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Pipe Cracking in U.S. BWRs: A Regulatory History

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Prepared By
J. R. Strosnider, Jr., W. H. Koo
J. A. Davis, M. C. Modes

Division of Engineering
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ABSTRACT

This paper was prepared, in part, to support discussions at an IAEA Regional Workshop on Environmentally Assisted Cracking of Nuclear Power Plant Austenitic Piping that was held in Slavutych, Ukraine from 22 to 26 June 1998 [1]. The paper presents a regulatory history and offers a perspective on intergranular stress corrosion cracking (IGSCC) in U.S. boiling-water reactor piping.

The paper focuses on regulatory and industry actions taken to assure that U.S. licensees manage IGSCC in a manner that provides safe and reliable plant operation. Although the paper does not offer extensive theoretical details on the IGSCC phenomenon, it does discuss some of the key technical issues that influenced regulatory positions and industry actions.

1. "Report of a Regional Workshop on Environmentally Assisted Cracking of NPP Austenitic Piping," IAEA TC Project RER/9/052, RBMK-SC-060, Vienna, November 4, 1998.

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EXECUTIVE SUMMARY

This paper presents a regulatory perspective on the history of intergranular stress corrosion cracking (IGSCC) in the piping of U.S. boiling-water reactor (BWR) plants. The paper focuses on regulatory and industry actions taken to assure that U.S. BWR licensees manage IGSCC in a manner that provides safe and reliable plant operation. Development of the U.S. regulatory framework for managing IGSCC in BWR piping was an evolutionary process driven by operating experience, research, and technological developments. IGSCC was initially observed in only small-diameter piping; however, in time it affected both large- and small-diameter piping systems. A key element in the management of IGSCC was an aggressive inspection program utilizing qualified inspection techniques and personnel. In parallel, the development of qualified repair and mitigation methods, along with replacement of piping with IGSCC-resistant materials resulted in an effective and economical solution to the problem. Today, a combination of regulatory documents, code rules, and industry guidelines is available to support effective programs for managing IGSCC in the piping of U.S. BWRs.

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1 INTRODUCTION

This paper presents a regulatory perspective on the history of intergranular stress corrosion cracking (IGSCC) in the piping of U.S. boiling-water reactor (BWR) plants. The paper is not intended to provide extensive theoretical details on the IGSCC phenomenon or other related subjects such as nondestructive testing and fracture mechanics. An extensive literature offers theoretical and engineering details on these subjects, and some selected references on these subjects are listed at the end of this paper. This paper focuses, rather, on regulatory and industry actions taken to assure that U.S. BWR licensees manage IGSCC in a manner that provides safe and reliable plant operation. Some key technical issues that influenced regulatory positions and industry actions are discussed.

Chapter 2 of this report briefly describes the typical design of BWRs operated in the U.S. Chapter 3 presents a brief discussion of the causal factors of IGSCC and BWR piping design features of particular interest with regard to IGSCC. The information in these sections serves as background to assist in understanding issues discussed in later sections of the paper. Chapter 4 briefly summarizes BWR pipe cracking experience in the U.S. In retrospect, given the material selections, fabrication methods, and operating environments of BWRs, the occurrence of IGSCC in BWR piping should not have been totally unexpected. However, the discovery of IGSCC in small-diameter piping in the late 1970s and in large-diameter BWR piping in the early 1980s required substantial efforts from the U.S. industry and the U.S. Nuclear Regulatory Commission to support development and implementation of the engineering and regulatory solutions to the problem. Chapter 5 summarizes the major regulatory initiatives taken in response to the BWR pipe cracking problem. Specific issues that had to be addressed were methods for mitigation, inspection, flaw evaluation, piping repair, and piping replacement. These issues are discussed in Chapters 6 through 10 of this paper, respectively.

Chapter 11 of this report discusses leak detection and leak-before-break (LBB) in the context of BWR IGSCC. It is important to note that LBB is not accepted in the U.S. as a regulatory basis for assuring the integrity of piping systems subject to IGSCC. Nonetheless, it is an important element in "defense-in-depth" and should be given close attention in any program for maintaining piping and component integrity. Chapter 12 of this report provides a current status of BWR piping in U.S. plants and Chapter 13 present major conclusions of the report.

2 TYPICAL DESIGN OF U.S. BOILING WATER REACTORS

The direct-cycle boiling-water reactor (BWR) nuclear system consists of a nuclear core located inside a reactor vessel and a conventional turbine-generator and feedwater supply system. Associated with the nuclear core are auxiliary systems to accommodate the operational and safeguards requirements. Water circulating through the reactor core produces saturated steam, which is separated from recirculation water, dried in the top of the vessel, and directed to the steam turbine-generator. The turbine employs a conventional regenerative cycle with condenser deaeration and condensate demineralization.

The nuclear core, the source of the heat, consists of fuel assemblies and control rods contained within the reactor vessel and cooled by the recirculating water system. The power level is maintained or adjusted by positioning control rods up and down within the core and by changing the recirculation flow rate through the core without changing the position of the control rods. The recirculation system utilizes jet pumps inside the reactor vessel. These pumps generate about two-thirds of the recirculation flow within the reactor vessel. The external recirculation system piping is made up of two separate parallel pump loops, each loop consisting of a reactor recirculation pump and associated piping. The pump discharge line in each loop feeds a pipe riser manifold that has five riser outlets. Each outlet supplies a pair of jet pumps. Connections are provided on the suction side of one loop to supply reactor coolant during the shutdown mode of the residual heat removal (RHR) system and to the reactor water cleanup system. The discharge piping of both loops has connections from the RHR system discharge to support both the shutdown cooling and low-pressure coolant injection modes of operation.
The BWR operates at a constant steam pressure of about 8.62 MPa (1250 psi) and a temperature of about 287.8°C (550°F). U.S. BWRs have fast-acting emergency core cooling systems (ECCSs).

The nuclear boiler system is supported by the following specialized functions of its auxiliary systems, which are used for normal plant operation:

- reactor water cleanup (RWCU) system
- shutdown cooling function of residual heat removal (RHR) system
- fuel building and containment pool cooling and filtering system
- closed cooling water system for reactor service
- radioactive waste treatment system

The following auxiliary systems are used as backup (standby) or emergency systems:

- standby liquid control (SLC) system

- reactor core isolation cooling (RCIC) system
- high-pressure core spray (HPCS) system
- low-pressure core spray (LPCS) system
- automatic depressurization function
- residual heat removal (RHR) system
  - low-pressure coolant injection (LPCI) function
  - steam condensing function
  - containment spray function
  - suppression pool cooling function

The emergency systems provide diverse and redundant means for achieving and maintaining safe shutdown of the reactor assuming design-basis pipe failures up to a full double-ended, offset break of the largest diameter pipe in the system.

BWRs in the U.S. are provided with multi-barrier pressure suppression containments. The primary containment consists of a drywell, which encloses the reactor vessel and recirculation system, and a pressure suppression chamber. The primary containment is designed to prevent the release of radioactive fission products to the environment in the event of a design-basis pipe break. A secondary containment consists of a reactor building that completely encloses the pressure suppression primary containment. This structure provides secondary containment when the primary containment is in service, and provides primary containment during periods when the primary containment is open, as during refueling.

3 CAUSE OF INTERGRANULAR STRESS CORROSION CRACKING

Intergranular stress corrosion cracking (IGSCC) occurs as a result of the effects of stress and environment on a susceptible material. These three factors - material susceptibility, environment, and stress - are discussed below.

The original piping material in U.S. BWRs was type 304 or 316 austenitic stainless steel. The corrosion resistance of austenitic stainless steel results from the addition of more than 12% chromium which combines with oxygen to form a passive chromium-oxide film on the surface of the material. However, if austenitic stainless steels that contain more than 0.03 % carbon are heated to or slowly cooled through the temperature range of 520 to 820 °C (900 to 1450 °F), chromium carbides will form at the grain boundaries. The regions adjacent to the precipitated chromium carbides will be depleted of chromium to less than 12%, resulting in a loss of corrosion resistance in those areas. Stainless steel in this condition is referred to as "sensitized." The most common cause of sensitization in BWRs is welding followed by a slow cooling through the sensitizing temperature range. The sensitization in these cases is contained within the weld heat-affected zone. Sensitization can be avoided by controlling the carbon content to below 0.03%, by cooling quickly through the sensitizing temperature range, or by solution annealing sensitized material and rapidly cooling. In the solution annealing process, the material is heated to 1100 to 1150 °C (1950 to 2050 °F) until the chromium carbides dissolve. The material is subsequently quenched to prevent the carbides from reforming. Another approach to reduce sensitization is to add strong carbide formers such as titanium or niobium to the steel. The carbon in the steel will preferentially react with the titanium or niobium to form titanium or niobium carbides instead of reacting with the chromium. Stainless steels with additions of titanium or niobium are called "stabilized." IGSCC can still occur in stabilized stainless steels if the carbon content is too high. Stabilized stainless steels generally have not been used in U.S. BWRs.

Environment plays a key role in the occurrence of IGSCC. The IGSCC mechanism is an electrochemical process; therefore the propensity for IGSCC is very dependent on the electrolytic nature of the environment in which the material is placed. Impurities in the BWR reactor coolant provide the electrolytic environment necessary to support IGSCC. These impurities include oxygen, which may be present as a result of radiolysis or the ingress of air into the reactor coolant, or anionic impurities such as sulfates or chlorides, which may be present as a result of leakage from the condenser into the reactor coolant. Higher levels of oxygen and impurities increase the potential for IGSCC. Controlling the oxygen concentration below 20 ppb by the addition of hydrogen will normally prevent IGSCC from initiating. Although it is important to control the bulk coolant chemistry, once a crack has initiated, the microchemistry within the crack is very difficult to change or control; thus, IGSCC may continue even after extensive efforts to improve the bulk chemistry have been initiated.

The third component necessary for IGSCC to occur is stress. Stress is generally considered to contribute to the IGSCC process through rupturing of the protective oxide film. The major stress contribution in BWR piping comes from residual welding stresses, which can be on the order of the yield stress of the material. Although lower in magnitude, other stresses - such as cyclic stresses from plant startup and shutdown or vibration - may play an important role.

The three factors discussed above, material susceptibility, environment, and stress, interact in a synergistic manner to produce IGSCC. The degree of sensitization, oxidizing potential of the environment and level of stress can be varied in different combinations to produce a range of initiation times and crack growth rates for IGSCC. This makes it very difficult to extrapolate laboratory data to field experience or to develop reliable models for predicting inservice performance, and care should be taken in utilizing such approaches to managing IGSCC.

4 IGSCC OPERATING EXPERIENCE

In this section, the staff summarizes IGSCC experience in U.S. plants. Reference is made to several pipe crack study groups, NUREG reports, and NRC bulletins that resulted in augmented inspections and other activities to address IGSCC. These regulatory activities are discussed in greater detail in Chapter 5 of this report. Consideration, in concert, of the contents of this Chapter and Chapter 5 provides greater insight to the interaction between operating experience and regulatory actions that ultimately lead to a comprehensive approach to managing IGSCC.

In NUREG-0531, "Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants" the NRC staff and its contractors summarized information regarding cracking in BWR piping from 1965 through 1978. IGSCC was identified in BWR piping as early as 1965. In December 1965, during a hydrostatic pressure test, a leak was observed in a 6-inch bypass line of the recirculation loop at Dresden Unit 1. From December 1965 until September 1974, cracks were found in the piping of six U.S. BWRs. All cracking was found in the heat-affected zone of welds in type 304 stainless steel piping of 8-inch or smaller diameter. From 1975 through 1978, IGSCC was found in recirculation bypass, core spray, reactor water cleanup and control rod drive return piping. NUREG-0531 contains a summary of pipe cracking in both domestic and foreign BWRs; this information is also summarized in Table 4.1 below.

Table 4.1 IGSCC Incidents by Line Type in U.S. and Foreign BWRs[(a)]

	Number IGSCC Incidents		
System Component (pipe diameter)	Before July 1975	July 1975 to January 1979	Totals
Recirculation Bypass Line (4-inch)	30	12	42
Core Spray Pipe (10-inch)	16	17	33
Control Rod Drive System Small Bore Pipe (CRD, 3-inch)	1	1	2
Reactor Water Cleanup (RWCU; 3- to 8-inch)	10	14	24
Large Recirculation (≥ 12-inch)	0	13	13
Small Bore Pipe (≤ 3-inch) other than CRD and RWCU	0	6	6
Other Piping Systems	7	6 [(b)]	13

Notes: (a) Cracking incidents reported to the NRC
(b) Cracking incidents in a large diameter pipe of a German BWR

Many of the cracks discovered after 1975 were found as the result of augmented inservice inspections performed consistent with the recommendations of the first Pipe Crack Study Group and the NRC positions in NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," as discussed in Chapter 5 of this report.

In NUREG-0531, the staff also noted that IGSCC had occurred in BWR Inconel safe-ends. Safe-ends are attached to the vessel to serve as transition pieces to facilitate welding of the austenitic stainless steel piping to the ferritic steel pressure vessel nozzles. The safe-end material can be sensitized when the ferritic steel components to which they are attached are given a post-weld heat treatment. In 1975, only three BWRs with sensitized safe-end material remained in operation in the U.S., and these plants were performing periodic inspections of the safe-ends.

One of the most significant IGSCC incidents reported in the U.S. occurred at Duane Arnold on June 17, 1978. During this event, a leak of approximately 684 L/hr (3 gpm) was identified from one of the eight 10- inch diameter recirculation inlet nozzle safe-ends. Each of these safe-ends is fabricated with an internal thermal sleeve that creates a deep crevice in the design. The safe-end had been leaking for about 3 days before the source of the leak was identified following an unrelated reactor scram. Following identification of the leaking safe-end, the licensee performed ultrasonic testing (UT) of the remaining safe-ends and subsequently replaced all of the safe-ends in the system.

Destructive examinations of the affected safe-ends revealed that all eight safe-ends had inside surface cracks which extended essentially completely around the circumference of the design; these cracks were located in the creviced region of safe-end design. The depth of the cracks typically ranged from 50 percent to 75 percent of the wall thickness, except for the leaking safe-end where an through-wall crack was present in an 80-degree segment of the circumference. The licensee attributed the cracking to IGSCC and concluded that the cracking was a result of the combined effects of an oxygenated reactor coolant environment, high residual stresses from welding, use of a sensitized material during the fabrication of the components, and the presence of the deep crevice in the design. However, the licensee concluded that the conditions favorable to supporting IGSCC existed even without the contributions from However, the licensee concluded that the conditions favorable to supporting IGSCC existed even without the contributions of these two factors.

The cracking at Duane Arnold was significant in that it demonstrated the potential for IGSCC to develop deep, 360-degree flaws. This type of flaw challenges the concept of LBB (refer to Chapter 11) and emphasized the need for comprehensive inspections.

In NUREG-1061, Volume 1, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," the fourth Pipe Crack Study Group, as discussed in Chapter 5, updated both domestic and foreign BWR operating experience through March 1984. Augmented inspections performed in accordance with NUREG-0313, Revision 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," published in July 1980, had revealed a significant amount of cracking in small-diameter piping in systems such as RWCU and core spray. The results of inspections of large-bore piping conducted in response to NRC Generic Letters 82-03 and 83-02 were of particular interest. NUREG-1061, Volume 1, provides a tabular summary of these inspection results. These inspection results are also summarized in Table 4.2.

Table 4.2 indicates that a wide variety in the extent of cracking occurred from plant to plant without any clear explanation of the discrepancy in the inspection results. This experience illustrates the observation in Chapter 2 that it is very difficult to develop reliable models for predicting inservice performance of materials. These results also demonstrated the need for a comprehensive, integrated approach to IGSCC inspection.

In NUREG 1061, Volume 1, the staff reported that as a result of this inspection experience, U.S. utilities were developing programs to preclude or minimize IGSCC. Several utilities were in the process of replacing (or planning to replace) susceptible material. Weld overlay repairs (refer to Chapter 9.0) and induction heating stress improvement (refer to Chapter 6, Section 6.2) had been implemented at several sites.

Since the issuance of Generic Letter 88-01, the industry has been performing IGSCC inspections on a regular basis. The inspection schedule for each weld was based on the weld's degree of susceptibility to IGSCC. In the earlier rounds of inspection, many large pipe welds were found to be cracked. Recently, there have been very few reported events pertaining to IGSCC in austenitic stainless steel piping. This reduction in reported cracking is most likely due to one of the following factors:

- The most highly susceptible piping was replaced with piping fabricated from more IGSCC-resistant materials.

- Cracks in the most highly susceptible locations occurred relatively early in life and were identified and overlay repaired during the early rounds of inspections.

- Implementation of effective mitigation measures such as stress improvement, improved water chemistry and hydrogen water chemistry has reduced the potential for IGSCC in the remaining welds.

Table 4.2: Summary of Inspection Findings on Large Piping in BWRs Inspected According to IEBs 82-03 and 83-02 [(a)]

Plant	Extent of Inspection (% of Welds Inspected)		Inspection Results (Number of Cracked Welds)		Number of Welds Repaired by Overlay
	Recirculation	RHR	Recirculation	RHR	
Big Rock Point	20% (11/59)	---	0	---	0
Browns Ferry 1	98% (103/109)	90% (36/40)	33	14	42
Browns Ferry 2	27% (25/91)	28% (9/31)	2	0	0
Browns Ferry 3	8% (	28% (9/31)	0	0	0
Brunswick 1	25% (29/115)	75% (3/4)	3	0	3
Brunswick 2	100% (102/102)	100% (5/5)	15	1	8
Cooper	100% (108/108)	100% (7/7)	20	0	13
Dresden 2	47% (47/101)	10% (4/40)	10	0	7
Dresden 3	100% (115/115)	90% (45/50)	53 [(b)]	11 [(b)]	61 [(b)]
Duane Arnold	42% (49/117)	40% (2/5)	0	0	0
FitzPatrick	47% (49/106)	45% (5/11)	1	0	0
Hatch 1	47% (47/100)	100% (11/11)	5	2	6
Hatch 2	94% (97/103)	100% (11/11)	36	3	27
Millstone 1	11% (11/100)	0% (0/46)	0	0	0
Monticello	100% (106/106)	78% (18/23)	6	0	6
Nine Mile Pt. 1	82% (62/76)	---	53	0	0
Oyster Creek	39% (31/80)	---	0	0	0
Peach Bottom 2	100% (91/91)	91% (32/35)	19	7	24
Peach Bottom 3	91% (77/85)	92% (35/38)	10	5	15
Quad Cities 1	8% (9/110)	20% (9/44)	0	0	0
Quad Cities 2	100% (106/106)	90% (45/50)	20	2	9
Vermont Yankee	56% (58/88)	7% 2/30)	33	1	22

Notes: (a) The Boston Edison Company (BECo) inspected seven (7) welds at the Pilgrim Atomic Power Plant and detected cracks in 4 of the welds. BECo decided to replace the piping with Type 316NG stainless steel, and therefore terminated further inspections of the systems.

(b) Eighteen welds that were originally reported to be cracked and later re-evaluated and determined not to be cracked are not included in these totals.

5 CHRONOLOGY OF REGULATORY ACTIONS

5.0 Overview

This chapter presents a chronology of significant U.S. regulatory activities addressing IGSCC in BWR piping. Three special Pipe Crack Study Groups were established to assess the IGSCC issue in BWR piping. These groups and their conclusions and recommendations were driven largely by operating experience and represent the evolution of an integrated and mature regulatory approach to managing IGSCC in BWR piping.

5.1 First Pipe Crack Study Group—October 1975

The first Pipe Crack Study Group was formed to coordinate and accelerate the Atomic Energy Commission's (AEC's) investigation of pipe cracking. The group published the results of its study in NUREG-75/067, "Technical Report, Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants." The recommendations stated in NUREG-75/067 were implemented in NUREG-0313, which was issued in July 1977.

5.2 NUREG-0313—July 1977

In NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," which was issued in July 1977, the staff concluded that the probability was extremely low that IGSCC would propagate far enough to create a significant safety hazard, but that actions should be taken to avoid or manage IGSCC in BWR piping. The report presented (1) the NRC technical positions on actions to take to avoid IGSCC and (2) augmented inspection and leak detection guidelines to be applied in the event that actions to avoid IGSCC were not practical.

5.3 Second Pipe Crack Study Group and NUREG-0531—February 1979

The NRC established the second Pipe Crack Study Group in 1978 to further study the IGSCC issue. This group was formed after IGSCC was reported in large-diameter (> 20-inch-diameter) piping in a German BWR and after significant IGSCC was discovered in a recirculation inlet nozzle safe-end at the Duane Arnold plant in the U.S. The charter of the second study group included consideration of (1) the implications of IGSCC in large diameter piping, (2) the resolution of issues related to ultrasonic testing (UT), (3) the implications of IGSCC in large diameter safe-ends, (4) the potential for IGSCC in PWR piping, and (5) evaluation of the Duane Arnold safe-end cracking and development of specific recommendations based on that experience. In NUREG-0531, "Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants" the study group documented its conclusions and recommendations. The report recommended avoiding further use of IGSCC-sensitive materials, considering mitigating actions (e.g., induction heating stress improvement(IHSI)), minimizing oxygen levels in BWR coolant, codifying IGSCC ultrasonic inspection methods, further optimizing of inspection methods, implementing augmented inspection programs, and performing further studies of operating factors influencing IGSCC. These study group recommendations were implemented by issuing NUREG-0313, Revision 1.

5.4 NUREG-0313, Revision 1—July 1979

Following completion of the second Pipe Crack Study Group investigation and issuance of NUREG-0531, in October 1979, the staff revised NUREG-0313 to present updated guidance and recommendations regarding materials and processes that could be used to minimize IGSCC, and to make recommendations regarding augmentation of the extent and frequency of inservice inspections (ISIs) on welds considered to be susceptible to IGSCC. Revision 1 also contained recommendations about upgrading leak-detection systems and leakage limits from plants with susceptible welds. The staff issued these recommendations to the industry in Generic Letter 81-03, "Implementation of NUREG-0313, Revision 1,"on February 26, 1981.

5.5 Third Pipe Crack Study Group and NUREG-0691—September 1980

During 1979, several instances of cracking in pressurized-water reactor (PWR) piping systems led the NRC to establish the third Pipe Crack Study Group. Since the third Pipe Crack Study Group focused on cracking in PWR rather than BWR piping, the work of this group is not discussed in this paper. It is listed here only for completeness.

5.6 Information Notice 82-39—September 1982

After cracking was detected at Nine Mile Point Unit 1, the NRC Office of Inspection and Enforcement (IE) issued Information Notice 82-39, "Service Degradation of Thick Wall Stainless Steel Recirculation System Piping at a BWR Plant," on September 21, 1982, to alert all BWR licensees to the problem.

5.7 Bulletin 82-03—October 1982

Early in 1982, the NRC met with all BWR licensees to discuss plans for near-term inspections of welds in large-diameter recirculation piping, and on September 27, 1982, issued IE Bulletin 82-03, "Stress Corrosion Cracking in Thick Wall, Large Diameter, Stainless Steel Recirculation system Piping at

BWR Plants." The bulletin required that licensees of nine BWRs with outages scheduled through January 31, 1983, inspect a sample of the welds in the recirculation system during the next outage. In order to assure effective ultrasonic inspection, the bulletin required that procedures and competence of the examiners be demonstrated on samples of cracked piping removed from the Nine Mile Point recirculation system.

5.8 Bulletin 83-02—March 1983

After cracking was found in large pipes in a number of plants inspected in accordance with Bulletin 82-03, the staff issued another bulletin, Bulletin 83-02, "Stress Corrosion Cracking in Large Diameter Recirculation System Piping at BWR Plants," in March 1983. Bulletin 83-02 extended the inspection requirement to all other BWRs with more than 2 years of operating service (14 units). It also required utilization of inspection personnel who had passed upgraded UT performance capability demonstrations. In addition, it required that a larger initial sample of welds be inspected than was required in accordance with IE Bulletin 82-03, and contained provisions for further increase in the inspection sample if cracks were found.

5.9 NRC Orders—August 1983

Because the inspections performed in accordance with Bulletin 83-02 continued to indicate that cracking in most BWRs was extensive and that many cracked welds required weld overlay repair for further operation, the NRC issued orders on August 26, 1983, to five plants to conduct as soon as practical the augmented inspection of IGSCC in large-diameter pipes in recirculation and residual heat removal (RHR) systems.

5.10 Generic Letter 84-11—April 1984

The results of the IGSCC inspections performed in accordance with the orders issued in August 1993 indicated that it was necessary to perform similar reinspections at all operating BWRs. In Generic Letter 84-11, that was issued on April 19, 1984, the staff presented criteria for conducting short-term reinspections, for detecting leaks and determining leakage limits, for evaluating flaw indications, and for making repairs.

5.11 NDE Coordination Plan on Inspection Qualification—July 1984

To gain confidence in the ability of ultrasonic inspection personnel to detect IGSCC cracks and determine their size, a program was established at the Electric Power Research Institute (EPRI) Non-destructive Examination (NDE) Center in Charlotte, North Carolina. This program was developed in accordance with an NDE Coordination Plan designed to provide training and to qualify the examination procedures, equipment, and inspection personnel through demonstration on flawed samples. The NRC, EPRI and the BWR Owners Group (BWROG) agreed on the NDE Coordination Plan on July 3, 1984, and upgraded it in September 1985 (the plan is sometimes referred to as the tripartite agreement).

5.12 Fourth Pipe Crack Study Group and NUREG-1061, Volume 1—August 1984

The NRC established the fourth Pipe Crack Study Group in August 1984. The charter of the group called for developing an integrated plan to deal with the entirety of the stress corrosion cracking problem in BWR piping. The results of inspections conducted according to IE Bulletins 82-03 and 83-02 showed a wide variation in the extent of cracking reported. The scope of the fourth Pipe Crack Study Group was very similar to that of the second group. The main difference was that the fourth group related its recommendations to the regulatory documents that required revision in order to implement the recommendations. In NUREG-1061, Volume 1, the group documented its conclusions and recommendations. The report updated the discussions of the IGSCC mechanism, nondestructive testing for IGSCC, short-term and long-term approaches to mitigating IGSCC, and flaw evaluation, repair, and replacement options. The report contains specific recommendations in each of these areas with the intent of providing an integrated regulatory approach to managing IGSCC. It also summarized the status of domestic and foreign BWR pipe cracking.

5.13 NUREG-0313, Revision 2—June 1986 and Generic Letter 88-01— January 1988

Subjects covered by Revision 2 to NUREG-0313, which was published in June 1986, included recommendations regarding piping and weld material selection, special processing to minimize IGSCC susceptibility, improvements in BWR coolant chemistry and control, inspection requirements, repair methods, and leak detection. The conclusions and recommendations in the report are largely consistent with those made in NUREG-1061, Volume 1, prepared by the fourth Pipe Crack Study Group. The NRC positions stated in the report were implemented by Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," dated January 25, 1988.

5.14 Generic Letter 88-01, Supplement 1—February 1992

The staff issued Generic Letter 88-01, Supplement 1 on February 4, 1992. This supplement presented acceptable alternatives to some of the staff positions delineated in Generic Letter 88-01. The alternatives presented concerned requirements for inspecting of reactor water cleanup (RWCU) system piping outside the containment isolation valves and requirements pertaining to the operability of leakage measurement instruments and the frequency of monitoring leakage rates. The supplement also clarified the staff's positions regarding the sample expansion of Category D welds, the effect of shrinkages resulting from weld overlay repairs and the need for technical amendments to incorporate IGSCC ISI and leak detection requirements, as delineated in GL 88-01, into the technical specifications.

5.15 Appendix VIII and the Performance Demonstration Initiative—1996

The 1989 edition of Section XI of the ASME Code incorporated a new Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," and the 1989 addenda to the 1989 edition incorporated a new Appendix VIII, "Performance Demonstration for Ultrasonic Examination". Together, these two appendices present code rules for qualifying personnel, techniques, and procedures. The qualification approach is intended to be performance based. In this regard, Appendix VIII does not present prescriptive code rules for inspection methods; rather, it relies on the demonstrated ability to detect and size defects in a prescribed number of test specimens. In order to implement the new-performance based approach, the industry developed the Performance Demonstration Initiative (PDI). This is a program, that is managed for the industry by EPRI, and which provides the necessary facilities, test samples, training, test procedures, and test administration to implement Appendix VIII. In March 1996, the NRC accepted the PDI qualification program for IGSCC in lieu of the program previously established under the NDE Coordination Plan (tripartite agreement), the NRC, EPRI , and the BWROG agreed to on July 3, 1984. The NRC is currently finalizing a revision to 10CFR50.55a that will endorse Appendix VIII of the ASME Code.

6 MITIGATION METHODS

6.0 Overview

As discussed in Chapter 2, IGSCC is controlled by three factors; material susceptibility, environment, and stress. Efforts to avoid or mitigate IGSCC focus on control of these three factors. Eliminating material susceptibility generally requires replacement of the piping. This subject is discussed in Chapter 10. Actions to mitigate IGSCC in operating plants, without piping replacement, focus on controlling environment or stresses.

6.1 Water Chemistry Control

Two approaches to modifying water chemistry have been used to mitigate IGSCC. The first approach has been to reduce the levels of impurities in the reactor coolant that can accelerate the initiation and propagation of IGSCC. The second approach has been to reduce the level of oxygen in the reactor coolant through the addition of hydrogen.

Controlling the level of impurities in the bulk reactor coolant is important for a variety of reasons including minimization of IGSCC, fuel performance degradation, and radiation buildup. In 1973 the NRC issued Regulatory Guide (RG) 1.56, "Maintenance of Water Purity in Boiling Water Reactors," and revised it in 1974. These regulatory guides recommended that licensees establish maximum conductivity levels that would (1) indicate breakthrough of one or more demineralizer units, (2) require orderly shutdown of the reactor, and (3) require immediate shutdown of the reactor. The regulatory guidance included representative conductivity levels for each of these conditions and representative chloride limits. However, research by the General Electric Company (GE) in the late 1970s showed that within the conductivity levels recommended in RG 1.56, dissolved anionic impurities could significantly affect IGSCC initiation and propagation. The two most common anions that were found to affect IGSCC were (1) sulfates introduced as a result of condenser leakage, chemical in-leakage, and resin ingress from the demineralizers; and (2) chlorides introduced as a result of condenser leakage. Chromates introduced through corrosion products can significantly contribute to room-temperature conductivity. This research resulted in the recognition that more stringent controls than those recommended in RG 1.56 were necessary to effect any change in the potential for IGSCC. Subsequently, in 1986 EPRI published recommended water chemistry control guidelines, and in 1987 EPRI issued hydrogen water chemistry guidelines. EPRI has conducted additional, extensive research into BWR water chemistry and in 1996 issued a revision to the BWR water chemistry guidelines that contains both the water chemistry control guidelines and the hydrogen water chemistry guidelines in one document. The EPRI guidelines establish three operational conditions: cold shutdown, startup/hot standby, and power operation (at power levels above 10-percent power). For each of these operational conditions, the guidelines establish action levels that should not be exceeded. Action Level 1 represents those impurity or chemistry parameter level which if exceeded, would threaten long-term system reliability of the plant system. The guideline suggests that the plant impurity or parameter conditions should be brought to levels below Action Level 1 within 96 hours, or else a review should be performed and a program and schedule for implementing corrective actions should be developed and presented to management. Action Level 2 represents the impurity or chemistry parameter level which, if exceeded, could cause damage to the plant system in the short-term, and if conditions are not reduced below the levels indicated by Action Level 2 within 24 hours, the plant should be brought to cold shutdown in an orderly fashion. Action Level 3 represents the impurity or chemistry parameter level which, if exceeded, indicates significant damage could occur to the plant system in the short-term and an orderly shutdown of the reactor should be initiated immediately. Action levels also are suggested in the guidelines for conductivity, chloride level, and sulfate level during cold shutdown and for dissolved oxygen during startup and hot standby. For power operation, action levels are suggested for zinc and electrochemical potential in addition to the other action levels. Implementation of these guidelines can be beneficial in extending IGSCC initiation times and in reducing IGSCC growth rates. However, research and operating experience show that improving the water chemistry purity in an operating plant cannot, alone, eliminate IGSCC.

U.S. BWRs are currently operating with much better water chemistries (conductivity $< 0.3\ \mu S/cm$), which have reduced the potential for IGSCC. However, as noted in Chapter 3, once IGSCC has initiated, the microchemistry at the crack cannot be effectively altered through control of the bulk water chemistry.

Hydrogen water chemistry (HWC) is a more aggressive approach for modifying the reactor coolant chemistry and has significant potential benefits. HWC involves the addition of hydrogen to the reactor coolant to reduce the level of dissolved oxygen in the coolant. By reducing the level of dissolved oxygen in the coolant, the electrochemical potential of the stainless steel will be shifted to a region in which IGSCC should not initiate or propagate. During the late 1970s, laboratory testing and field testing using special test loops installed at Oskarshamn and Ringhals in Sweden, and at Dresden Unit 2 in the U.S., showed that reduction in typical BWR dissolved oxygen levels from 200 ppb to about 20 ppb could halt IGSCC propagation.

Currently, about 15 BWRs in the U.S. operate with HWC. The operating experience for these plants has shown various levels of arrest of IGSCC, depending on how much hydrogen is added.

It should be noted that use of HWC increases the radiation levels in the steam system because of increased levels of N-16. Operating experience indicates an increase in radiation by a factor of up to 8, which can require additional radiation protection measures. Also, radiation fields from Co-60 have increased by more than 30 percent at some plants that have changed to HWC. As of November, 1993, 13 plants were injecting zinc into the feedwater to control Co-60 radiation levels.

6.2 Stress Improvement

Experimental and analytical studies show that significant variability can exist in the through-wall axial stress distributions of both small- and large-diameter pipe. But generally, the as-welded residual stress distributions in large-diameter pipes have a sinusoidal shape beginning with a high (on the order of 25 to 40 ksi) inside surface tensile stress, decreasing to a compressive stress field in the inside half of the pipe wall, and returning to a tensile stress field in the outer half of the pipe wall. Several methods of stress improvement have been developed. These include induction heating stress improvement (IHSI), last pass heat sink welding (LPHSW), heat sink welding (HSW), and mechanical stress improvement. The objective of all these methods is to create a compressive, or at least lower, tensile stress field at the inside surface of the pipe.

The IHSI process has been used extensively in the U.S.. In the IHSI process, induction coils are used to heat the pipe at the weld location and create a temperature gradient across the wall so that the material at the inside surface of the pipe yields in tension and the material at the outside surface of the pipe yields in compression. This results in a compressive residual stress field at the inside surface of the pipe. When combined with operating stresses, the resultant stress field may be tensile, but of much lower magnitude than the original as-welded stress field. Experimental work in the late 1970s and early 1980s showed improved life for defect-free specimens that received IHSI. The benefits of IHSI for specimens with pre-existing cracks were less clear, and concerns were expressed regarding the possible extension of preexisting cracks during the IHSI process, stress relaxation over time, and limited data with regard to pipe-to-component welds. It was also noted that the process parameters need to be carefully controlled to achieve the desired results. In NUREG 0313, Revision 2, the staff allowed a reduction in the frequency of IGSCC inspections for susceptible welds treated with IHSI. For IGSCC-susceptible welds treated with IHSI in the first 2 years of operation, the staff reduced the inspection scope and frequency in NUREG 0313, Revision 2,the inspection scope and frequency from all welds every two refueling cycles to 50 percent of the welds every 10 years. All welds that received IHSI after 2 years of operation had to be inspected once within the two refueling outages following the IHSI application and then every 10 years following that inspection. A large number of susceptible welds have been mitigated with IHSI to improve their resistance to IGSCC. The operating experience of IHSI-treated welds has been reasonably good. However, the effectiveness of IHSI treatment has been questioned because small circumferential or axial cracks have been found on a few IHSI-treated welds. The probable root cause of this cracking has been attributed either to the failure of NDE examiners to detect the indications during the inspections that were performed preceding the application of IHSI or the failure of IHSI to achieve the desired compressive residual stress field on account of the complex weld joint geometry. IHSI has not been incorporated in the ASME Code; however, EPRI has presented industry guidelines regarding the proper procedures for effective application of IHSI.

In addition to IHSI, the mechanical stress improvement process (MSIP) has been developed as an alternate stress improvement process. Instead of using a large temperature gradient through the wall of the pipe to achieve the desired plastic strain pattern, MSIP uses mechanical methods stated to be less expensive and time-consuming, thereby exposing the technicians performing the operation to less radiation. In the MSIP, the pipe is "squeezed" concentrically at a location about 2 inches to one side of the weld being treated. The force is provided hydraulically, working through split rings and flexible metallic pads between the rings and the pipe. Through this process a small permanent reduction in diameter is achieved. After the equipment is removed, the elastic recovery of the pipe results in residual tensile stresses in the squeezed area, balanced by compressive residual stresses in the weld and heat-affected zone (HAZ) area at the inside surface of the pipe. The residual stress pattern produced by MSIP has been confirmed by finite element analyses. Some surface residual stresses were also experimentally evaluated and confirmed the analytical results. Through-wall residual stresses were measured on MSIP-treated 12-inch and 28-inch weldments. The axial compressive residual stresses were found to extend to almost 50 percent of the wall thickness. MSIP was accepted in NUREG 0313, Revision 2, as a stress improvement (SI) process for mitigation of IGSCC in BWR plants. This process has not been incorporated in the ASME Code. On the basis of a 1990 report, MSIP had been applied to 532 welds in 12 BWR plants in the world, including 157 nozzle welds ranging from 4-inch to 28-inch diameter. There is no reported cracking in MSIP-treated welds.

Heat sink welding (HSW) is a process of butt welding pipes or fittings in which the major portion of the weld is produced while cooling water is flowing inside the pipe. The cooling effect of the water minimizes the sensitization caused by the welding process and, in addition, produces a steep temperature and stress gradient through the pipe wall during welding. After the weldment is cooled, the inner portion of the weld is under high compressive residual stress. The combination of reduced sensitization and high beneficial residual stresses provides significant resistance to IGSCC. Last pass heat sink welding (LPHSW) is a welding process similar to HSW, except that only the last welding passes are performed when there is cooling water inside the pipe. Initial assessment of these processes in the early 1980s indicated that they were more sensitive to the weldment geometry than IHSI, and data on the effectiveness of the LPHSW process in mitigating IGSCC was limited. Thus, although NUREG-0313 gave credit only for the HSW process in reducing inspections, these processes have been determined to be good practice in making repairs or replacements. There is very little operating experience reported on the effectiveness of these processes. The processes have not been incorporated into the ASME Code.

7 INSPECTION

7.0 Overview

Extensive inspections have been a critical element of the NRC and U.S. industry's efforts to manage IGSCC in BWR piping. Inspection is key to identifying ongoing degradation, assessing the significance of the degradation and assuring the effectiveness of repairs. For an inspection program to be effective several key areas have to be addressed. These are 1) the qualification of inspection personnel and methods i.e., techniques and procedures; 2) definition of the inspection scope and frequency; and 3) criteria for dispositioning inspection results. The NRC and U.S. industry expended significant resources during the early 80s to improve the reliability of IGSCC inspections and continues to expend significant resources to perform effective inspections.

7.1 Qualification of Inspection Personnel and Methods

IGSCC inspection capabilities in the late 1970s and early 1980s were in need of significant improvement. Plant examinations and laboratory studies showed that the inspection methods and personnel qualification requirements in place at the time were resulting in poor reliability of detection and poor flaw sizing capabilities. It was clear that a more rigorous approach to qualifying inspection methods and personnel was needed. As discussed in Section 5.0, pipe crack study groups formed in 1975, 1979, and 1984 assessed and made recommendations for improvements in ultrasonic inspection of IGSCC. This subject was addressed in the NUREG reports published by the pipe crack study groups and in NUREG-0313, Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping." However, the first significant change in qualification requirements for NDE personnel and procedures came in 1982, when the NRC issued IE Bulletin 82-03. Bulletin 82-03 requested that operating BWR plants planning outages in 1982 must demonstrate the capability of their ultrasonic testing to detect IGSCC in large-diameter piping.

In response to Bulletin 82-03, the industry established a limited ultrasonic performance demonstration at Battelle Columbus Laboratories. Using specimens removed from Nine Mile Point, the demonstration consisted of blind examinations of five pipe sections containing a total of five cracks. In order to assure ultrasonic capability sufficient to detect IGSCC, a passing criterion was established requiring detection of four of the five cracks. In response to the results of this demonstration and the inspections that followed, the NRC issued IE

Bulletin 83-02 requiring the continued inspection of piping using qualified personnel. This bulletin also established criteria for ultrasonic false calls and the number of inspection personnel representing a facility.

During the same time period the industry had established the Boiling Water Reactor Owners Group (BWROG) IGSCC Research Program. One of the objectives of the program was the development of ultrasonic techniques that would minimize the outage time required for inspection and repair. To further this objective, the Electric Power Research Institute (EPRI) established the Nondestructive Evaluation (NDE) Center in Charlotte, North Carolina. The center was responsible for establishing a formal qualification program for the ultrasonic inspection of IGSCC. As discussed in Chapter 5, the BWROG, EPRI and the NRC agreed to this program on July 3, 1984 in a memorandum of understanding that became known as the "tripartite agreement." As a consequence, no ultrasonic inspector could perform augmented IGSCC inspections at power plants unless he or she was qualified by the EPRI NDE Center.

Research continued at the NRC and EPRI into the accuracy of UT. The NRC and several U.S. corporations participated in an international series of round robin testing as well. The body of research led ultimately to the conclusion that UT, as performed at nuclear power stations in the U.S., had significant room for improvement. This issue was taken up by the American Society for Mechanical Engineers, Subcommittee Section XI. At the Subgroup for NDE, the need for a separate and comprehensive qualification program was established. This specification was established as mandatory Appendix VIII to the ASME Code and published in 1989. The industry established a program to comply with Appendix VIII as the "Performance Demonstration Initiative" (PDI). This program for comprehensive qualifications was established at the NDE Center under the administration of EPRI personnel.

In March 1996, the NRC agreed with the BWROG and EPRI that the performance qualification for IGSCC examiners could be administered through the PDI in lieu of the previous tripartite agreement. PDI currently provides the test specimens, training, procedures, and other services necessary to qualify IGSCC inspection personnel and methods, and the current PDI qualification carries a separate certification for the examination of piping for IGSCC.

Details of the most effective inspection methods are available in the literature. A few key aspects of IGSCC inspection methodology are presented here. UT is the most common and effective method of inspection. Isotopic radiography (RT), using Ir-192 or Co-60 has not be shown to be very effective for the detection or sizing of IGSCC. A field-deployable high-energy X-ray was developed and used successfully at one plant. It was demonstrated as an effective technique for detecting IGSCC; however, the equipment was too cumbersome for regular use and the approach was abandoned. The UT methods demonstrated most effective for IGSCC detection are 60-degree and 70-degree shear waves at 1.5 MHz and 60-degree and 70-degree refracted longitudinal waves in the range of 2-2.25 MHz. These configurations are used to generate corner trap and tip diffraction signals with a confirmatory creeping wave used to determine the degree of surface ligament to the tip. This method is effective in generating ultrasonic signals that are less susceptible to beam redirection caused by anisotropic wave velocities in the large weld grain size. A side benefit is the ability to direct the beam under interfering crown geometries, thus avoiding the need for a full "V" ultrasonic interrogation and its additional susceptibility to beam misdirection and complicated geometric calculations for flaw location. For simple length determination, amplitude drop is a conservative and readily applied method.

7.2 Scope and Frequency of Inspections

Extensive and frequent inspections are a major part of the NRC and U.S. industry approach to managing IGSCC in BWR piping. IE Bulletin 82-03, issued in October 1982, and IE Bulletin 83-02, issued in March 1983, requested that licensees owing BWR-design reactors (nine with scheduled outages through January 31, 1983, and the remaining licensees owning BWRs with more than two years of amassed operating service) inspect the large diameter recirculation piping for their units. In addition, the NRC issued orders on August 26, 1983, to five plants requiring inspections to be performed as soon as practical. Inspections conducted in response to these bulletins and orders identified IGSCC in large-diameter recirculation and RHR systems. Subsequently, on April 19, 1984, the NRC issued Generic Letter 84-11 requesting all affected BWR licensees to implement an augmented inspection program for IGSCC reinspection. The reinspection had to be performed within 2 years from the previous inspection performed in accordance with the IE Bulletins and Orders. On the basis of the industry's inspection experience, the scope of the inspections requested in GL 84-11 was expanded to include all stainless steel welds, susceptible to IGSCC, in piping equal to or greater than 4 inches in diameter, or in systems operating over 93.3 °C (200°F) that are part of or connected to the reactor coolant pressure boundary, out to the second isolation valve.

The NRC positions stated in NUREG-0313, Revision 2, and implemented by Generic Letter 88-01 and its supplement constitute the primary basis for licensee

IGSCC inspection programs still being implemented today. All IGSCC-susceptible welds made of austenitic stainless steel and nickel-based materials are included, irrespective of ASME Code classification. The extent and frequency of the inspections were dependent on the resistance of the materials and the effectiveness of any processes used to prevent cracking. Welds were categorized as "A" through "G," in accordance with their susceptiblity to IGSCC based on material type, processing history, and applied mitigating actions. A substantial initial inspection scope and aggressive expansion criteria (capable of driving to a 100 percent inspection) were required. The inspection sample was to be biased toward those welds with the highest propensity for cracking. It was further noted that the inspection sample should be based on operating experience and that other factors that could increase the susceptibility to IGSCC, such as weld preparation, excessive grinding, extensive repairs, and high stress locations, be considered in developing the inspection program. As a result of implementing mitigating measures or of replacing piping, the scopes and frequencies of inspection may have changed (within the bounds of the guidance in GL 88-01 and its supplement); however, ultrasonic inspections remain a key aspect of managing IGSCC.

8 FLAW EVALUATION METHODS

Early versions of Section XI of the ASME Code included flaw evaluation criteria in Table IWB 3514-3. These criteria allowed only small flaws (e.g., essentially 10 percent of the wall thickness or less) to remain in service. In the Winter Addenda to the 1983 edition of the ASME Code, a new section, IWB 3640, was added to the code whereby larger flaws could be analyzed and accepted for continued service in austenitic piping. The fracture mechanics analyses in IWB 3640 for austenitic stainless steels were based on limit load analyses. IWB 3640 also requires that the potential flaw growth during the period of operation be included in the analysis. In NUREG-0313, Revision 2, the NRC staff presented fairly detailed guidance on how to analyze IGSCC cracks for continued service. Two of the more important analysis areas addressed in Revision 2 of NUREG-0313 and in GL 84-11 were (1) uncertainties in weld residual stresses and crack growth rates and (2) uncertainties in UT sizing.

To address the uncertainties in crack growth rate, NUREG-0313, Revision 2, established a residual stress distribution and stress-intensity crack-growth relationship that were expected to provide a conservative estimate of IGSCC growth rate during the analyzed operating period.

The staff also established an acceptance criterion in GL 84-11for continued operation of the plant without having to perform a repair of the piping system. In order to address UT measurement uncertainties, this criterion limited the calculated flaw dimensions at the end of the operating period to be within two-thirds of the acceptable flaw dimensions listed in IWB 3640. This assumption allows for UT sizing uncertainty up to 100 percent of the crack depth for cracks up to 25 percent of the wall thickness. In NUREG-0313, Revision 2, uncertainties for UT sizing were considered only if the ultrasonic examination was performed by personnel not fully qualified in accordance with the NRC, EPRI, and BWROG NDE Coordination Plan, or if limitations to the examination (e.g., wide weld crowns), obstructions, or other adverse geometric configurations existed. For such examinations, the flaw was analyzed assuming the depth was at least 75 percent of the pipe wall thickness.

For low-toughness materials such as fluxed welds fabricated using shielded metal arc welding (SAW) or submerged metal arc welding (SMAW), additional criteria to evaluate flaws in such materials are given in IWB 3640 of the 1989 edition of the ASME Code. This is because for these less tough materials, crack extension and pipe failure may occur at load levels below fully plastic limit load. IWB 3640 includes factors to account for the lower fracture toughness of these materials. It also includes the effect of pipe size and secondary stresses in the flaw evaluation of such materials.

9 REPAIR METHODS

IGSCC in BWR piping initiates at the inner surface of the pipe near the weld root. The cracks commonly propagate along the heat-affected zone (HAZ) adjacent to the weld toward the outside diameter surface of the pipe. Therefore, it is not practical to perform code repair of cracked welds because the normal code repair requires grinding out the defective area and backfilling the area with weld metal.

In NUREG-0313, Revision 2, the staff identifies the following repair methods:

- reinforcement by weld overlay
- partial replacement
- SI (for minor circumferential cracks with crack depths less than 30 percent of the wall thickness, crack lengths less than 10 percent of the piping circumference, and service stress less than the allowable S_m value cited in the ASME Code)
- approved clamping devices

Reinforcement using weld overlays consists of applying weld metal on the outside diameter of the pipe over the weld and beyond the weld on both sides. This is done completely around the circumference of the pipe. IGSCC-resistant low-carbon, high-ferrite, type 308L weld metal is used, and the process is usually performed with an automatic welding machine using the gas tungsten arc weld (GTAW) or gas metal arc weld (GMAW) process. Weld overlaying can be performed with water in the pipe during welding, without the need for draining the pipe during repair.

A limit load analysis is used to determine the minimum acceptable thickness of the weld overlay design. The criteria for the limit load analysis is given in IWB 3640 of Section XI of the ASME Boiler and Pressure Vessel Code. NRC accepts two approaches for weld overlay designs: (1) the standard overlay design and (2) specially designed overlay. The standard overlay design provides a nominal margin of 2.77 against limit load failure, assuming that the crack goes completely (360 degrees) around the circumference and through the wall of the pipe. For minor cracking extending less than about 10 percent of the pipe's circumference, with no more than four axial cracks, a specially designed overlay can be used for repair. Credit for part of the uncracked area of the original pipe is taken in the special overlay design. The special overlay design also extends 360 degrees around the pipe circumference but may be thinner than the standard overlay. One application of the special overlay design is to prevent leakage from minor axial cracking.

The implementation of weld overlay repair has been delineated in Code Case N-504 as an acceptable alternative for repair of Class 1, 2, and 3 austenitic stainless steel piping.

In accordance with GL 88-01, the cracked welds reinforced by weld overlays or mitigated by SI are required to be inspected every two refueling cycles to ensure that the structural integrity of the repaired or mitigated weld joint is maintained.

10 PIPING REPLACEMENT

As discussed in Chapter 2, IGSCC can be avoided through a proper selection of materials and fabrication processes. In NUREG-0313, Revision 2, the staff made specific recommendations regarding the selection of materials and processes to be used when replacing the original IGSCC-susceptible piping materials.

In NUREG-0313, the staff provided the following guidance for selecting materials that would be considered resistant to sensitization, and therefore to IGSCC:

- Base metals of low carbon wrought austenitic stainless steels with designations 304L, 304NG, 316NG, and similar low carbon grade steels with a maximum carbon content of 0.035 percent were considered to be IGSCC resistant.

- Type 347 austenitic stainless steel, as modified for nuclear use, would be considered to be resistant with a somewhat higher carbon content, the usual maximum of 0.04 percent being an adequate limit for carbon content.

- Austenitic stainless steel materials not meeting these criteria were considered to be resistant if given a solution heat treatment after welding.

- Low carbon weld metals, including types 308L, 316L, 309L, and similar grades, with a maximum carbon content of 0.035 percent and a minimum of 7.5 percent (or FN) ferrite, as deposited, were considered resistant. Low carbon weld filler material especially developed for joining modified type 347 was considered resistant.

- Welds joining resistant materials that meet the ASME Code requirement of 5 percent (or FN) ferrite, but that are below 7.5 percent, were to be evaluated on a case by-case-basis considering carbon content and other factors affecting their sensitivity.

- Piping weldments were considered resistant to IGSCC if the weld heat-affected zone on the inside of the pipe is protected by a cladding of resistant weld metal, often referred to as "corrosion-resistant" cladding.

- Cast austenitic stainless steels with a maximum of 0.035 percent carbon and a minimum of 7.5 percent (or FN) ferrite were considered resistant. Weld joints between resistant piping and cast valve or pump bodies that do not meet these requirements were subject to more inspection.

- Other austenitic materials, including nickel-based alloys such as Inconel 600, were to be evaluated on a case-by-case basis. Inconel 82 was the only commonly used nickel-based weld material considered to be resistant.

The NRC staff also recommended in NUREG-0313, Revision 2, that no austenitic material be considered resistant to IGSCC in the presence of a crevice, such as is formed by a partial penetration weld, where the crevice is exposed to reactor coolant.

In NUREG-0313, Revision 2, the staff discussed several process controls that may be used to mitigate a material's susceptibility to IGSCC. The following processes were considered to be qualified for providing resistance to IGSCC in BWR pipe welds:

(1) solution heat treatment (SHT)
(2) heat sink welding (HSW)
(3) induction heating stress improvement (IHSI)
(1) mechanical stress improvement process (MSIP)

1: LEAK DETECTION AND LEAK-BEFORE-BREAK

As noted in the introduction to this report, the NRC has not accepted leak-before-break (LBB) as a regulatory basis for assuring the integrity of piping systems susceptible to IGSCC. The NRC's position on LBB is established in Criterion 4 to 10 CFR Part 50 Appendix A, in the associated statements of consideration (SOCs) for this Criterion, and in draft Standard Review Plan (SRP) Section 3.6.3. In total, these documents establish the position that for piping systems that are demonstrated to satisfy LBB criteria, the dynamic effects (e.g., blowdown loads, pipe whip reactions, etc.) associated with high energy pipe breaks can be eliminated from the design basis. However, even when LBB criteria are satisfied, the design basis for emergency core cooling systems and containment are not changed. Furthermore, the NRC's position is that piping systems that are subject to active degradation modes (e.g., IGSCC, fatigue, or erosion-corrosion) do not satisfy the criteria for application of LBB. Therefore, LBB does not provide a regulatory basis for assuring the integrity of BWR piping. NRC's approach has been one of defense-in-depth but with particular emphasis on inspection in order to avoid challenges to operators and safety systems.

Nonetheless, NRC does recognize that LBB can be an important element in a defense-in-depth approach to assuring piping integrity. Operating experience shows that careful monitoring for leakage and prompt corrective actions can identify conditions that, if left unaddressed, could result in pipe failure. Therefore, the NRC has established stringent requirements for monitoring leakage in BWRs.

In NUREG-1061, Volume 1, the fourth Pipe Crack Study Group recommended a decrease in allowable unidentified leakage in BWRs from 1140 L/hr (5 gpm) to 684 L/hr (3 gpm). However, upon further evaluation, the staff concluded that the existing leakage limits, if effectively implemented along with other recommendations in the areas of inspection, and so forth, would provide an acceptable level of protection, and that the reduction in the leakage limit could not be supported by a regulatory analysis performed in accordance with the "Backfit Rule" (e.g., 10 CFR 50.109.c).

In NUREG-0313, Revision 2, and in Generic Letter 88-01, the NRC staff established the position that leakage detection systems should be in conformance with Position C of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," or else as otherwise approved by the NRC. The following items summarize this position in more detail:

- Plant shutdown should be initiated for inspections and corrective action when, within any period of 24 hours or less, any leakage detection system indicates an increase in rate of unidentified leakage in excess of 456 L/hr (2 gpm) or its equivalent, or when the total unidentified leakage attains a rate of 1140 L/hr (5 gpm) or equivalent, whichever comes first. For sump level monitoring systems with fixed measurement intervals, the level should be monitored at approximately 4-hour intervals or less.

- Unidentified leakage should include all leakage other than the following:

  (a) leakage into closed systems, such as pump seal or valve packing leaks that are captured, flow metered, and conducted to a sump or collecting tank, or

  (b) leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operations of unidentified leakage monitoring systems or not to be from a through wall crack in the piping within the reactor coolant pressure boundary.

- For plants operating with any IGSCC Category D, E, F, or G welds (i.e., welds with a high susceptibility to IGSCC, at least one of the leakage measurement instruments associated with each sump shall be operable, and the outage time for inoperable instruments shall be limited to 24 hours, or immediately initiate an orderly shutdown.

The preceding position emphasizes several important aspects that are necessary in order to identify and take corrective action for defective piping systems that do exhibit LBB. These are (1) timely leakage measurements, (2) criteria related to rate of increase in leakage, and (3) prompt corrective action. In order for LBB to be effective, leakage must be measured on a timely basis (i.e., as close to real time as possible). Many systems that exhibit LBB show large (e.g., exponential) increases in leakage as the defect becomes more significant. Delays in measurements or monitoring of leakage could result in loss of opportunity for timely corrective action. Similarly, criteria that are based solely on magnitude of leakage may not recognize a rapidly developing situation. Monitoring the trend in leakage makes better use of available data and provides the opportunity for more timely action. Finally, operators must have clear procedures for responding to increasing unidentified leakage. Rapid, decisive response is critical in taking advantage of LBB to avoid gross component failure.

12 CURRENT STATUS OF BWR PIPING

In Table 12.1, the staff presents the status of piping in operating U.S. BWRs. The table indicates whether full or partial replacements of recirculation and RHR system piping has occurred and whether or not the unit has implemented hydrogen water chemistry.

As indicated in Table 12.1, 12 BWR plants in the U.S. have replaced either all or part of the main recirculation system stainless steel piping with low carbon stainless steel materials. The materials used in the piping replacement are either 316NG or 316SS with low carbon content. Eight plants have replaced all the piping in the recirculation and RHR systems. One plant has replaced all the piping in the recirculation system and a portion of the piping in the RHR system. Three plants have replaced only the riser portion of the recirculation system piping.

Hydrogen water chemistry (HWC) has been applied at 16 BWR plants.

At least 20 domestic BWRs have found IGSCC in large-diameter stainless steel piping. Except for very minor cracking, almost all cracked welds inside the containment were repaired with weld overlays. However, in general, operating BWRs have not operated with cracked welds for more than one or two fuel cycles, because repair or mitigation has the benefit of requiring less inspection. Those plants that have not replaced all susceptible piping or that did not startup with resistant piping are currently operating with numerous overlay repaired welds.

Since the staff issued GL 88-01, the U.S. industry has been performing inspections for IGSCC on a regular basis. The inspection schedule for each weld is based on the degree of susceptibility to IGSCC, as recommended in the guidance of GL 88-01 and its supplement. In the early rounds of inspection, many large pipe welds were found to be cracked. However, as discussed in Section 4.0, as a result of piping replacements, repairs, and implementation of other measures to reduce the sensitivity to IGSCC, there have been very few recently reported events of IGSCC in U.S. BWRs.

Table 12.1 Status of U.S. BWR Piping

Plant		Date of Operating License	Original Design Material	Replacements for Recirculation Piping	Replacements for RHR Piping[b]	Hydrogen Water Chemistry Implemented?
Design	Name					
BWR2	Nine Mile Point 1	12/26/74	304SS or 316SS	Full, 316SS (low carbon)	Part, 316SS (low carbon)	Yes
	Oyster Creek	08/01/69	304SS or 316SS	None	None	Yes
BWR3	Dresden 2	02/21/70	304SS or 316SS	None	None	Yes
	Dresden 3	03/02/70	304SS or 316SS	Full, 316NG	Full, 316NG	No
	Millstone 1	10/31/86	304SS or 316SS	None	None	Yes
	Monticello	01/09/81	304SS or 316SS	Full, 316NG	Full, 316NG	Yes
	Pilgrim	09/15/72	304SS or 316SS	Full, 316NG	Full, 316NG	Yes
	Quad Cities 1	12/14/72	304SS or 316SS	None	None	Yes
	Quad Cities 2	12/14/72	304SS or 316SS	None	None	Yes
BWR4	Browns Ferry 1	12/20/73	304SS or 316SS	None	None	No
	Browns Ferry 2	08/02/74	304SS or 316SS	Part[a] (riser) 316NG	None	No
	Browns Ferry 3	08/18/76	304SS or 316SS	None	None	No
	Brunswick 1	11/12/76	304SS or 316SS	Part[a] (riser) 316NG	None	Yes
	Brunswick 2	12/27/74	304SS or 316SS	Part[a] (riser) 316NG	None	Yes

Plant		Date of Operating License	Original Design Material	Replacements for Recirculation Piping	Replacements for RHR Piping[b]	Hydrogen Water Chemistry Implemented?
Design	Name					
BWR4 cont.	Cooper	01/18/74	304SS or 316SS	Full, 316NG	Full, 316NG	No
	Duane Arnold	02/20/74	304SS or 316SS	None	None	Yes
	Fermi 2	07/15/85	304SS or 316SS	None	None	Yes
	FitzPatrick	10/17/74	304SS or 316SS	None	None	Yes
	Hatch 1	10/13/74	304SS or 316SS	None	None	Yes
	Hatch 2	06/13/78	304SS or 316SS	Full 316NG	Full 316NG	Yes
	Hope Creek	07/25/86	316NG REC, RHR RWCU	N/A	N/A	Yes
	Limerick 1	08/08/85	316NG REC, RHR, Core Spray, RWCU	N/A	N/A	Yes
	Limerick 2	08/25/89	316NG REC, RHR, Core Spray, RWCU	N/A	N/A	Yes
	Peach Bottom 2	12/14/73	304SS or 316SS	Full 316NG	Full 316NG	Yes
	Peach Bottom 3	07/02/74	304SS or 316SS	Full 316NG	Full 316NG	Yes
	Susquehanna Unit 1	11/12/82	304SS or 316SS	None	None	Yes
	Susquehanna Unit 2	06/27/84	304SS or 316SS	None	None	No
	Vermont Yankee	02/28/73	304SS or 316SS	Full 316NG	Full 316NG	No

Plant		Date of Operating License	Original Design Material	Replacements for Recirculation Piping	Replacements for RHR Piping[b]	Hydrogen Water Chemistry Implemented?
Design	Name					
BWR5	La Salle 1	08/13/82	304SS or 316SSL[c]	None	None	No
	La Salle 2	03/23/84	304SS or 316SSL[c]	None	None	No
	Nine Mile Point 2	07/02/87	316NG for All Piping Systems	N/A	N/A	No
	WNP 2	04/13/84	304SS or 316SS	None	None	No
BWR6	Clinton 1	04/17/87	316NG for REC, RWCU	N/A	None	No
	Grand Gulf 1	11/01/84	304SS or 316SS	None	None	Yes
	Perry 1	11/13/86	304SS or 316SS	None	None	No
	River Bend 1	11/20/85	316NG for REC	N/A	None	No

Notes:

(a) Recirculation system riser piping only.
(b) Residual Heat Removal piping inside containment that is classified as ASME Code Class 1 pipe.
(c) 12 inch inlet safe-ends

Abbreviation Descriptions:
Full - full replacement of the piping
Part - partial replacement of the piping
304SS - Type 304 austenitic stainless steel
316SS - Type 316 austenitic stainless steel
316NG - Type 316 austenitic stainless steel, nuclear grade quality
None - no replacement of the piping performed to date
N/A - initial material of the piping is already Type 316NG steel; replacement is not applicable in this case
REC - Recirculation SystemPiping
RWCU - Reactor Water Cleanup System Piping
RHR - Residual Heat Removal System Piping

13 CONCLUSIONS

Development of the U.S. regulatory framework for managing IGSCC in BWR piping was an evolutionary process driven by operating experience, research, and technological developments. IGSCC was initially observed in only small-diameter piping; however, in time it affected both large- and small-diameter piping systems.
A key element in the management of IGSCC was an aggressive inspection program utilizing qualified inspection techniques and personnel. In parallel, the development of qualified repair and mitigation methods, along with replacement of piping with IGSCC-resistant materials resulted in an effective and economic solution to the problem. Today, a combination of regulatory documents, code rules, and industry guidelines are available to support effective programs for managing IGSCC in the piping of U.S. BWRs.

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APPENDIX A

ABBREVIATIONS

Chemistry Abbreviations

Cr	Chromium
Ir-192	Irridium Isotope 192
Co-60	Cobalt Isotope 60

Terminology Abbreviations

ASME	American Society of Mechanical Engineers
BWR	boiling water reactor
BWROG	Boiling Water Reactor Owners Group
CRD	control rod drive
ECCS	emergency core cooling system
EPRI	Electrical Power Research Institute
FN	ferrite number
GE	General Electric Company
GL	Generic Letter
GMAW	gas metal arc welding
GTAW	gas tungsten arc welding
HAZ	heat-affected zone
HPCS	high pressure coolant spray
HSW	heat sink weld
HWC	hydrogen water chemistry
IE	Office of Inspection and Enforcement
IGSCC	intergranular stress corrosion cracking
IHSI	induction heating stress improvement
ISI	inservice inspection

LBB	leak-before-break
LPCI	low pressure coolant injection
LPHSW	last pass heat sink welding
LPCS	low pressure coolant spray
MSIP	mechanical stress improvement process
NDE	nondestructive examination
PDI	performance demonstration initiative
PWR	pressurized-water reactor
RCIC	reactor core isolation cooling
RG	Regulatory Guide
RHR	residual heat removal
RT	Radiography Testing
RWCU	reactor water cleanup
SAW	submerged arc welding
SHT	solution heat treatment
SLC	standby liquid control
SMAW	shielded metal arc welding
SOC	statement of consideration
SRP	standard review plan
UT	ultrasonic testing

Unit Abbreviations

gpm	gallons per minute, English System unit of flow
L/hr	liters per hour, SI Metric System unit of flow
MPa	megapascal, SI Metric System unit of stress or pressure
ppb	parts per billion, a unit of concentration
psi	pounds per square inch, English System unit of stress or pressure
°C	degrees Celsius, Metric System unit of temperature
°F	degrees Fahrenheit, English System unit of temperature
µS/cm	micro-Siemens per centimeter, a metric unit for the measurement of conductivity
MHz	megahertz, a unit for frequency equal to one million cycles per second

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This paper was prepared, in part, to support discussions at an IAEA Regional Workshop on Environmentally Assisted Cracking of Nuclear Power Plant Austenitic Piping that was held in Slavutych, Ukraine from 22 to 26 June 1988. This paper presents a regulatory history and offers a perspective on intergranular stress corrosion cracking (IGSCC) in U.S. boiling-water reactor piping. The paper focuses on regulatory and industry actions taken to assure that U.S. licensees manage IGSCC in a manner that provides safe and reliable plant operation. Although the paper does not offer extensive theoretical details on the IGSCC phenomenon, it does discuss some of the key technical issues that influenced regulatory positions and industry actions.

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boiling-water reactor (BWR); intergranular stress corrosion cracking (IGSCC); recirculation system piping; residual heat removal (RHR) system piping; reactor water cleanup (RWCU) system piping; leak-before-break (LBB); austenitic stainless steel; fracture toughness; ultrasonic testing (UT); non-destructive examination (NDE); weld overlay; stress improvement (SI); hydrogen water chemistry (HWC)

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