- Reference 6 reported FOI > 7.5 for pipes treated with last pass heat sink welding (LPHSW).

Since only the hours for leak type failure are reported, it is not clear what fraction of that time represents crack growth phase. This distinction is important since most of the BWR plants currently have either HWC or noble metal chemical addition (NMCA) plus low level HWC, that tends to significantly reduce the crack growth rate. Currently, the NRC conservatively allows a factor of 2 credit for crack growth whereas the test data support much higher credit. There are no comparable PTL test data available to quantify this effect in terms of FOI calculation methodology. With the current IGSCC mitigation measures in place at most BWR plants, one can expect a FOI considerably higher than 20, the value assumed in Reference 1. A FOI value of at least 33, as supported by the field data in Reference 1, can be used in the LOCA frequency analyses. Even this value may be conservative since it is based on the data from 1989 through 1997 and many more reactor years since then have been accumulated without any reported IGSCC leak incidents.

Based on the preceding discussion, it is concluded that a FOI of at least 33 may be used in the LOCA analyses.

## Mean Probability of Rupture Given a Through-wall Crack

For pipes 10-inch diameter and greater, Reference 1 assumes a frequency estimate of $1 \times 10^{-2}$ for mean probability of rupture given a through-wall crack ($P_{R/TW}$). This estimate is quite conservative by several orders of magnitude. This is supported by a review of the results for leak and break probabilities presented in References 7 through 9.

Figure 3 from Reference 7 presents the results for the calculated values of leak and break 40-year cumulative probabilities as a function of leak detection threshold for a 6-inch diameter pipe. The authors of Reference 7 used pc-PRAISE computer code to calculate the probabilities. The data at a leak detection threshold of less than 5 gpm are important for this discussion for the following reasons.

Most of the BWRs can detect 1 gpm change in the unidentified leakage and technical specifications typically require plant shutdown when the unidentified leakage exceeds 5 gpm. In the creviced safe end leakage incident of 1978 at the Duane Arnold plant, the plant was shut down when the unidentified leakage rate reached 3 gpm. Therefore, a 5 gpm leak detection threshold is a reasonable value for most BWRs.

Figure 3 does not show a cumulative probability curve for 5 gpm leak case. However, from looking at the trend of the 300 gpm and 30 gpm leak curves, it is reasonable to assume that the 5 gpm leak curve would be very close to the upper solid line labeled 'Leak Probability'. In that case, there is almost four orders of magnitude difference between the leak and break probabilities. That means a $P_{R/TW}$ value of $1 \times 10^{-4}$. The differences between the leak and break cumulative probabilities for a 28-inch diameter pipe are even higher than those for the 6-inch pipe case.

Reference 7 considered only a fatigue mechanism and did not explicitly consider IGSCC. Currently no pc-PRAISE calculations are available in which a piping system with IGSCC remedial measures have been implemented. Nevertheless, it is judged that the $P_{R/TW}$ can be conservatively estimated as $1x10^{-3}$ if the calculations were conducted where IGSCC and the impact of remedial measures were considered.

Reference 8 also gave lower probability for $P_{R/TW}$ compared to the assumed value of 0.01 assumed in Reference 1 for pipe diameters greater than 6-inches in diameter. It is judged that Reference 8 value would have been even lower had it considered IGSCC mitigation measures such as HWC and the impact of in-service inspections.

Reference 9 showed a wide variation from $5x10^{-2}$ to $5x10^{-7}$ in the conditional failure probability given a leak of 3 to 5 gpm. However, the analysis implicitly assumed a probability of 1.0 for the occurrence of a safe shutdown earthquake (SSE) during the time the leak remains undetected. Generally, the probability of occurrence of a SSE event during any specific interval of few days or weeks is several orders of magnitude lower.

Based on the preceding discussion, it is concluded that the $P_{R/TW}$ for medium and large pipes is equal to or less than $1x10^{-3}$.

## Summary

A review of the IGSCC improvement factor value and the assumed value of $P_{R/TW}$ for large diameter pipes in Reference 1, is the subject of this interim report. Based on this review it is concluded that:

- FOI of at least 33 for IGSCC mitigation measures is justified in the medium and large break LOCA analyses
- The $P_{R/TW}$ value for the medium and large pipes is equal to or less than $1x10^{-3}$.

## References

[1] "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995," NUREG/CR-5750, February 1999.

[2] "Alternate Alloys for BWR Pipe Applications," Report prepared by General Electric for EPRI, Report No. NP-2671-LD, October 1982.

[3] "Qualification of Solution Heat Treatment, Corrosion resistant Cladding and Heat Sink Welding," Paper No. 17 in Volume 1 of Proceedings: Seminar on Countermeasures for Pipe Cracking in BWRs, EPRI Report No. WS-79-174, May 1980.

[4] "Induction Heating Stress Improvement for Stainless Steel Piping," GE Report No. NEDE-25394 (Proprietary), July 1981.

[5] "Pilgrim Nuclear Power Station Recirculation Nozzle Repair Program and Hydrogen Water Chemistry Material Qualification," GE Report No. NEDO-30730, September 1984.

[6] "Last-Pass Heat Sink Welding," Report prepared by General Electric for EPRI, Report No. NP-3479-LD, March 1984.

[7] Simonen, F.A., D.O. Harris and D.D. Dedhia, "Effect of Leak Detection on Piping Failure probabilities," ASME Pressure Vessels and Piping Conference, San Diego, CA July 1998, PVP-373, pp 105-113.

[8] "Probability of Failure in BWR Reactor Coolant Piping," NUREG/CR-4792, Volume 1, March 1989.

[9] Rahman, S., et al., "Pipe Fracture Evaluations for Leak-Rate Detection: Applications to BWR and PWR Piping," ASME PVP Volume 266 (1993).

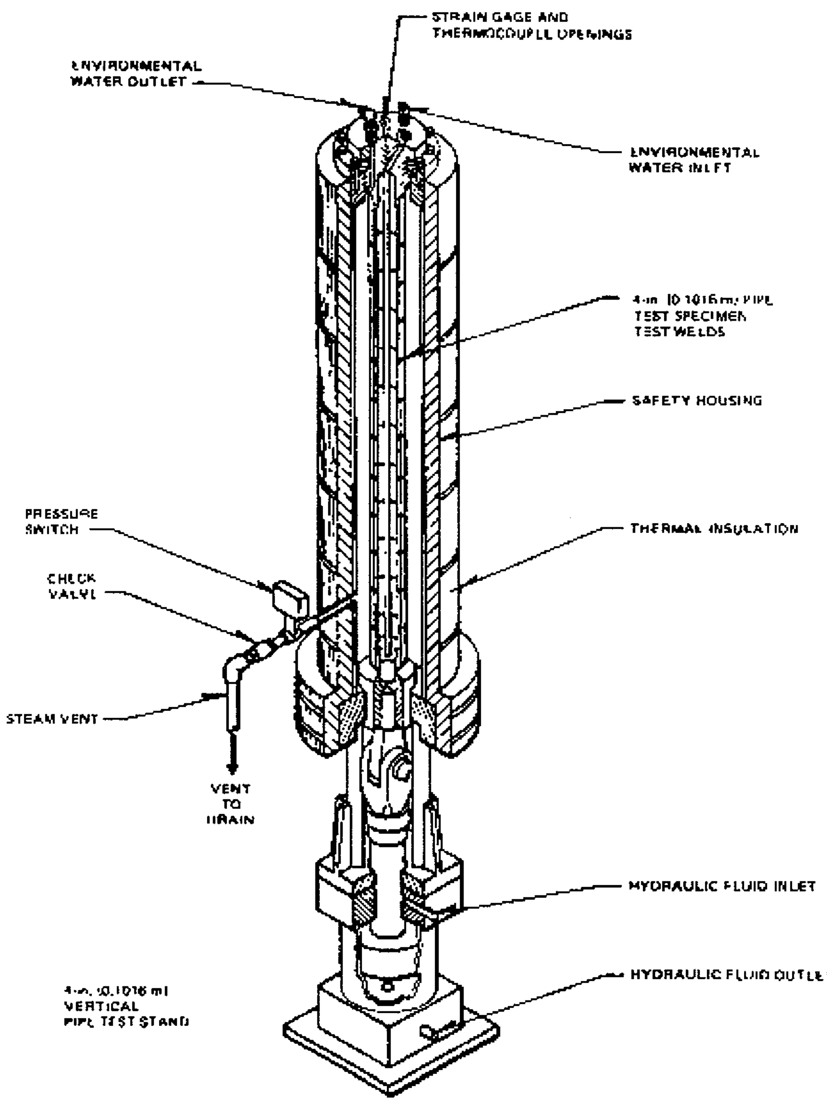


**Figure 1 Pipe Test Specimen Loading Stand**

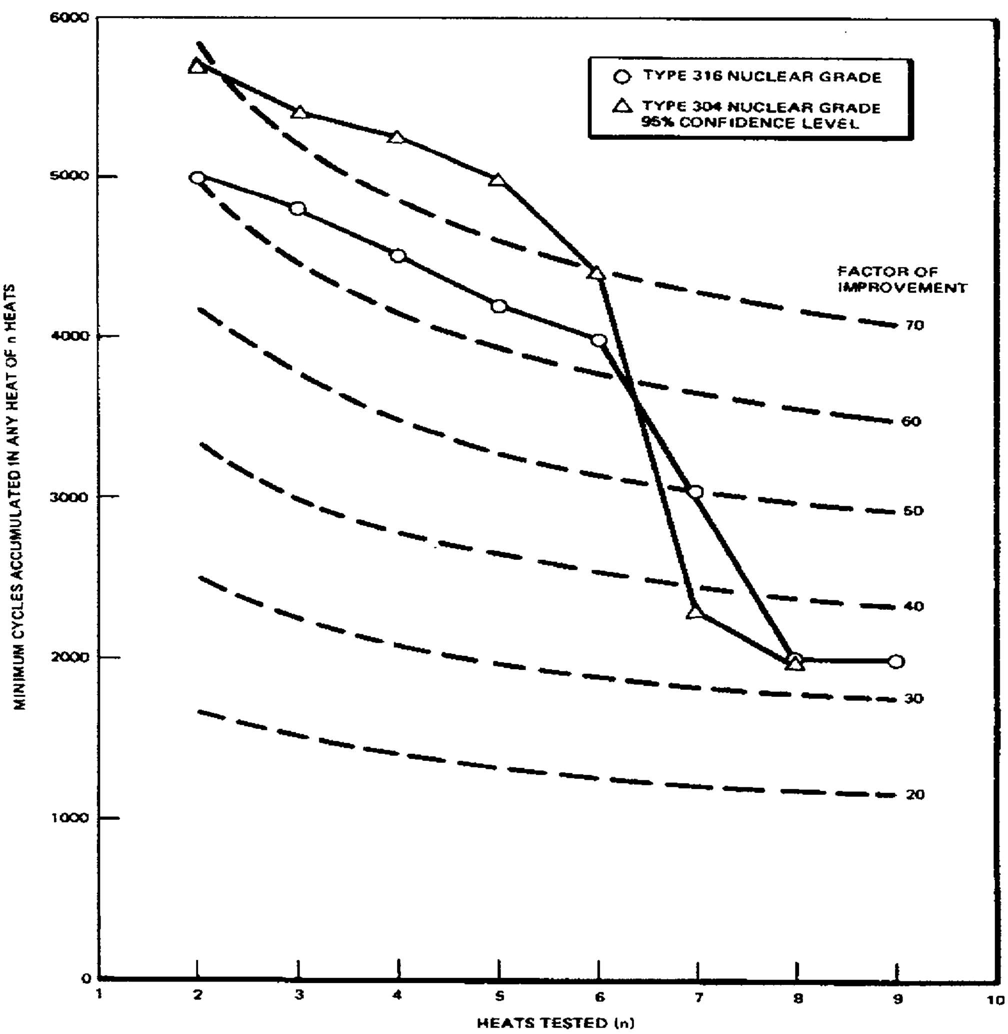


**Figure 2 Summary of Alternative Alloy Factor of IGSCC Improvement**

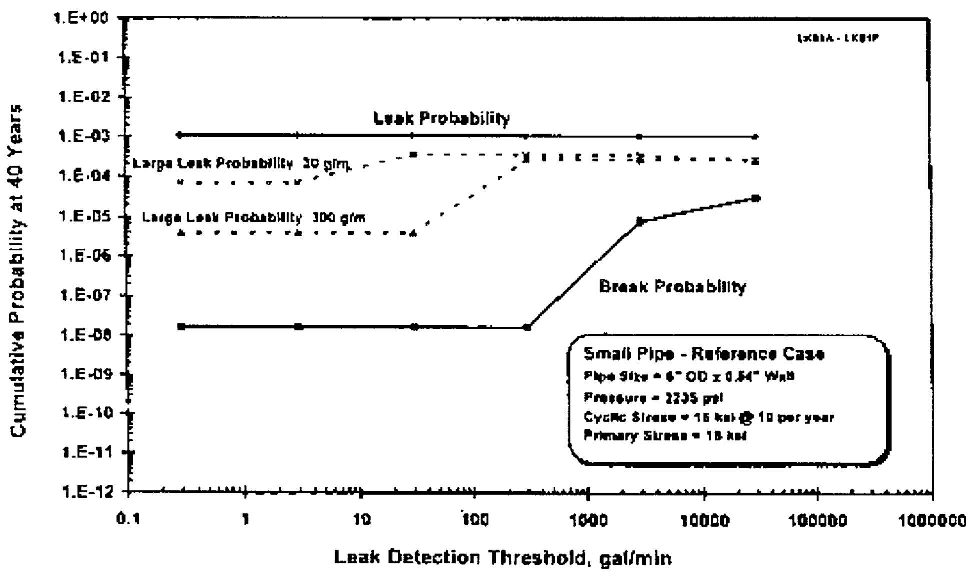


**Figure 3 Leak and Break Probabilities for Reference Case – Small Pipe with Low Fatigue Stress and High Primary Stress – High Internal Pressure**