ELECTRIC POWER
RESEARCH INSTITUTE

Nondestructive Evaluation: Historical View of Nondestructive Evaluation for Stress Corrosion Cracking at Nuclear Power Plants

1021159

Nondestructive Evaluation: Historical View of Nondestructive Evaluation for Stress Corrosion Cracking at Nuclear Power Plants

1021159

Technical Update, September 2010

EPRI Project Manager

B. Rassler

Work to develop this product was completed under the EPRI Nuclear Quality Assurance Program in compliance with 10 CFR 50, Appendix B and 10 CFR 21,

YES [✓ NO]

ELECTRIC POWER RESEARCH INSTITUTE
3420 Hillview Avenue, Palo Alto, California 94304-1338 • PO Box 10412, Palo Alto, California 94303-0813 • USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com

DISCLAIMER OF WARRANTIES AND LIMITATION OF LIABILITIES

THIS DOCUMENT WAS PREPARED BY THE ORGANIZATION(S) NAMED BELOW AS AN ACCOUNT OF WORK SPONSORED OR COSPONSORED BY THE ELECTRIC POWER RESEARCH INSTITUTE, INC. (EPRI). NEITHER EPRI, ANY MEMBER OF EPRI, ANY COSPONSOR, THE ORGANIZATION(S) BELOW, NOR ANY PERSON ACTING ON BEHALF OF ANY OF THEM:

(A) MAKES ANY WARRANTY OR REPRESENTATION WHATSOEVER, EXPRESS OR IMPLIED, (I) WITH RESPECT TO THE USE OF ANY INFORMATION, APPARATUS, METHOD, PROCESS, OR SIMILAR ITEM DISCLOSED IN THIS DOCUMENT, INCLUDING MERCHANTABILITY AND FITNESS FOR A PARTICULAR PURPOSE, OR (II) THAT SUCH USE DOES NOT INFRINGE ON OR INTERFERE WITH PRIVATELY OWNED RIGHTS, INCLUDING ANY PARTY'S INTELLECTUAL PROPERTY, OR (III) THAT THIS DOCUMENT IS SUITABLE TO ANY PARTICULAR USER'S CIRCUMSTANCE; OR

(B) ASSUMES RESPONSIBILITY FOR ANY DAMAGES OR OTHER LIABILITY WHATSOEVER (INCLUDING ANY CONSEQUENTIAL DAMAGES, EVEN IF EPRI OR ANY EPRI REPRESENTATIVE HAS BEEN ADVISED OF THE POSSIBILITY OF SUCH DAMAGES) RESULTING FROM YOUR SELECTION OR USE OF THIS DOCUMENT OR ANY INFORMATION, APPARATUS, METHOD, PROCESS, OR SIMILAR ITEM DISCLOSED IN THIS DOCUMENT.

THE FOLLOWING ORGANIZATION PREPARED THIS REPORT:

Electric Power Research Institute (EPRI)

THE TECHNICAL CONTENTS OF THIS DOCUMENT WERE **NOT** PREPARED IN ACCORDANCE WITH THE EPRI QUALITY PROGRAM MANUAL (WHICH FULFILLS THE REQUIREMENTS OF 10 CFR 50, APPENDIX B AND 10 CFR 21, ANSI N45.2-1977 AND/OR THE INTENT OF ISO-9001 (1994)). USE OF THE CONTENTS OF THIS PRODUCT IN NUCLEAR SAFETY OR NUCLEAR QUALITY APPLICATIONS REQUIRES COMMERCIAL GRADE DEDICATION OR ADDITIONAL ACTIONS BY THE RECEIVING ORGANIZATIONS.

This is an EPRI Technical Update report. A Technical Update report is intended as an informal report of continuing research, a meeting, or a topical study. It is not a final EPRI technical report.

NOTE

For further information about EPRI, call the EPRI Customer Assistance Center at 800.313.3774 or e-mail askepri@epri.com.

Electric Power Research Institute, EPRI, and TOGETHER…SHAPING THE FUTURE OF ELECTRICITY are registered service marks of the Electric Power Research Institute, Inc.

Copyright © 2010 Electric Power Research Institute, Inc. All rights reserved.

ACKNOWLEDGMENTS

The following organization prepared this report:

Electric Power Research Institute (EPRI)
1300 West W.T. Harris Blvd.
Charlotte, NC 28262

Principal Investigator
B. Rassler

This report describes research sponsored by EPRI.

This publication is a corporate document that should be cited in the literature in the following manner:

Nondestructive Evaluation: Historical View of Nondestructive Evaluation for Stress Corrosion Cracking at Nuclear Power Plants. EPRI, Palo Alto, CA: 2010. 1021159.

ABSTRACT

The commercial nuclear power industry has been dealing with stress corrosion cracking (SCC) for almost 50 years. A tremendous and broad-ranging amount of nondestructive evaluation (NDE) experience has transpired, from thin-wall tubing in steam generators to thick-wall dissimilar metal piping welds connecting to the reactor pressure vessel (RPV). However, the industry might not be aware of the extent of the work that has been conducted in the United States and other countries.

In the United States, the use of alternative SCC flaws for dissimilar metal welds (DMWs), weld overlay (WOL), and other performance demonstration applications has long been accepted by the regulatory authorities. Outside the United States, where DMW and WOL performance demonstration programs are now being introduced, utilities are also realizing benefits of using alternative flaws to represent SCC. It would be beneficial for these utilities' DMW and WOL performance demonstration applications to allow the use of non-SCC flaws (alternative flaws) that respond to NDE in a similar way as actual field-removed SCC and give realistic NDE signals when compared with field-removed SCC.

The objective of this project in its first year is to conduct a global literature review across the nuclear industry with a focus on the historical NDE inspection issues of SCC. This information has been condensed into this report, which can be used as a reference by the industry. It includes why, when, and where SCC occurs as well as how the industry and the regulators have reacted. It focuses on studying the issues of SCC occurrences and how the industry reacted in its use of NDE. These issues are listed chronologically. This information can be used as a reference to determine whether an SCC issue has been addressed by the nuclear industry.

The second year of this project will capture and compare NDE response from components in the field, when available, with NDE responses from mockups used for NDE training and demonstration and collect this in a separate reference manual.

Keywords
Alloy 600
Intergranular stress corrosion cracking (IGSCC)
Primary water stress corrosion cracking (PWSCC)
Ultrasonic testing (UT)
Visual testing (VT)
Stress Corrosion Cracking (SCC)

CONTENTS

LIST OF TABLES

1

INTRODUCTION

Background

The commercial nuclear power industry has been dealing with stress corrosion cracking (SCC) for almost 50 years. A tremendous and broad ranging amount of nondestructive evaluation (NDE) experience has transpired; from thin-wall tubing in steam generators to thick-wall dissimilar metal piping welds connecting to the reactor pressure vessel (RPV). However, the industry might not be aware of the extent of the work that has been conducted in the United States and other countries. Globally, utilities are looking for an SCC reference that describes the NDE experience with SCC. They can then use this information to support a technical basis and technical justification in support of performance demonstrations.

In the United States, the use of alternative SCC flaws for dissimilar metal welds (DMWs), weld overlay (WOL), and other performance demonstration applications has long been accepted by the regulatory authorities. Outside the United States, where DMW and WOL performance demonstration programs are now being introduced, utilities are also realizing benefits of using alternative flaws to represent SCC. It would be beneficial for the non-U.S. DMW and WOL performance demonstration applications to allow the use of non-SCC flaws (alternative flaws) that respond to NDE in a similar way as actual field-removed SCC and give realistic NDE signals when compared with field-removed SCC.

Objectives

The objective of this project in its first year is to conduct a global literature review across the nuclear industry with a focus on the historical NDE inspection issues of SCC. This information has been condensed into this report that can be used as a reference by the industry. It includes why, when, and where SCC occurs as well as how the industry and the regulators have reacted, which NDE techniques have been applied with their results when available, and where the industry is today because of SCC. It focuses on studying the issues of SCC occurrences and how the industry reacted in its use of NDE. These issues are listed chronologically. This information can be used as a reference to determine whether an SCC issue has been addressed by the nuclear industry and the reaction of the regulators and how the NDE inspections improved.

The second year of this project will capture and compare NDE response from components in the field, when available, with NDE responses from mockups used for NDE training and demonstration and collect this in a separate reference manual.

Organization

A short description of how SCC occurs is given in Section 2. This report describes some of the information that has been discovered over the last 30 to 40 years concerning SCC flaws, including; inspection procedures on components that have been susceptible to these types of flaws and how the mockups used to train and test the inspectors have been developed. Section 3 gives a historical overview of the history of SCC and includes the following:

- Discovery of SCC in nuclear plants
- Development of training mockups with SCC and alternative flaws to demonstrate NDE effectiveness
- NDE training and testing on SCC and alternative flaws prior to inspection
- NDE detection of SCC at nuclear plants

Section 4 condenses the NDE history of SCC into a timeline that makes it easier to determine when a particular issue arose and how it was addressed by the industry and the regulators and how NDE adapted to address it.

Section 5 contains the references used to support this report along with a bibliography of related industry experience not referenced in this report.

2
WHAT IS STRESS CORROSION CRACKING?

Why Stress Corrosion Cracking Occurs

SCC occurs when the effects of material stress and environment on the susceptible material are combined in the various specific measures to make it happen. The combination of three factors—tensile stress, corrosive environment, and susceptible material—are necessary for SCC to occur. SCC has occurred in as short a time as 1 year to as much as more than 30 years in operating nuclear reactors. The time to initiation depends on the extent of the three factors that are at work.

When Stress Corrosion Cracking Occurs

Tensile Stresses

Applied stresses required to induce SCC must be tensile and of sufficient magnitude. The most common cause for residual stress is welding, which is followed by a slow cooling through the sensitizing temperature range. The welded material can contain residual stresses that might approach the material yield strength. These tensile stresses alone might be sufficient to induce SCC in the sensitized material in an aggressive environment. In general, increasing the applied stress level decreases the time to crack initiation. Additional contributors to tensile stresses could include repairs or operating stresses; whether induced by mechanical or thermal influences.

Corrosive Environment

Several conditions can make the environment more corrosive to the susceptible materials. A corrosive environment can be caused by an electrochemical process, which depends on the electrolytic nature of the environment. Impurities in the reactor coolant can provide the electrolytic environment necessary to support SCC. Increasing the operating temperature of the reactor increases the corrosive environment and decreases the time to crack initiation in the material. Exposure to high levels of neutron flux can also cause materials to become susceptible to SCC; this is a special form of SCC known as irradiation assisted stress corrosion cracking (IASCC), which has the potential to occur in Boiling Water Reactors (BWRs) and Pressurized Water Reactors (PWRs) RPV internal components.

Susceptible Material

Material susceptibility is related to the environment and can be influenced by the metallurgical condition of the material. For example, sensitized materials are more susceptible to SCC than fully annealed materials. The most common way for the material to become sensitized is welding, followed by a slow cooling through the sensitizing temperature range. Cold working without heat treatment also makes the material susceptible to SCC. In general, the less chromium in the material, the greater its susceptibility to SCC. It has been found that when a material that has been properly heat treated with a chromium content greater than 20%, it will be less susceptible to SCC.

Where Stress Corrosion Cracking Occurs

Stress corrosion cracks usually propagate perpendicular to the applied tensile stress. Cracks propagate with little or no plastic deformation and vary in degree of branching or formation of satellite cracks. Variation in crack morphology depends on environment and material microstructure as well as the type and orientation of the applied stresses.

Susceptibility to SCC varies from one alloy to another and with the metallurgical condition of the material under consideration. The original piping material in the U.S. BWRs was of type 304 or 316 austenitic stainless steel (SS), which can have less than 20% chromium content [1]. SCC in BWRs is referred to as *intergranular stress corrosion cracking* (IGSCC). Alloy 600 base metal and weld Alloy 182 both have less than 20% chromium, while Alloy 82 usually has greater than 20% chromium and is resistant to SCC [2]. Alloy 82 and Alloy 182 are both used in vessel attachment welds in BWR and PWR reactor vessels. Generic Letter (GL) 88-01 [3] indicates that Alloy 82 is resistant to SCC but that Alloy 182 is susceptible. SCC in a PWR is normally referred to as *primary water stress corrosion cracking* (PWSCC).

3

HISTORICAL OVERVIEW

3.1 Discovery of Stress Corrosion Cracking in Nuclear Power Plants

Safe operation of nuclear utility reactors in the United States is the driving force behind all NDE in these facilities. The outstanding safety record of this industry directly attests to the success of these efforts. NDE technologies are constantly improving and being upgraded to accommodate evolving material technology. Owners and operators of nuclear power plants in the United States are required by law to perform periodic in-service inspection (ISI) of components of the primary pressure boundary. Basic inspection requirements (in terms of specific components), frequency of inspection, coverage, and so on, are detailed in the ASME Code, Section XI [4].

The first detected occurrence of IGSCC was in one of the secondary heat exchanger tubes in the Vallicitos BWR in Pleasanton, California, in 1959. The first occurrence of IGSCC in a commercial supplied BWR was at Dresden 1 located near Morris, Illinois, in 1965. The Dresden 1 IGSCC occurred in a type 304 SS 6-in. (152.4-mm) bypass line in the weld heat-affected zone (HAZ). Until 1974, very few incidents of IGSCC were reported or detected in BWR plant piping systems. However, in 1974 and 1975, IGSCC was detected in nine plants, either in the 4-in. (101.6-mm) diameter recirculation bypass piping or in the 10-in. (254-mm) schedule 80 core spray piping. The occurrence of IGSCC in SS lines had previously always been found in systems that had low flow or were stagnant during normal plant operation. In late 1977, this was found to no longer be true when IGSCC was detected at three non-U.S. BWR plants in recirculation riser piping—systems with high flow during normal plant operation [5].

Historically, NDE of BWR SS pipe has been associated with the incidence of IGSCC in these components. In September of 1974, the U.S. Atomic Energy Commission (AEC) ordered that 15 BWR plants be shut down to inspect their 4-in. (101.6-mm) piping after cracking was found in similar piping at three other BWR plants [6]. In 1975, the newly formed U.S. Nuclear Regulatory Commission (NRC) ordered that 23 BWR plants be shut down for inspection after cracking was discovered in larger 10-in. (254-mm) diameter piping at Dresden 2 [7].

3.2 Industry Actions and Initiatives (1979–1990)

A group of BWR utilities organized an owners group in late 1979 that commenced funding R&D programs in January 1980. The Electric Power Research Institute (EPRI) was assigned the responsibilities of integrating these resources into a single, unified program combined with EPRI's then ongoing baseline program under way since 1975. The resulting program was designated as the BWR Owners Group (BWROG) for Intergranular Stress Corrosion Cracking Research Program. The detailed name was necessary to distinguish this program from other BWROG programs; however, in this report, the use of BWROG refers to the program focused on IGSCC. Comparisons of the 1975, 1978, and 1979 data showed that the pipe cracking continued to occur and that larger pipes were becoming affected. When the cracking started occurring in piping that was part of the primary coolant recirculation system, the NRC took an active interest because of the potential safety implications [8].

Until the discovery of through-wall IGSCC in two safe ends at Nine Mile Point Unit 1 (NMP-1) located near Syracuse, New York, in March 1982, traditional shear-wave pulse-echo (PE) ultrasonic testing (UT) inspection of the weld area had been the accepted NDE technology for these components. However, this event cast doubt on traditional ultrasonic capability because the areas found leaking from through wall IGSCC had been inspected ultrasonically only nine months prior to this incident. Because of this, the NRC issued Inspection and Enforcement (IE) Bulletin 82-03 [9] to provide a reasonable level of assurance that state-of-the-art inspections were sufficient to detect cracking in BWR thick-wall recirculation piping welds and to determine the generic significance of the NMP-1 piping degradation.

The issuance of IE Bulletin 82-03 set into motion a series of events that provided the environment and framework to create a unique relationship between BWROG, EPRI, and the NRC. BWROG's previously established IGSCC program gave it the ability to respond immediately through EPRI to answer the questions of the NRC bulletin and to outline a program to correct deficiencies in ultrasonic detection and sizing capabilities. The NRC, recognizing the quality of the proposed program and the sincerity of the BWROG in correcting these deficiencies, agreed to proceed without major plant shutdowns. EPRI had the dual role of supporting the BWROG while also acting as an independent entity to provide training and qualification programs as required and overseen by the NRC. This unique relationship has continued and has resulted in significant advances in ultrasonic detection and sizing of IGSCC in SS piping and successful countermeasures to prevent or mitigate IGSCC growth.

In September and October of 1982, EPRI conducted a series of studies to prepare test specimens at Battelle Columbus Laboratories, aimed at setting up a qualification program to test NDE teams in IGSCC detection methods. Five samples of 28-in. (711.2-mm) diameter piping from NMP-1 were transported to Battelle Columbus. The field-removed IGSCC samples were cleaned and secured to ensure that the inside diameter (ID) surface was not visible to an inspector. In nine days, EPRI prepared a procedure, selected test samples, and organized a protocol for qualification of NDE teams. Each utility was invited to schedule an ISI team to perform NDE inspections of the samples, with the evaluation of the results and a rendering of pass/fail judgment to be made by NRC personnel. This successful effort resulted in the first IGSCC detection qualification test on October 8, 1982 [8]. NRC IE Bulletin 82-03 [9] required a response and testing of NDE teams from nine utilities. The NRC analysis of the data from the testing of the nine utility teams, with results from an EPRI team using advanced techniques, demonstrated that adequate performance in detecting IGSCC could be achieved, but that well-trained, experienced personnel with appropriate procedures and NDE methods were necessary. However, retesting of the teams was required to achieve this stated competence level.

In December 1982, EPRI NDE Program staff members, aided by a group of utility personnel, initiated the development of a full 40-hour course for training and qualifying ultrasonic operators in the detection and characterization of IGSCC in BWR piping. The course was characterized by a large amount of hands-on experience with realistic cracked samples, plus a series of examinations containing a practical examination that used NMP-1 samples. The EPRI NDE Program provided the IGSCC detection training course for the first time in June 1983 [8].

The mounting evidence of a generic IGSCC problem in large-diameter BWR SS piping led to the NRC issuance of NRC IE Bulletin No. 83-02 [10]. Action was required of an additional 14 licensees. Under severe pressure on all the parties involved to respond to the growing IGSCC problem, BWROG, EPRI, and the NRC demonstrated their organizational and managerial expertise and negotiated a unique plan to address this problem. Negotiated under pressure, but in an atmosphere of cooperation, a plan was developed: “Coordination Plan for the NRC/EPRI/BWROG Training and Qualification Activities of NDE Personnel,” known in the industry as, “the three party agreement” [11]. The plan provided a method in which the utility industry, through BWROG and EPRI, could overcome this problem through training, testing, and qualification of their NDE personnel—while aggressively pursuing advanced NDE technologies—and provide the NRC the necessary oversight to ensure effectiveness in the plan’s implementation. The result was a successful program that has managed the issues surrounding the IGSCC inspection problem. The IGSCC detection training courses and qualification tests offered at the EPRI NDE Program grew directly from this Coordination Plan and BWROG/EPRI efforts to meet the needs of the utility industry [8].

Axial cracking was detected in the nozzle-to-safe-end DMW during a planned safe-end replacement at the Pilgrim Nuclear Station (Boston Edison Company) in May of 1984. Subsequent evaluation of the cracks showed them to be IGSCC originating in the Inconel 182 weld material. On June 1, 1984, the NRC issued Information Notice (IN) No. 84-41 informing all BWR facilities of these events [12]. On June 8, 1984, the Technical Advisory Committee and the Executive Committee of the BWROG approved a research program to address the cracking problem. NDE and repair and replacement studies were included in the research program. The development of an improved ultrasonic examination procedure for use in detecting cracking in the nozzle-to-safe-end weld is covered in EPRI report NP-4606-LD, *Development of Improved Procedure for Examination of Dissimilar-Metal Welds in BWR Nozzle-to-Safe-End Welds* [13]. This report describes the flaws in the mockups used to develop the improved ultrasonic examination procedure. No field-removed samples of safe-end welds with IGSCC were available. The flaws used were electrodischarge-machined (EDM) notches (alternative flaws) and weld solidification flaws. The EDM notches were of various depths and lengths placed in the mockups in patterns that would not result in any notch interfering with the examination of any other notch. EPRI report NP-4606-LD describes a generic procedure for longitudinal wave ultrasonic examination of nozzle-to-safe-end welds in BWR plants. This EPRI-developed procedure was used in BWR units to successfully detect IGSCC.

In May 1990, the NRC issued IN No. 90-30 [14] that described how conventional S-wave ultrasonic examinations were being used to examine DMWs containing Inconel 600 series base materials, Alloy 82 and 182 weld butter, and/or Alloy 82 and 182 filler materials at PWR facilities. In the notice, the question was raised regarding whether PWR facilities might be using inadequate S-wave ultrasonic examination techniques for DMWs. To answer this question, an EPRI program was initiated to verify whether the procedures developed for BWR plants were also applicable to PWR designs and also to update EPRI NP-4606 with new information on NDE of DMWs in BWR plants. This new program was documented in EPRI report TR-102148, *Examination of Dissimilar Metal Welds in BWR and PWR Piping Systems* [15]. Because the samples used for the NP-4606 report were no longer available, it was determined that several available Westinghouse Owners Group (WOG) samples would be appropriate. These WOG samples contained artificial flaws of mechanical and thermal fatigue cracking. The conclusion was that the NDE procedures are applicable to both BWR and PWR plants based on the ultrasonic responses of alternative flaws and implanted cracks in mockups of BWR and PWR weld configurations and on field ultrasonic examinations on actual IGSCC in BWR piping systems.

3.3 Performance Demonstration Initiative (1991–Present)

Performance demonstration requirements were added to ASME Code, Section XI in the 1991 Addenda and are described in Appendix VIII [16]. These requirements are applicable to ultrasonic examination of piping, bolting, and selected portions of the RPV but exclude the shell-to-flange and head-to-flange welds. Appendix VIII requires that procedure effectiveness and personnel proficiency must be demonstrated on mockups containing realistic flaws. Appendix VIII differs from previous Code approaches in that it does not specify a particular approach. It does require that the capabilities of the personnel, procedures, and equipment be demonstrated, and it describes in detail the demonstration requirements and acceptance criteria for ultrasonic examinations. Recognizing the importance and complexity of Appendix VIII implementation, representatives from all U.S. nuclear utilities formed the Performance Demonstration Initiative (PDI) to implement Appendix VIII. The objective of the PDI Program is to provide the following:

- A unified industry approach to high credibility
- A generic program that minimizes the need for site-specific or repeated demonstrations by vendors or individual examiners
- A basis for negotiating implementation approaches and dates with regulatory authorities
- A lower-cost alternative through the use of combined resources
- A fair and competitive environment for ISI vendors by minimizing the cost

Uniformity of this approach is important to credibility and acceptability and to ensure that qualifications earned at one location can be applied at another. This helps to avoid the substantial cost of repeating demonstrations. The PDI program was initiated in 1991, with the first demonstrations starting in April 1994 [17].

Appendix VIII allows some flexibility in the selection of flaw types and locations. The technical committee established the rules for selecting a matrix of flaws for the PWR and BWR sample sets that are both realistic and representative. Fabrication techniques that provide accurate knowledge of the flaw size are used in accordance with EPRI developed specifications. Samples to be used for depth sizing demonstrations are fabricated using flaw-implantation techniques that provide a precise knowledge of the true flaw depth. The characteristics of the intentional flaws in the samples are a major consideration for a valid performance demonstration. A considerable portion of the sample fabrication cost is associated with flaw fabrication and verification. The PDI program has developed technical criteria for constructing samples and introducing intentional flaws that describe the following:

- Accuracy of flaw sizes and locations
- Flaw characteristics
- Use of clean base material
- Sample security

All U.S. and several non-U.S. nuclear utilities have joined the PDI program, confirming the importance placed by utilities on NDE performance demonstrations. The PDI program is organized and led by the utilities with technical and administrative support being provided by EPRI and the EPRI NDE Program [18].

3.2.1 Performance Demonstration Initiative Flawed-Sample Mockups

ASME Supplement 10 of Appendix VIII addresses dissimilar metal (DM) piping welds. The specimens used for these performance demonstrations must meet both the ASME and the Code of Federal Regulations 10CFR50.55a requirements. To determine a starting place for mockup geometries and configurations, the PDI steering committee requested that each plant submit configuration design drawings for each of their units. Survey responses were received from a large cross section of the industry. Each of the four nuclear steam system suppliers (NSSSs) was represented, and a wide range of configurations were covered. Material and configuration surveys described in earlier EPRI reports [13, 15] were also used for PDI sample set design information. The resultant selection of representative examples was based on the following:

- Field failure information
- Common shop and field weld configurations currently used by the industry
- Specific measurement of a set of developmental mockups that contain a selection of materials and configurations

This design review revealed that several configurations exist in the industry. A limited review of as-found conditions also indicates that many variations were made to the design during fabrication, adding to the complexity of selecting representative test samples. Based on this complexity and the limited resources available, it was determined by PDI that the sample sets to be fabricated would not cover every configuration but would be a sampling of representative configurations covering the largest range practical. Licensees would be required to fabricate site-specific mockups to address configurations not covered in the sample set. The list of mockups encompassed by PDI is extensive in size—ranging from piping mockups that can be carried by hand to WOL mockups, DMW mockups, and vessel mockups that weigh several tons. A comprehensive overview of the flawed-sample mockups used in the PDI program can be located in the EPRI NDE Program mockup catalog [19].

3.2.2 Performance Demonstration Initiative Service-Related Experience and Assistance

This section contains service-related experience from the mid 1980s through September 2009 and is a selection from EPRI report 1019134, *Dissimilar Metal Piping Weld Examination – Guidance and Technical Basis for Qualification: Volume 7* [20]. EPRI NDE Program personnel have been involved in the resolution of many DM piping weld failures in the United States. In many cases, the staff obtained recorded ultrasonic data from these flaws. The sizes of these flaws are less well known because empirical data points are on only two PWR incidents and one BWR incident that have been subjected to destructive testing. In several cases, the flaws in question leaked, establishing their depth at 100% through-wall. The sizes of other flaws were estimated ultrasonically. The field failures are described in Table 3-1, which also provides a brief description of each flaw and how it was dispositioned.

Table 3-1
Field Piping Failures

Plant (listed in order of occurrence)	System	Detection	Comment
Pilgrim	Reactor coolant system (RCS) N1 and N2	(see Note 1)	Replaced
Vermont Yankee	Core spray N5A/B	UT	Overlaid
Palisades	Pressurizer relief	Leak	Replaced
River Bend	Feedwater N4	UT	Destructive (see Note 2)
Hope Creek	Core spray	Leak	Overlaid
Perry	Feedwater N4E	UT	Overlaid
Perry	Feedwater N4C	UT	Monitoring
Nine Mile Point	Feedwater N4	UT	Overlaid
Duane Arnold	Two RCS N2 nozzles	UT	Both overlaid
Fitzpatrick	Control rod drive N9 (capped)	UT	Overlaid
Duane Arnold	RCS N2B, N2D, N2F nozzles	UT	N2B/D overlaid
V.C. Summer	RCS hot leg nozzle	Leak	Destructive
Ringhals	RCS hot leg nozzle	UT	Destructive
JAPC Tsuruga Unit 2	5.12-in. (130-mm) ID pressurizer relief	Leak	Replaced
Pilgrim	Control rod drive mechanism (CRDM) capped N-10	Leak	Overlaid
Susquehanna Unit 1	N1 and N2	UT	Overlaid
Hope Creek	RCS N2 nozzle	UT	Overlaid
Three Mile Island (TMI)	Surge to hot leg	UT	Overlaid
D. C. Cook	Pressurizer relief	UT	Overlaid
Davis-Besse	RCS drain line	UT	Overlaid
San Onofre	Pressurizer relief	UT	Overlaid
Byron (WOL flaws)	Pressurizer surge, spray, relief, and safety	UT (see Note 3)	Preemptive WOL
Wolf Creek	Pressurizer safety, relief, and surge	UT	Overlaid
Joseph M. Farley	Surge nozzle	UT	Overlaid
Duane Arnold	N2C and N2F nozzles	UT	Overlaid
Nine Mile	N2C nozzle	UT	Acceptable
Nine Mile	N2D nozzle	UT	Monitoring
Pilgrim	N2K nozzle	UT	Overlaid
Davis-Besse	Decay heat nozzle	Leak (see Note 4)	WOL

Table 3-1 (continued)
Field Piping Failures

Plant (listed in order of occurrence)	System	Detection	Comment
Crystal River	Decay heat nozzle	UT	Overlaid
Limerick	Core spray nozzle	UT	NRI
St. Lucie	Safety nozzle from retired pressurizer	UT	Not in service
HB Robinson	RPV inlet and outlet nozzle DMWs	UT	Monitoring
Salem Unit 1	Outlet nozzle	UT	Monitoring
Oyster Creek	RCS N1A	UT	Acceptable
Nine Mile Unit 1	N1A and N1B	UT	Acceptable
Hatch Unit 2	N2 inlet nozzle	UT	Monitoring

Notes:

1. Detected during safe-end replacement (see *Nondestructive Evaluation: Dissimilar Metal Weld (DMW) Configuration Database* [21]).
2. Detected in 1989. Growth was monitored over three years, and the joint was replaced in 1992.
3. Support and analysis included detected and postulated flaws in WOL.
4. The crack went 100% through-wall when the first layer of the preemptive overlay was applied.

The characteristics of failures in DMW are quite different from those related to IGSCC in SS. A significant difference is that the DMW flaws were typically located in the weld material, whereas IGSCC is located along the HAZ. With DMW flaws, the cracking is interdendritic rather than intergranular. The growth pattern can appear to be disconnected or discontinuous.

DMWs are normally wider than similar SS welds, which can lead to difficulties, particularly when the flaw is located on the far side of the weld, on welds with limited scan access. Many of the DMW are ground flush, allowing for good scan access to the weld. Where the DMW is close to (or part of) a diameter or thickness transition, scanning access to the weld may be restricted.

3.4 Vessel Head Capability Studies and Inspections, Prior to the Materials Reliability Program (1990–1999)

In 1990, the NRC issued IN 90-10 [22], informing PWR licensees that PWSCC of Alloy 600 or Inconel was an emerging technical issue. PWSCC was noted in 1989 during a refueling outage in Alloy 600 pressurizer heater sleeve penetrations at Calvert Cliffs Unit 2 (CC-2). The NRC staff determined that the safety significance of the cracking was low because the cracks were axial, had a low growth rate, were in a material with an extremely high flaw tolerance (high fracture toughness), and the cracks were unlikely to propagate very far. Later in 1990, EPRI published a report [23] that documented the problem of SCC of Alloy 600 penetrations in PWRs and identified corrective actions that the utilities could take to address this issue. One of the Alloy 600 components listed as susceptible to SCC was the CRDM nozzle penetration.

3.4.1 Control Rod Drive Mechanism Primary Water Stress Corrosion Cracking Discovered Through Nondestructive Evaluation (1991)

In September 1991, a leak from a peripheral CRDM nozzle occurred during a 10-year hydrotest at Bugey 3, a PWR in France, which had operated for approximately 84,000 hours (72,000 effective full power hours or 8.2 years) since 1979. The leak was detected by acoustic emission monitoring and its rate was approximately 0.003 gpm (0.70 l/h). Visual examination revealed that the leaking crack was oriented axially and located on the downhill side at the elevation corresponding to the lowest portion of the partial penetration weld. Destructive examination of the damaged Bugey 3 nozzle revealed that the through wall crack was initiated on the nozzle inside surface and was caused by PWSCC and not by fatigue.

In December 1991, after cracks were found in the Bugey 3 CRDM penetration, an NRC action plan was implemented to address PWSCC at all U.S. PWRs [24]. Because ASME Section XI requirements for CRDM nozzle ISI would not be sufficient to detect cracks in the nozzle or a small leak from a through wall crack, the NRC staff met with the WOG, the Babcock & Wilcox Owners Group, and the Combustion Engineering Owners Group to discuss their respective programs for investigating PWSCC of Alloy 600 and to assess the possibility of cracking of CRDM penetrations in their respective plants. The NRC also asked the Nuclear Management and Resources Council (NUMARC) (now the Nuclear Energy Institute [NEI]) to coordinate future industry actions because the issue was applicable to all PWRs. Each owners group submitted individual safety assessments through NEI to the NRC on the CRDM penetration cracking issue.

3.4.2 U.S. Control Rod Drive Mechanism Demonstration Program (1992–1994)

In sections 3.4.2 through 3.5.3, the term "demonstration" is used. In present day terms, it would now be called a "capability study", as the individual inspector was not qualified. The capability study only "demonstrated" the effectiveness of the vendors' inspection techniques and procedures.

In October 1992, the NUMARC Ad Hoc Advisory Committee (AHAC) formed an inspection subgroup to develop and implement the inspection capability demonstration program. The inspection subgroup coordinated closely with the main Alloy 600 AHAC to ensure that the results of the inspection demonstration program would provide adequate data for utilities in planning or performing inspections.

The approach selected by the AHAC was to develop a program to measure specific performance capabilities of inspection procedures. Therefore, the program did not address qualification of personnel or development of quantitative acceptance standards for inspection processes. Demonstration and qualification of robotic positioning devices were not included in the program because these were to remain the responsibility of the individual utilities. The program developed through the AHAC included:

- Design and construction of mockups
- Demonstration protocol development
- Conducting demonstrations for U.S. plants with upcoming inspections

CRDM performance demonstrations were performed with realistic full-scale mockups containing intentional flaws. Mockups representative of installed designs were used. It was determined that the mockups would be representative of typical geometric conditions in installed penetrations rather than include a range of conditions. This approach made it possible to limit the required number of mockups to a reasonable number. The mockups contained typical geometric features of production penetrations and were fabricated with the same materials and construction procedures used for actual power plant components. Removable thermal sleeves were included to enable demonstrations of gap scanners and scanners that require removal of the thermal sleeves.

Intentional flaws simulating PWSCC were induced in each mockup using well-controlled manufacturing techniques to ensure accurate knowledge of the flaw sizes and locations. The eddy current and ultrasonic response from the flaws were measured and compared to responses from actual PWSCC in penetration samples removed from Electricité de France's (EdF's) Bugey plant to confirm that the responses were similar to those reported in field examinations. EPRI personnel made visits to Swedish and French utilities and inspection organizations in Europe and the United States to assess the characteristics of PWSCC found in service and the various approaches to inspection qualification. Assistance provided by these organizations was essential in developing a realistic and economical demonstration program.

The flaws in the mockups were selected to span a wide range of length, depth, and orientation on the ID nozzle surface because this was determined to be the crack initiation surface. Performance of NDE procedures for a wide range of flaw sizes was determined to be important to utilities to evaluate possible repair or reinspection strategies. Similarly, if no flaws were found in service, the flaw detection performance for small flaws would provide useful information on the smallest flaw that could have escaped detection.

Demonstrations were conducted blind—that is, candidates were not given any information about the number, size, or location of flaws. The demonstration protocol required documentation of the essential variables of the procedures applied. This documentation was required by the utilities so that they could verify that the procedures that were demonstrated were the same as the procedures applied in their plants. This EPRI developed demonstration program was in place by early 1994 [25].

NEI and the PWR owners groups submitted proposed acceptance criteria for flaws identified during ISI of the CRDM penetrations to the NRC [26]. On the basis of the owners groups' analyses and the European experience, the NRC staff concluded, in a safety assessment dated November 19, 1993 [27], that there was a high probability that CRDM penetrations at U.S. plants might contain similar axial cracks caused by PWSCC but that RPV head cracking was not an immediate safety concern. The basis for this conclusion is, if PWSCC occurred in RPVs, the following would be true:

- The cracks would be predominately axial in orientation
- The cracks would result in detectable leakage before catastrophic failure
- The leakage would be detected during visual examinations performed as part of surveillance walkdown inspections before significant damage to the RPV closure head would occur

3.4.3 U.S. Control Rod Drive Mechanism Inspections (1994–1999)

The first U.S. inspection of CRDMs specifically for identifying ID-initiated flaws took place in the spring of 1994 at the Point Beach Nuclear Generating Station. No indications were detected in any of its 49 CRDM penetrations. Flaws detected during this timeframe were all ID-initiated and shallow because the PWSCC tended to stop growing when the ID stresses were relieved. In 1997, the NRC issued GL 97-01 [28] to 1) request addressees to describe their program for ensuring the timely inspection of PWR CRDM and other vessel closure head penetrations and 2) require that all addressees provide the NRC with a written response to the requested information.

3.5 Materials Reliability Program Demonstration Program (2000–Present)

The EPRI MRP was formed in 1998 to identify and address issues that could affect operability of major components in PWR plants. Major activities were coordinated with the NSSS vendors, the PWR owners groups, NEI, and the NRC. MRP provided for a unified industry approach to resolution of technical and regulatory issues related to PWR materials degradation [29].

3.5.1 U.S. Upper Head Penetration Inspections (2000–2001)

Cracking in the CRDM nozzles and seal welds was first noted at Oconee Nuclear Station, Unit 1 (ONS-1) when small amounts of boron residue were found on the top of the RPV head during their Fall 2000 refueling outage inspection. Similar problems were found at ONS-3 in early 2001 during a planned maintenance outage and at ONS-2 during their Spring 2001 refueling outage. Subsequent examinations of the CRDM nozzles that had boron residue found through wall axial cracking in these nozzles and through wall circumferential cracking above the weld in two nozzles at Unit 3 and in one nozzle at Unit 2 [30]. The NRC issued an IN [31] to inform the PWR fleet of this issue.

The identification of circumferential cracking in CRDM nozzles at ONS-2 and ONS-3, along with axial cracking in the J-groove welds at these two units and at ONS-1 and Arkansas Nuclear One Unit 1 (ANO-1), has prompted the NRC to reassess its conclusion in GL 97-01 [28]: that cracking of CRDM nozzles is not an immediate safety concern. Specifically, the findings indicate that circumferential cracks outside the J-groove welds can occur in contrast to an earlier conclusion that the cracks would be predominantly axial in orientation. The findings indicate that cracking of the J-groove weld metal can precede cracking of the base metal. These findings raised questions regarding the industry approach that uses PWSCC susceptibility modeling based on the base metal conditions and does not consider those of the weld metal. In addition, the presence of circumferential cracking at ONS-3, where only a small amount of boric acid residue indicated a problem, called into question the adequacy of current visual examinations for detecting either axial or circumferential cracking in vessel head penetration nozzles. The NRC issued a bulletin [32] to address these inspection issues.

3.5.2 U.S. Control Rod Drive Mechanism Inspection Demonstration (2001–2002)

The CRDM demonstration process had been evolving to address the inspection findings and the resulting changes to inspection requirements. An EPRI NDE demonstration program has been in place since 1994 to address cracking on the ID of the CRDM penetration and was expanded after discovery of OD surface and attachment weld cracking.

The MRP demonstrations for including OD flaws were assembled rapidly to support Fall 2001 inspections. The mockups used for the demonstrations included field-removed penetration tube segments with SCC and full-scale mockups containing conditioned EDM notches via the cold isostatic pressing (CIP) process in the penetration tube. The demonstrations continued in 2002, and the mockups used contained flaws in both the penetration tube and the attachment weld. There were enough tube flaws to evaluate ultrasonic depth-sizing capabilities and included a range of flaw sizes in the wetted surface area of the attachment weld, adequate to evaluate eddy current testing (ET) detection capabilities.

The performance demonstration activities were attended by regulatory personnel. Many utilities contributed to this activity by also attending the performance demonstration activities. The objectives of this demonstration program were the following:

- To develop a performance demonstration program to quantify the capability of NDE techniques and procedures for the examination of the CRDM penetration base material, the interface of the base material and welds, and the wetted surface of the J-groove attachment weld to the RPV head
- To support performance demonstrations of the inspection vendors at the direction of the MRP
- To report findings to the MRP-participating utilities on an ongoing basis
- Provide utilities with critical parameters and performance of techniques to assist in selecting the approach used for examinations

Several performance demonstration activities were conducted. Flaw manufacturing techniques were developed and refined to allow quantification of the inspection procedure performance while providing a realistic simulation of SCC. Practice mockups were manufactured and made available to assist inspection vendors in developing inspection equipment and procedures. These practice mockups were also used to ensure preparedness for the blind performance demonstrations. Results were accepted by the NRC as an adequate demonstration of inspection capabilities [33].

3.5.3 U.S. Upper Head Penetration Inspections (2002–2008)

On February 16, 2002, the Davis-Besse Nuclear Power Station in Oak Harbor, Ohio, began a refueling outage that included inspecting the CRDMs. In early March 2002, while inspecting the CRDM penetrations that were prompted by NRC Bulletin 2001-01 [32], Davis-Besse Nuclear Power Station identified a large cavity in the RPV head near the top of the dome. The cavity was adjacent to a nozzle that was leaking as a result of a through wall axial crack and was located in an area of the RPV head. The wastage volume, the volume of material that was eroded away, was found to extend approximately 5 in. (127 mm) downhill on the RPV head from the leaking penetration and was approximately 4–5 in. (102–127 mm) at its widest part. The minimum remaining thickness of the RPV head in the wastage area was found to be approximately 3/8 in. (9.525 mm). This thickness was attributed to the thickness of the SS cladding on the inside surface of the RPV head, which is nominally 3/8 in. (9.525 mm) thick. The NRC issued an Information Notice 2002-11 [34] to inform the PWR fleet of this issue. In March 2002, the NRC issued Bulletin 2002-01 [35] requesting PWR licensees to provide information on RPV head inspection and maintenance programs, the material condition of their RPV head, past incidents of boric acid leakage that could have reached the RPV head, and the basis for concluding that the boric acid inspection programs for the rest of the reactor coolant pressure boundary are effective. In their responses, they were also to provide information on the extent to which they could conclude that their respective plant(s) did not have RPV head degradation like that identified at

Davis-Besse. In August 2002, the NRC issued Bulletin 2002-02 [36] to further establish inspection methods and frequencies.

On February 11, 2003, the NRC issued Order EA-03-009 [37] establishing interim inspection requirements for RPV heads at PWRs. On February 20, 2004, the NRC issued the First Revised Order EA-03-009 [38], which revised the interim inspection requirements for RPV heads at PWRs.

3.5.4 U.S. Upper Head Penetration Weld Inspection Demonstration (2009–Present)

On September 10, 2008, the NRC published a Rule in the Federal Registry [39, 40, 41] to address the qualification of the upper head penetration inspections. The NRC amended its regulations to incorporate by reference the 2004 Edition of Section XI, Division 1 of the ASME Boiler and Pressure Vessel Code (B&PV Code) and the 2004 Edition of the ASME Code for Operation and Maintenance of Nuclear Power Plants to provide updated rules for inspecting components in light water nuclear power plants. The NRC also incorporated by reference ASME Code Case N-729-1 [42], with conditions. The amendment removes certain obsolete requirements specified in NRC's regulations. This action is in accordance with the NRC's policy to periodically update the regulations to incorporate new editions and addenda of the Codes and is intended to maintain the safety of nuclear reactors and improve NRC activities to make them more efficient.

Beginning September 1, 2009, this Rule requires that ultrasonic examinations be performed using personnel, procedures, and equipment that have been qualified by blind demonstration on representative mockups using a methodology that meets the conditions specified within the Rule and includes the following:

- The specimen set shall have an applicable thickness qualification range of +25% to -40% for nominal depth through-wall thickness. The specimen set shall include geometric and material conditions that normally require discrimination from PWSCC flaws.
- The specimen set shall have a minimum of 10 flaws that provide an acoustic response similar to PWSCC indications. All flaws shall be greater than 10% of the nominal pipe wall thickness. A minimum of 20% of the total flaws shall initiate from the inside surface and 20% from the outside surface. At least 20% of the flaws shall be in the depth ranges of 10–30% through-wall thickness and at least 20% within a depth range of 31–50% through-wall thickness. At least 20% and no more than 40% of the flaws shall be oriented axially.
- Procedures shall identify the equipment and essential variables and settings used for the qualification and verify that they are consistent with Subarticle VIII-2100 of Section XI, Appendix VIII. The procedure shall be requalified when an essential variable is changed outside the demonstration range as defined by Subarticle VIII-3130 of Section XI, Appendix VIII and as allowed by Articles VIII-4100, VIII-4200, and VIII-4300 of Section XI, Appendix VIII. Procedure qualification shall include the equivalent of at least three personnel performance demonstration test sets. Procedure qualification requires at least one successful personnel performance demonstration.

- Personnel performance demonstration test acceptance criteria shall meet the personnel performance demonstration detection test acceptance criteria of Table VIII-S10-1 of Section XI, Appendix VIII, Supplement 10. Examination procedures, equipment, and personnel are qualified for depth sizing and length sizing when the root mean square (rms) error, as defined by Subarticle VIII-3120 of Section XI, Appendix VIII of the flaw depth measurements, as compared to the true flaw depths, do not exceed 1/8 in. (3 mm), and the rms error of the flaw length measurements, as compared to the true flaw lengths, do not exceed 3/8 in. (9.525 mm), respectively.

The aforementioned requirements have been detailed and incorporated into the EPRI Performance Demonstration (PD) program, Quality Assurance (QA) instructions. Additional representative mockups will be built, as required, to meet the geometric conditions of the range of upper head penetrations for this qualification program.

3.5.5 U.S. Lower Head Penetration Primary Water Stress Corrosion Cracking Discovered Through Nondestructive Evaluation (2003)

In April 2003, the South Texas Project (STP) Unit 1 PWR site identified small boron deposits around two of the 58 STP Unit 1 bottom-mounted instrumentation (BMI) nozzle penetrations during a boric acid corrosion control (BACC) program walkdown. STP personnel performed a 100% bare metal visual examination of the STP Unit 1 RPV bottom head as part of the BACC program inspections. Similar inspections had been performed during prior STP Unit 1 and Unit 2 outages, and no evidence of boron deposits had been identified. This was the first, and currently the only, evidence of BMI nozzle penetration leakage reported by a U.S. facility. The STP Unit 1 BMI penetrations were constructed from a bored Inconel 600 bar stock and welded into the reactor vessel lower head by an Inconel 82/182 J-groove weld.

The NDE inspection, which included ultrasonic, visual, and eddy current testing, resulted in the identification of three axially oriented crack-like indications in one penetration nozzle wall and two axially oriented crack-like indications in another penetration nozzle wall. One of the indications was characterized as an axial crack with a length of approximately 1.38 in. (35.052 mm), surface breaking on the outside diameter (OD) of the nozzle above and below the J-groove weld, as well as surface breaking on the ID of the nozzle. The other two indications in this penetration were characterized as being small embedded cracks near the interface between the nozzle wall and the root pass of the J-groove weld. The indication in the other penetration was characterized as an axial crack with a length of approximately 0.98 in. (24.892 mm), surface breaking on the OD of the nozzle above and below the J-groove weld.

The results of the vendors' ultrasonic inspection identified other features within the BMI penetrations that were deemed relevant by STP personnel. UT reflectors were observed and characterized as "discontinuities" at the interface of the nozzle and the J-groove weld in all 58 of the STP Unit 1 BMI penetrations. These discontinuities were particularly evident in seven penetrations including the leaking penetrations. These discontinuities in the leaking penetrations were located in the same general azimuthal locations as the crack-like indications. During the boat sample destructive examination, the UT discontinuities at the tube-to-weld interface were determined to be lack of fusion (LOF), which occurred during initial fabrication. In August, the NRC issued an Information Notice 2003-11 [43] to inform the PWR fleet of this issue.

3.5.6 U.S. Lower Head Penetration Inspection Demonstration (2004–present)

Following the BMI leak at STP Unit 1 during the spring outage of 2003, a demonstration program was quickly developed by MRP based on the RPV upper head demonstration program. Because the bottom head penetrations are considerably smaller than the upper head penetrations, there was not as much volume or surface area available within the tube or on the J-groove wetted surface to implant flaws. The vendors used similar UT and ET inspection techniques to those used for the CRDM upper head demonstration program. The performance demonstration activities were attended by utility personnel and reviewed by regulatory personnel. The objectives of this demonstration program were the following:

- To develop a performance demonstration program to quantify the capability of NDE techniques and procedures for the examination of the various BMI penetration design base material, the interface of the base material and welds, and the wetted surface of the J-groove attachment weld to the RPV head.
- To support performance demonstrations of the candidate inspection vendors at the direction of the MRP.
- To report findings to the MRP participating utilities on an ongoing basis and to provide utilities critical parameters and performance of techniques to assist in selecting the approach used for examinations.

Several performance demonstration activities were conducted. Flaw manufacturing techniques were developed and refined to allow quantification of the inspection procedure performance while providing a realistic simulation of SCC. Practice mockups were manufactured and made available to assist inspection vendors in developing inspection equipment and procedures and to ensure preparedness for the blind performance demonstrations. Results were accepted by the NRC as an adequate demonstration of inspection capabilities. The demonstrations were conducted by MRP, and the results were published in a report [44].

4

TIMELINE OF STRESS CORROSION CRACKING ISSUES, INDUSTRY RESPONSE, AND NONDESTRUCTIVE EVALUATION

Most incidents of PWSCC in nickel-base weld metals have, until recently, been detected by visual observation, typically revealed by the presence of a white boric acid deposit from primary water leaks. Dye penetrant testing, UT, and ET were then used to characterize defects after the initial visual detection. More recently, following the implementation of the EPRI inspection guidelines for dissimilar metal butt welds in MRP 139 [45], defects have been detected by ultrasonic inspection before leakage and usually inferred as SCC from the proximity of the indications to the internal primary water wetted surface. Only in relatively few cases have samples been removed for destructive examination to confirm that the indications were, in fact, SCC. Table 4-1 condenses the NDE history of SCC into a timeline that makes it easier to determine when a particular issue arose and how it was addressed by the industry and the regulators as well as how NDE procedures responded to it.

Table 4-1
Chronology of Stress Corrosion Cracking at Nuclear Plants and the Resultant Industry Response

Year	Stress Corrosion Cracking Issue	Response by Industry and Regulators	Nondestructive Evaluation Used or Training/Demonstration Completed
1959	First IGSCC is seen at Vallicitos BWR in the secondary heat exchanger [5].		
1965	First IGSCC is seen at commercial BWR plant, Dresden 1, in a 6 in. SS bypass line in the HAZ [5].		
Sept. 1974	IGSCC is discovered at Dresden 2 in a 4 in. (102 mm) bypass piping system [5].	The U.S. AEC orders 15 BWR plants to shut down and inspect their 4 in. (102 mm) piping [6].	IE Bulletins 74-10, 74-10A, and 74-10B required code volumetric examination.
1974–1975	IGSCC is detected at nine commercial BWR plants in 4 in. (102 mm) recirculating bypass piping or 10 in. (254 mm) schedule 80 core spray piping [5].		IE Bulletins 74-10, 74-10A, and 74-10B required code volumetric examination.
Jan. 1975	IGSCC is discovered at Dresden 2 in a 10 in. (254 mm) core spray piping system [5].	The newly formed NRC orders that 23 BWR plants be shut down for inspection [6].	IE Bulletin 75-01 required code volumetric examination.
Late 1977	IGSCC is detected at three non-U.S. BWR plants in recirculation riser piping—systems that have high flow during normal plant operation as opposed to past occurrences in stagnant lines [5].		
1979–1980		The BWROG for IGSCC is integrated to create an industry-unified EPRI R&D program [8].	
March 1982	Through-wall IGSCC is discovered at NMP-1 in two safe ends. This event cast doubt on traditional UT capability because the areas found leaking from through-wall IGSCC had been inspected with UT only nine months prior to the incident [8].	NRC issues bulletin to ensure that inspections are sufficient to detect cracking in BWR thick-wall recirculation piping welds and to determine the generic significance of the NMP-1 piping degradation [9].	In September and October, five samples of 28-in. (710-mm) diameter piping with field-removed IGSCC from NMP-1 are transported to Battelle Columbus. The samples are cleaned and secured to ensure that the ID surfaces are not visible to an inspector. In nine days, EPRI prepares a procedure, selects test samples, and organizes a protocol for qualification of NDE teams. Utilities are invited to schedule an ISI team to perform NDE inspections of the samples with the evaluation of the results and a rendering of pass/fail judgment to be made by NRC personnel. This successful effort results in the first IGSCC detection qualification test on October 8, 1982. The NRC analysis of the data from the testing of the nine utility teams demonstrates that adequate performance in detecting IGSCC can be achieved. However, retraining and retesting of the teams is required to achieve the stated competence level [8].
1983		The NRC issues bulletin because of mounting evidence of a generic IGSCC problem in large-diameter BWR SS piping. Action is required of an additional 14 licensees. [8, 10].	
1984		The NRC issues Generic Letters GL 84-07 [46] and GL 84-11 [46].	
May 1984	Axial cracking is detected in the nozzle-to-safe-end DMW during a planned safe-end replacement at the Pilgrim Nuclear Station (Boston Edison Company). Subsequent evaluation of the cracks shows them to be IGSCC originating in the Inconel 182 weld material [8].	The NRC issues an IN informing all BWR facilities of this event [12].	On June 8, 1984, the BWROG approves a research program to address the cracking problem with NDE and repair and replacement studies included. An improved longitudinal-wave UT procedure is developed for detecting cracking in the nozzle-to-safe-end welds in BWR plants. The flaws in the mockups used to develop the improved UT procedure are EDM notches and weld solidification flaws because SCC flaws are not available. The EDM notches are of various depths and lengths in the mockups. This EPRI-developed procedure is used in BWR units to successfully detect IGSCC, which indicates that the artificial flaws used are both realistic and representative [9].
1989	PWSCC is noted during a refueling outage in Alloy 600 pressurizer heater sleeve penetrations at CC-2 [23].	The NRC issues an IN informing PWR licensees that PWSCC of Alloy 600, or Inconel, is an emerging technical issue. The NRC staff determines that the safety significance of the cracking is low because the cracks are axial, have a low growth rate, are in a material with an extremely high flaw tolerance (high fracture toughness), and the cracks are unlikely to propagate very far [22].	

Table 4-1 (continued)
Chronology of Stress Corrosion Cracking at Nuclear Plants and the Resultant Industry Response

Year	Stress Corrosion Cracking Issue	Response by Industry and Regulators	Nondestructive Evaluation Used or Training/Demonstration Completed
1990	Conventional S-wave UT examinations are being used to examine DMWs containing Inconel 600 series base materials, Alloy 82 and 182 weld butter, and/or Alloy 82 and 182 filler materials at PWR facilities [13].	The NRC issues an IN that raises the question as to whether PWR facilities may be using inadequate UT examination techniques for DMWs [14].	An EPRI program is initiated to verify whether the procedures developed for BWR plants are also applicable to PWR designs. The samples used for the 1984 research were no longer available, and it was determined that several available WOG samples would be appropriate to use for this program because of similarity in construction between BWR and PWR welding techniques and materials. These WOG samples contain artificial flaws of mechanical and thermal fatigue cracking. The conclusion is that the NDE procedures are applicable to both BWR and PWR plants based on the UT measurements of notches and artificially introduced cracks in mockups of BWR and PWR weld configurations and on field UT examinations on actual IGSCC in BWR piping systems [13].
1991	The industry determines that a unified approach to demonstrations is important to credibility and acceptability and to ensure that qualifications earned at one location can be applied at another. This will help to avoid the substantial cost of repeating demonstrations [15].	Performance demonstration requirements are added to ASME Code, Section XI in the 1991 Addenda and are described in Appendix VIII. These requirements are applicable to UT examination of piping, bolting, and selected portions of the RPV, but they exclude the shell-to-flange and head-to-flange welds. Appendix VIII requires that procedure effectiveness and personnel proficiency must be demonstrated on realistic mockups containing real flaws. Appendix VIII differs from previous Code approaches in that it does not specify a particular approach. It does require that the capabilities of the personnel, procedures, and equipment be demonstrated and describes in detail the demonstration requirements and acceptance criteria for ultrasonic examinations [16].	Recognizing the importance and complexity of Appendix VIII implementation, representatives from all U.S. nuclear utilities form the PDI at the EPRI NDE Program to implement Appendix VIII. The PDI program is initiated in 1991, with the first demonstrations starting in April 1994. Appendix VIII allows considerable flexibility in the selection of flaw types and locations. The technical committee established the rules for selecting a matrix of flaws for the PWR and BWR sample sets that are both realistic and representative. Fabrication techniques that provide accurate knowledge of the flaw size are used in accordance with PDI specifications. Samples to be used for depth-sizing demonstrations are to be fabricated using flaw implantation techniques that provide a precise knowledge of the true flaw depth. The characteristics of the intentional flaws in the samples are a major consideration for a valid performance demonstration. A considerable portion of the sample fabrication cost is associated with flaw fabrication and verification. The PDI program develops technical criteria for constructing samples and introducing intentional flaws. These criteria describe the accuracy of flaw sizes and locations, flaw characteristics, use of clean base material, and security of the mockups [15, 16].
1991–1992	In September 1991, a leak from a peripheral CRDM nozzle occurred during a 10-year hydrotest at EdF's Bugey 3, a French PWR, which has operated for approximately 84,000 hours (72,000 effective full power hours or 8.2 years) since 1979. Visual examination revealed that the leaking crack is oriented axially and located on the downhill side at the elevation corresponding to the lowest portion of the partial penetration weld. Destructive examination of the Bugey 3 nozzle reveals that the through-wall crack initiated on the nozzle inside surface and is caused by PWSCC and not by fatigue [24].	In December 1991, the NRC implements an action plan to address PWSCC at all U.S. PWRs. The NRC staff meets with the WOG, the Babcock & Wilcox Owners Group, and the Combustion Engineering Owners Group to discuss their respective programs for investigating PWSCC of Alloy 600 and to assess the possibility of cracking of CRDM penetrations in their respective plants. The NRC also asks the NUMARC (now the NEI) to coordinate future industry actions because the issue is applicable to all PWRs. Each owners group submits individual safety assessments through NEI to the NRC on the CRDM penetration cracking issue [24].	In October 1992, NUMARC forms a group to develop and implement the inspection capability demonstration program, which measures specific performance capabilities of inspection procedures. This does not address qualification of personnel or development of acceptance standards. The EdF crack initiation is found to be on the CRDM base metal ID surface, and development is conducted by EPRI to simulate this condition with realistic implanted flaws. It is found that squeezed EDM notches provide realistic NDE responses. The ET and UT response of the manufactured flaws are measured and compared to responses from actual PWSCC samples removed from EdF's Bugey plant to confirm that the responses are similar to those reported in field examinations. The flaws in the mockups span a wide range of lengths, depths, and orientation. This EPRI-developed demonstration program is in place by early 1994 [25].
1994	The first U.S. inspection of CRDMs, specifically for identifying ID flaws, takes place at the Point Beach Nuclear Generating Station. No indications are detected [25].		
1994–1997	Other CRDM inspections find that the flaws detected are all ID-initiated and shallow because the PWSCC tends to stop growing when the ID stresses are relieved [28].	The NRC issues a GL to request utilities to describe their program for ensuring the timely inspection of PWR CRDM and other vessel closure head penetrations, and it require utilities to provide the NRC with a written response to the requested information [28].	
1998	Additional PWSCC in Alloy 600 is found, making the industry realize that additional work must be performed to address this issue [29].	The EPRI Materials Reliability Program (MRP) is formed to identify and address issues that could affect operability of major components in PWR plants. Major activities are coordinated with the NSSS vendors, the PWR owners groups, NEI, and the NRC. MRP provides for a unified industry approach for resolution of technical and regulatory issues related to PWR materials degradation [30].	

Table 4-1 (continued)
Chronology of Stress Corrosion Cracking at Nuclear Plants and the Resultant Industry Response

Year	Stress Corrosion Cracking Issue	Response by Industry and Regulators	Nondestructive Evaluation Used or Training/Demonstration Completed
2000	Small amounts of boron residue are found on the top of the RPV head during the ONS-1 fall refueling outage. PWSCC is found in the leaking CRDM nozzles and seal welds [30].	The NRC issues an IN to inform the PWR fleet of this issue [31].	
2001	Small amounts of boron residue are found on the top of the RPV heads during the ONS-3 and ONS-2 planned spring outages. Examinations of the CRDM nozzles with boron residue find through-wall axial and through-wall circumferential cracking above the welds in two nozzles at Unit 3 and in one nozzle at Unit 2. ANO-1 detects axial through-wall PWSCC during its CRDM RPV head spring inspection [28, 31].	The NRC issues a bulletin addressing these inspection issues. The findings indicate that cracking of the J-groove weld metal can precede cracking of the base metal, which raises questions regarding the industry approach that uses PWSCC susceptibility modeling based on the base metal conditions and do not consider those of the weld metal. In addition, the presence of circumferential cracking at ONS-3 calls into question the adequacy of current visual examinations for detecting either axial or circumferential cracking in RPV head nozzles [32].	Additional representative full-scale and partial-scale blind CRDM mockups are manufactured with squeezed EDM notches on the OD surface of the penetration tube to better replicate what is being seen in the field. NRC personnel attend some of the vendor demonstrations and agree with the demonstration process and the mockups being used [33].
2002–2004	During the Davis-Besse Nuclear Power Station spring refueling outage, a large cavity in the RPV head is discovered. The cavity is adjacent to a CRDM nozzle that was leaking as a result of a through-wall axial crack and is located in an area of the RPV head that the licensee had left covered with boric acid deposits for several years. The wastage area extends 5 in. (127 mm) downhill on the RPV head from the leaking penetration and is 4–5 in. (102–127 mm) at its widest part. The minimum remaining thickness of the RPV head in the wastage area is 3/8 in. (9.525 mm). This thickness is the SS cladding on the inside surface of the RPV head [34, 35].	The NRC issues an IN to inform the PWR fleet of this issue [34]. In March and August, the NRC issues bulletins requesting PWR licensees to provide information on RPV head inspection and maintenance programs, the material condition of their RPV head, past incidents of boric acid leakage that could have reached the RPV head, and the basis for concluding that the boric acid inspection programs for the rest of the reactor coolant pressure boundary are effective [35, 36]. In February 2003 and 2004, the NRC issues orders establishing interim inspection requirements for PWR RPV heads [37, 38].	
2003	In April, the first evidence of BMI nozzle penetration leakage is reported by the STP Unit 1 PWR. STP identified small boron deposits around 2 of the 58 BMI nozzle penetrations during a 100% bare metal visual examination. The STP Unit 1 BMI penetrations are constructed from a drilled Inconel 600 bar stock connected to the reactor vessel lower head by an Inconel 82/182 J-groove weld. The NDE inspection—which includes UT, visual testing, and ET—resulted in the identification of three axially oriented crack-like indications in one penetration nozzle wall and two axially oriented crack-like indications in another penetration nozzle wall. During the boat sample destructive examination, the UT discontinuities at the tube-to-weld interface were determined to be LOF, which occurred during initial fabrication [44].	In August, the NRC issues an IN to inform the PWR fleet of this issue [43].	EPRI MRP develops a blind demonstration program for the BMI configuration similar to the CRDM program. Realistic full-scale mockups are manufactured with squeezed EDM notches. The performance demonstration activities were attended by utility personnel and reviewed by the NRC [44].
2006–2009		In 2006, ASME releases a code case to better address the examination requirements of the RPV upper head penetrations [42]. In 2008, the NRC releases a Rule to address the ASME code case with exemptions and goes into effect September 2009 [39, 40, 41].	Additional representative full-scale and partial-scale blind RPV upper head mockups are manufactured with squeezed EDM notches in the penetration tube. In addition to the CRDM configurations, mockups are manufactured to address the in-core instrumentation nozzles and the vent line nozzles.

5

REFERENCES AND BIBLIOGRAPHY

In-Text References

1. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. *Pipe Cracking in U.S. BWRs: A Regulatory History.* NUREG-1719. Washington, D.C., May 2000.
2. *Materials Reliability Program: Review of Stress Corrosion Cracking of Alloys 182 and 82 in PWR Primary Water Service (MRP-220).* EPRI, Palo Alto, CA: 2007. 1015427.
3. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. *NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping.* Generic Letter 88-01. Washington, D.C., January 25, 1988.
4. *ASME Boiler and Pressure Vessel Code, Section XI, Rules for In-Service Inspection of Nuclear Power Plant Components.* American Society of Mechanical Engineers, New York.
5. B. R. Rajala and P. D. Watson. "History of IGSCC in BWR Systems." Workshop on In-Service Inspection of Stainless Steel Piping in Nuclear Systems, Research Project 892. EPRI, Palo Alto, CA: January 1979.
6. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. *Failures in 4-Inch Bypass Piping at Dresden-2.* IE Bulletin No. 74-10B. Washington, D.C., January 24, 1975.
7. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. *Through-Wall Cracks in Core Spray Piping at Dresden-2.* IE Bulletin No. 75-01. Washington, D.C., January 30, 1975.
8. *BWR Nondestructive Evaluation Source Book.* EPRI, Palo Alto, CA: 1990. NP-6879.
9. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. *Stress Corrosion Cracking in Thick-Wall, Large-Diameter, Stainless Steel, Recirculation System Piping at BWR Plants.* IE Bulletin No. 82-03. Washington, D.C., October 14, 1982.
10. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. *Stress Corrosion Cracking in Large-Diameter, Stainless Steel, Recirculation System Piping at BWR Plants.* IE Bulletin No. 83-02. Washington, D.C., March 4, 1983.
11. Unpublished letter agreed on by NRC, EPRI, and BWROG, "Coordination Plan for the NRC/EPRI/BWROG Training and Qualification Activities of NDE Personnel." Effective January 1984 and Revised September 1985.
12. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. *IGSCC in BWR Plants.* Information Notice No. 84-41. Washington, D.C., June 1, 1984.
13. *Development of Improved Procedure for Examination of Dissimilar-Metal Welds in BWR Nozzle-to-Safe-End Welds.* EPRI, Palo Alto, CA: 1986. NP-4606-LD.

14. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. *Ultrasonic Inspection Techniques for Dissimilar Metal Welds.* Information Notice No. 90-30. Washington, D.C., May 1, 1990.
15. *Examination of Dissimilar Metal Welds in BWR and PWR Piping Systems.* EPRI, Palo Alto, CA: 1993. TR-102148.
16. *ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Appendix VIII, Performance Demonstration for Ultrasonic Examination Systems.* American Society of Mechanical Engineers, New York, NY.
17. *Experience with Inspection Qualifications for Austenitic Piping.* EPRI, Palo Alto, CA: 1997. SP-108338.
18. "Performance Demonstration Initiative: US Implementation of ASME B&PV Code Section XI Appendix VIII." GD16/5. Twelfth International Conference on Structural Mechanics in Reactor Technology (SMiRT 12), August 15–20, 1993, Stuttgart, Germany.
19. *Nondestructive Evaluation: NDE Program Mockup Catalog.* EPRI, Palo Alto, CA: 2009. 1019291.
20. *Nondestructive Evaluation: Dissimilar Metal Piping Weld Examination Guidance.* EPRI, Palo Alto, CA: 2009. 1019134.
21. *Nondestructive Evaluation: Dissimilar Metal Weld (DMW) Configuration Database.* EPRI, Palo Alto, CA: 2006. 1014504.
22. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. *Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600.* Information Notice No. 90-10. Washington, D.C., February 23, 1990.
23. *Literature Survey of Cracking of Alloy 600 Penetrations in PWRs.* EPRI, Palo Alto, CA: 1991. NP-7094.
24. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. *Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking.* NUREG/CR-6245. Washington, D.C., October 1994.
25. *Demonstrations of Inspection Technology for Alloy 600 CRDM Head Penetrations.* EPRI, Palo Alto, CA: 1996. TR-106260.
26. W. Bamford et al. "Development of Acceptance Criteria for Flaw Indications in Reactor Vessel Head Penetrations," *Proceedings: 1994 EPRI Workshop on PWSCC of Alloy 600 in PWRs.* EPRI, Palo Alto, CA: August 1995.
27. Letter from W.T. Russell (U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation) to W. Rasin (Nuclear Management and Resources Council, now Nuclear Energy Institute), dated November 19, 1993. NRC Accession No. 9403020162, *Safety Evaluation for Potential Reactor Vessel Head Adaptor Tube Cracking.*
28. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. *Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations.* Generic Letter 97-01. Washington, D.C., April 1, 1997.
29. *Materials Reliability Program: Pressurized Water Reactor Issue Management Tables – Revision 1 (MRP-205).* EPRI, Palo Alto, CA: 2008. 1018400.
30. *Proceedings: EPRI MRP Alloy 600 Industry Workshop: June 12–14, 2001, Atlanta, Georgia.* EPRI, Palo Alto, CA: 2001. 1006278.

31. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. *Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3.* Information Notice No. 2001-05. Washington, D.C., April 30, 2001.
32. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. *Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles.* Bulletin 2001-01. Washington, D.C., August 3, 2001.
33. *Demonstrations of Vendor Equipment and Procedures for the Inspection of Control Rod Drive Mechanism Head Penetrations (MRP-89).* EPRI, Palo Alto, CA: 2003. 1007831.
34. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. *Recent Experience with Degradation of Rector Pressure Vessel Head.* Information Notice No. 2002-11. Washington, D.C., March 12, 2002.
35. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. *Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity.* Bulletin 2002-01. Washington, D.C., March 18, 2002.
36. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. *Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs.* Bulletin 2002-02. Washington, D.C., August 9, 2002.
37. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. *Issuance of Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors.* Order EA-03-009. Washington, D.C., February 11, 2003.
38. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. *Issuance of First Revised Order (EA-03-009) Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors.* Order EA-03-009. Washington, D.C., February 20, 2004.
39. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. *Federal Register,* Vol. 73, No. 176, pp 52749 and 52750. U.S. Government Printing Office, Washington, D.C. September 10, 2008.
40. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. *Federal Register,* Vol. 73, No. 192, p. 52235. U.S. Government Printing Office, Washington, D.C. October 2, 2008.
41. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. Federal Register, Vol. 74, No. 149, pp 38890–38893. U.S. Government Printing Office, Washington, D.C. August 5, 2009.
42. *ASME Boiler and Pressure Vessel Code Case N–729–1, Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1.* American Society of Mechanical Engineers, New York, 2006.
43. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. *Leakage Found on Bottom-Mounted Instrumentation Nozzles.* Information Notice No. 2003-11. Washington, D.C., August 13, 2003.
44. Materials Reliability Program (MRP): Demonstration of Equipment and Procedures for the Inspection of Alloy 600 Bottom Mounted Instrumentation (BMI) Head Penetrations (MRP-166). EPRI, Palo Alto, CA: 2006. 1010092.

45. Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines (MRP-139). EPRI, Palo Alto, CA: 2005. 1010087.
46. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. Procedural Guidance for Pipe Replacements at BWRs. Generic Letter 84-07. Washington, D.C., 1984.
47. U.S. Nuclear Regulatory Commission. Office of Nuclear Reactor Regulation. *Inspection of BWR Stainless Steel Piping.* Generic Letter 84-11. Washington, D.C., April 19, 1984.

Bibliography

The following reading materials offer additional information of related industry experience not referenced in this report concerning NDE used on SCC-type flaws.

Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: BWR Pressure Vessels. International Atomic Energy Agency, Vienna, Austria, 2005.

Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: PWR Pressure Vessels (2007 Update). International Atomic Energy Agency, Vienna, Austria, 2007.

Becker, F. L. and Findlan, S. J. “Hot Isostatic Pressure (HIP) Bonding for the Fabrication of Ultrasonic Performance Demonstration Test Blocks.” 10th International Conference on NDE in the Nuclear and Pressure Vessel Industries, Glasgow, June 1990. ASM International, Materials Park, OH.

BWRVIP-75-A: BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules. EPRI, Palo Alto, CA: 2005. 1012621.

Development of Flaw Making Techniques for Use in NDE Mockups of Austenitic Materials. EPRI, Palo Alto, CA: 2005. 1011620.

Dissimilar Metal Piping Weld Examination—Guidance and Technical Basis for Qualification. EPRI, Palo Alto, CA: 2003. 1008007.

Ekström, P. and Wåle, J. “Crack Characterisation for In-Service Inspection Planning.” SKI Report 95:70, Stockholm, Sweden, 1995.

Eriksson, B. and Stepinski, T. “Ultrasonic Characterization of Defects – Part 2. Theoretical Studies.” SKI Report 2006:24, Stockholm, Sweden, 2006.

Evaluation of Dissimilar Metal Weld Examinations Performed at Duane Arnold Unit 1 During Refueling Outage 20 (RFO20). EPRI, Palo Alto, CA: 2007. IR-2007-265.

Heasler, P. G., Taylor, T. T., and Doctor, S. R. *Statistically Based Reevaluation of PISC II Round Robin Test Data.* NUREG/CR-5410, PNL-8577. Nuclear Regulatory Commission, Washington, D.C.; Division of Engineering, Pacific Northwest Laboratory, Richland, WA. May 1993.

Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines (MRP-139). EPRI, Palo Alto, CA: 2005. 1010087.

Materials Reliability Program: Proceedings of the 2005 International PWSCC of Alloy 600 Conference and Exhibit Show (MRP-154). EPRI, Palo Alto, CA; AREVA, Lynchburg, VA; Westinghouse, Madison, PA; Structural Integrity Associates, Inc., Centennial, CO; and Welding Services, Inc., Norcross, GA: 2005. 1012089.

Materials Reliability Program: PWSCC of Alloy 600 Type Materials in Non-Steam Generator Tubing Applications—Survey Report Through June 2002: Part 1: PWSCC in Components Other than CRDM/CEDM Penetrations (MRP-87). EPRI, Palo Alto, CA: 2003. 1007832.

Materials Reliability Program: Stress Corrosion Cracking of Stainless Steel Components in Primary Water Circuit Environments of Pressurized Water Reactors (MRP-236). EPRI, Palo Alto, CA: 2007. 1015540.

Nichols, R. W. and Crutzen, S. *Ultrasonic Inspection of Heavy Section Steel Components: The PISC II Final Report.* Elsevier Applied Science, New York, NY, 1988.

Nondestructive Evaluation: Comparison of Field and Manufactured Flaw Data in Austenitic Materials. EPRI, Palo Alto, CA: 2007. 1015143.

PDI Generic Procedure for the Ultrasonic Examination of Dissimilar Metal Piping Welds. PDI-UT-10, December 20, 2002. EPRI NDE Center, Charlotte, NC.

Proceedings: 1991 EPRI Workshop on PWSCC of Alloy 600 in PWRs. EPRI, Palo Alto, CA: 1992. TR-100852.

Proceedings: 1992 EPRI Workshop on PWSCC of Alloy 600 in PWRs. EPRI, Palo Alto, CA: 1993. TR-103345.

Proceedings: 1994 EPRI Workshop on PWSCC of Alloy 600 in PWRs: Parts 1 & 2. EPRI, Palo Alto, CA: 1995. TR-105406.

Proceedings: 1997 EPRI Workshop on PWSCC of Alloy 600 in PWRs: Parts 1 & 2. EPRI, Palo Alto, CA: 1997. TR-109138.

Proceedings: 2000 EPRI Workshop on PWSCC of Alloy 600 in PWRs (PWRMRP-27). EPRI, Palo Alto, CA: 2000. 1000873.

PWR Materials Reliability Project: Analysis of Baffle Former Bolt Cracking in EDF CPO Plants (PWRMRP-03). EPRI, Palo Alto, CA: 1999. TR-112209.

Schuster, G. et al. “An International Round-Robin Test of NDE Reliability for PWSCC.” Nineteenth International Conference on Structural Mechanics in Reactor Technology (SMiRT 19), August 1993, Toronto, Canada.

Technical Review of Surge Nozzle Dissimilar Metal Weld Examination Performed at Joseph M. Farley Unit 2 during Refueling Outage 18 (2R18). EPRI: Palo Alto, CA: 2007. IR-2007-275.

Wåle, J.. “Crack Characterisation for In-Service Inspection Planning - An Update.” SKI Report 2006:24, Stockholm, Sweden, 2006.

Wirdelius, H. and Österberg, E. “Study of Defect Characteristics Essential for NDT Testing Methods ET, UT, and RT.” SKI Report 00:42, Stockholm, Sweden, 2000.

Export Control Restrictions

Access to and use of EPRI Intellectual Property is granted with the specific understanding and requirement that responsibility for ensuring full compliance with all applicable U.S. and foreign export laws and regulations is being undertaken by you and your company. This includes an obligation to ensure that any individual receiving access hereunder who is not a U.S. citizen or permanent U.S. resident is permitted access under applicable U.S. and foreign export laws and regulations. In the event you are uncertain whether you or your company may lawfully obtain access to this EPRI Intellectual Property, you acknowledge that it is your obligation to consult with your company's legal counsel to determine whether this access is lawful. Although EPRI may make available on a case-by-case basis an informal assessment of the applicable U.S. export classification for specific EPRI Intellectual Property, you and your company acknowledge that this assessment is solely for informational purposes and not for reliance purposes. You and your company acknowledge that it is still the obligation of you and your company to make your own assessment of the applicable U.S. export classification and ensure compliance accordingly. You and your company understand and acknowledge your obligations to make a prompt report to EPRI and the appropriate authorities regarding any access to or use of EPRI Intellectual Property hereunder that may be in violation of applicable U.S. or foreign export laws or regulations.

The Electric Power Research Institute Inc., (EPRI, www.epri.com) conducts research and development relating to the generation, delivery and use of electricity for the benefit of the public. An independent, nonprofit organization, EPRI brings together its scientists and engineers as well as experts from academia and industry to help address challenges in electricity, including reliability, efficiency, health, safety and the environment. EPRI also provides technology, policy and economic analyses to drive long-range research and development planning, and supports research in emerging technologies. EPRI's members represent more than 90 percent of the electricity generated and delivered in the United States, and international participation extends to 40 countries. EPRI's principal offices and laboratories are located in Palo Alto, Calif.; Charlotte, N.C.; Knoxville, Tenn.; and Lenox, Mass.

Together…Shaping the Future of Electricity

1021159

Electric Power Research Institute
3420 Hillview Avenue, Palo Alto, California 94304-1338 • PO Box 10412, Palo Alto, California 94303-0813 • USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com