SKI Report 96:24

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ISSN 1104-1374
ISRN SKI-R--96/24--SE

STATENS KÄRNKRAFTINSPEKTION
Swedish Nuclear Power Inspectorate

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Summary

Stress corrosion has been a major problem in the piping of boiling-water reactors and a lesser problem in pressurized-water reactors. This report examines the problems that US and certain foreign reactors have experienced with intergranular stress corrosion cracking (IGSCC) and transgranular stress corrosion cracking (TGSCC). Included is a review of the failure modes and mechanisms, various corrective measures, and the techniques available to detect and size the cracks. The information has been organized into four time periods: from the late 1950s to the mid-1960s; from the mid-1960s into 1975; from 1975 to 1985; and from 1985 to 1991.

The key findings concerning BWRs are:

- Corrective actions at BWRs have led to a substantial reduction of IGSCC.
- Control of carbon level, either through use of ELC grades or NG grades of austenitic stainless steel, should minimize IGSCC.
- Control of residual stresses, particularly with IHSI, greatly reduces the incidence of IGSCC.
- Hydrogen water treatment controls the oxygen and should limit IGSCC.
- The problem with furnace-sensitized safe ends is well recognized and should not recur.
- In most cases, severe circumferential SCC should lead to detectable leakage so that leak-before-break can be identified. However, there can be special cases where the crack size approaches instability. A rule of thumb is that a through-wall crack of 30% of the circumference is stable. This should be detectable by leakage.
- IGSCC of austenitic stainless steels can occur in all pipe sizes from smallest to largest, especially when stress, sensitization, and oxygen are all present.

In the case of PWRs, it is clear that the incidents of primary water stress corrosion cracking appear to be increasing. Cases containing steam generators, austenitic stainless steels, and Inconels have been known for years. Now it is occurring in safe-ends and piping at very low oxygen levels. Secondary side water chemistry must be controlled to prevent SCC in PWRs.

Concerning the detection of IGSCC:

- Leak detection is a good indication of through-wall cracks before they approach instability -- particularly for tight cracks, however, the process may be marginal.

- Acoustic emission will detect small leaks and may detect cracking; however, it is dependent on where transducers are located and how many are used.

- There is increased uncertainty concerning the ability of ultrasonic detection methods to detect and size IGSCC. Appendix VIII of ASME Section XI has improved the situation. Operators will need to qualify on actual IGSCC (or other forms of cracks) before being permitted to conduct UT. With correct use of UT, the probability of detection should be acceptable, though sizing capability still is a problem.

Although there has been considerable progress in controlling and mitigating IGSCC in nuclear plants, problems still exist.

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An Overview of Stress Corrosion in Nuclear Reactors from the Late 1950s to the 1990s

Introduction

The Swedish Nuclear Power Inspectorate (SKI) is continuing to improve their process for the inspection of potential piping failures at Swedish Nuclear Power plants. As part of this effort this report was prepared by Chockie Group International, Inc. and Review & Synthesis Associates to assist SKI in understanding the failure mechanisms, degree of success in resolving problems, and effectiveness of techniques such as ultrasonic detection methods, to detect and determine the size of intergranular stress corrosion cracking (IGSCC) and transgranular stress corrosion cracking (TGSCC). This report presents a historical overview of stress corrosion in the nuclear industry.

Scope of Study

Four time periods were selected to present the information:

- The period from the late 1950s to the mid-1960s, which encompassed the wide variety of prototype reactors;
- The period from the mid-1960s into 1975 when some second generation reactors began to experience problems;
- The period from 1975 to 1985 during which many boiling-water reactors (BWRs) and some pressurized-water reactors (PWRs) experienced stress corrosion cracking (SCC) in many types of piping systems; and
- The period from 1985 to 1991 when most SCC problems should have been resolved but were not.

In this study, US domestic reactors are emphasized, and similar problems experienced in various foreign reactors are also recognized. The findings and conclusions of this study are based on the publicly available data.

Extensive papers could be written on issues such as: 1) the problems faced in detecting IGSCC with ultrasonics and corrective measures taken; 2) the development of alloys resistant to IGSCC such as 304NG and 316NG; and 3) the steps taken to control water chemistry. The intent of this report, however, is to review these topics briefly and without technical detail. The extensive technical and scientific studies on stress corrosion in the nuclear industry will not be discussed.

The First Period: Late 1950s to Mid-1960s

The early years of the peaceful atom were a fertile period for a wide variety of prototype reactors, most of them being relatively short-lived as the field narrowed rapidly to a few viable candidates. A cutoff of early 1966 was used arbitrarily. During this period several of the prototype reactors had suffered SCC, a harbinger of future events. Table 1 lists some incidents of SCC affecting some of these reactors.

The cracking of uranium oxide fuel clad in stainless steel (usually 304 SS) soon caused a shift to the use of zircalloy cladding; the 17-4 PH problem was corrected by raising the aging temperature from 900°F to 1100°F, reducing the hardness. Additionally, some of the steam generator and heat exchanger problems were reduced by better control of chlorides and of free caustic. A change from boron-SS to boron powder encapsulated in 304 SS was a partial solution. The problem of highly sensitized 304 SS or 316 SS (usually caused by furnace sensitization) was recognized; however, failures continued because of this cause for several years.

The Second Period: Mid-1960s into 1975

The number of operating reactors expanded substantially between 1965 and 1975, including the early small BWRs and BWR-1s; and all BWR-2 and BWR-3 plants were operating by 1975. Ten BWR-4 plants also were in commercial operation. Though foreign BWRs are not detailed, a large number were operating (Nuclear Engineering International, 1989). Over 20 domestic BWRs were operating by 1975, and approximately the same number were in operation in other countries.

Table 2 (Bush, 1978) presents a weld count for the various classes of BWRs. Although Dresden 1 is a BWR-1, it differs in several respects and is presented separately. As can be seen, the average number of welds increased substantially for the BWR-2, 3, and 4 class. In the case of Dresden 1, the first five years from 1960 through 1965 were uneventful in terms of the occurrence of SCC. However, the situation changed markedly thereafter, and Table 3 gives IGSCC failure rates during the 1965-75 time frame. As identified by Table 3, the Dresden 1 failure rates are markedly higher than for the BWR-2, 3, and 4 plants. In fact this was caused by shorter operating times. However, the construction practices at Dresden 1, such as high weld heat inputs, also contributed to higher failure rates. A comparison of Tables 2 and 3 confirms that the 4- and 6-inch lines were most susceptible to failure.

Of greater concern were the furnace-sensitized safe ends. Standard practice on the early BWRs was to attach the safe ends before stress relief of the reactor pressure vessel.

Table 1: *Incidents of stress corrosion cracking before 1966.*

Reactor	Material/Component	Year	Type of SCC	References
EBWR	Boron/SS Control Rod Wrought 304SS Cladding on RPV Sensitized	1961 ?	 IGSCC	Power Reactor Tech. 4(4):43-6 9/61; ANL-7117 11/65
Dresden 1	17-4 PH Index Tubes Boron-SS Control Rods 304 SS Fuel Clad	1961 1961 Pre-64		Nuc. Safety 4(3) 32-5 3/63; PRT 4(4):43-6; 9/61 Geneva 1964 Paper 233
Vallecitos BWR	304SS Steel Clad 17-4 PH CRDs	Pre-63 1961		GEAP - 4400; Nuc. Safety 3(1):39-46 9/61
Vallecitos Exp. Superheat Reactor	304SS Fuel Clad			Nuc. Safety 4(3):32-5 3/63
BONUS	304L Steam Pipe (Carbon High; Sensitized) Leaks (Dryer Preheater)	65-66	IGSCC	ORNL-TM-1282 10/65
Shippingport	304SS S.G. Tubes; High Secondary Side Caustic	Pre-59	TGSCC	Nuc. Safety 2(1):102-7 9/60
Hallam	Expansion Tank Failure Sodium Products (Caustic)	1965		NAA-SR-Memo 11607 8/65
Savannah River	Heat Exchanger Tubes 304 High Chloride from Elastomer Gaskets Reactor Bottom Shield Header 304SS Reactor Outlet Nozzle 304SS (Sensitized)	1960 1960 1962 1960	TGSCC IGSC IGSCC	DP-539 DP-539 DP-539
NPR	Steam Generator, 304SS Furnace Sensitized Cleaning Solution S.G. 308SS Overlays Cladding Delta Ferrite	1962 1968	IGSCC IGSCC	
Peach Bottom HTGR	Sensitized 304-304L SS (1100 ° F) Bellows, Superheater Tubes/Pipe	1966	IGSCC TGSCC	FSAR Amend. 13 1966
Elk River	308L SS Weld Overlay Cladding 17-4PH CRDs H-900 H.T.	1961 1962 1965	IGSCC	SWRI-1228-9 8 6/66; 3/65 letter RL Doan
Fermi 1	2 1/4% Cr - 1% Mo Steel SG Tubes Caustics in Cleaning	1963	TGSCC	IAEA Op. Exp. 1963 pp 151-87
VBWR	304SS Recirculation Piping 304SS S.G. Tubes HAZ (Rack) 304SS	1962	IGSCC/ Fatigue IGSCC	NEDG-13851 6/72 APED-4116 11/62
Dresden 1	6-inch Pipe 304SS Steam System	1965	IGSCC IGSCC	Reactor Op. Exp. 68-73 5/6867
Pathfinder	Sensitized/Pickled	1966	TGSCC	Ltr 10/20/67 to Peter Morris
Yankee Rowe	Wrought 304SS Stitch-welded Cladding Cracked in Gas Region of Pressurizer	1965	IGSCC	WCAP 2859
CVTR	SS Sweep Gas System; Chlorides		TGSCC	CVNA - 285

Table 2: *Number of welds for various pipe sizes in BWR plants.*

Pipe Size Inches	mm	Dresden 1	BWR 1 and 2	BWR 3 and 4
2	51	31	328	
4	102	71	198	1033
6	152	95	246	
8	204	22	249	425
10	254	36		642
Other		353	798	4535
Total		**608**	**1819**	**6635**

Table 3: *Failure rates due to IGSCC in welded-heat-affected zones of BWRs (multiply all values by 10^{-4}).*

Failures	Dresden 1 per Weld	Dresden 1 per Weld-Yr	BWR 1 and 2 per Weld	BWR 1 and 2 per Weld-Yr	BWR 3 and 4 per Weld	BWR 3 and 4 per Weld-Yr
All Welds	395	25.1	22.0	2.59	60.3	15.1
2-inch (51mm) line welds	--	--	30.5	3.59	--	--
4-inch (102mm) line welds	845	53.9	50.5	5.94	251.6	62.9
6-inch (152mm) line welds	1684	106.8	40.7	4.79	--	--
8-inch (204mm) line welds	909	57.7	40.2	4.73	141.0	35.3
10-inch (254 mm) line welds	--	--	--	--	125.0	31.2

The stress relief temperature of 6200°C (11500°F) led to severe sensitization of the 304 stainless steel. Table 4 presents this phenomenon. As can be seen, 13 reactors representing almost all the older domestic BWRs operating had a total of 244 sensitized safe ends. The calculated failure rate is quite high; however, most sensitized safe ends were removed or weld clad on the inside. Nine Mile Point 1 nearly suffered a double ended guillotine break (DEGB) after three months of commercial operation. A design error placed supports on the 10-inch core spray line without accounting for the change in height of the vessel due to heat-up. As a result very high bending stresses, combined with the severe sensitization, led to extensive cracking. The crack was detected during an outage because of water leaking through the IGSCC cracks. Figures 1 and 2 illustrate the geometry of safe end and crack size, and the micrograph in Figure 3 illustrates a through-wall crack.

Table 4: *Failure rates in furnace-sensitized (FS) BWR components.*

Item	General Electric Reactors	Other Reactors
Reactors with FS components	13	2 or 3
Total number of FS components (safe ends)	244	18 or 26
Reactor-years of service (through January 1976)	107	13 or 20
Service* life of FS components	2037	122 (190)
Number of IGSCC failures	18	4(5)
Failure rate per FS component	732×10^{-4}	$2222 (1923) \times 10^{-4}$ **
Failure rate per FS component year*	88×10^{-4}	$328 (263) \times 10^{-4}$

* These data are invalid, since FS components were replaced or were clad by welding in most instances. Therefore, the failure rate should be substantially larger.

** Numbers are based on 4 failures and 18 components or 5 failures and 26 components to yield failure rates per component.

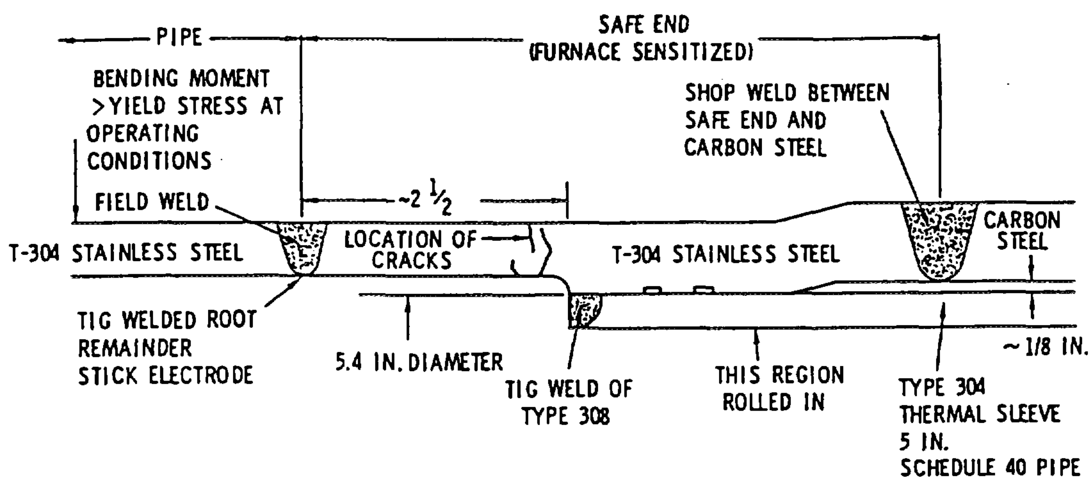
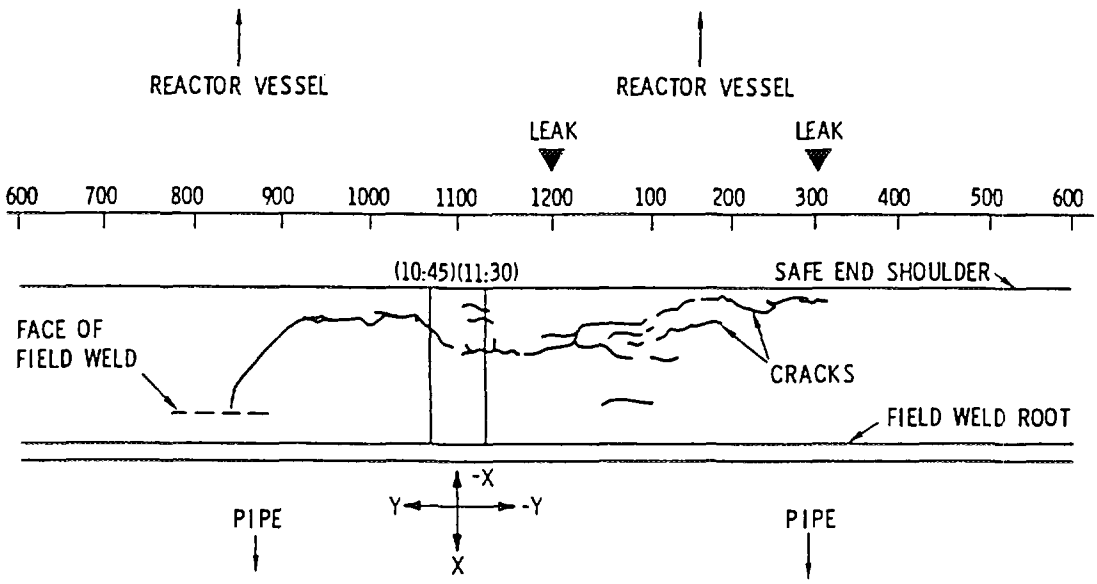


Figure 1: *Schematic diagram showing safe end configuration.*

Table 5 (NUREG-0679, 1980) is an incomplete list of IGSCC in BWR piping through July 1975. Several incidents are not included; however, it does confirm the susceptibility of Dresden 1 (Bush, 1973). Humboldt Bay 3, Garigliano, GKN, and Elk River suffered IGSCC of piping. Other components in BWRs and PWRs also suffered SCC. For example, cap screws, bolts, studs, nuts, control rod drives (17-4 PH), turbines, buckets, and stainless fuel cladding pins were attacked, mostly in the form of IGSCC, although there were instances of TGSCC.

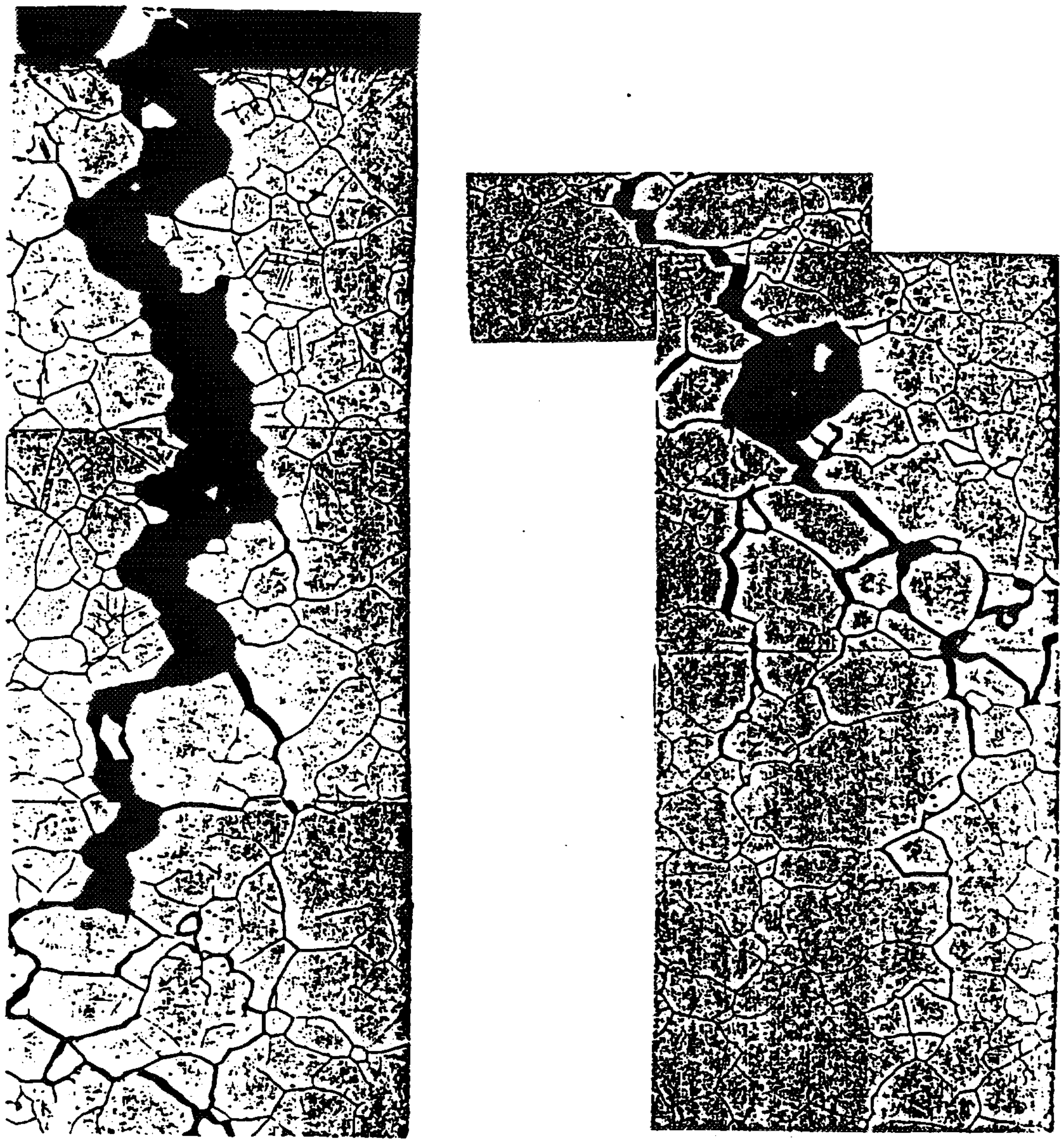


Note: The 12:00 position was at the top of the tube. Radiographs were taken through the double thickness of the tube before specimens were removed. Approximate location of leaks noted.

Figure 2: *Schematic description of radiographic indications from safe end showing location of cracks.*

In PWRs the steam generator tubes in several reactors were attacked from the secondary side. Both Inconels and stainless steel tubes suffered IGSCC. Several reactors, including BWR, PWR, and GCR, suffered cracking of both turbine buckets and disks. In the case of Hinkley Point (Bush, 1973), the disks failed leading to a catastrophic failure with penetration of the turbine shell. Cracking initiated at the keyways and grew until critical size. NUREG-75/067 represents a logical breakpoint in time because of the multiple cases of SCC in late 1974 and early 1975. Table 6, compiled from NUREG-75/067, illustrates the problems with Dresden 1 over the decade 1965-75.

The general opinion among nuclear scientists and engineers in 1974 was that the SCC in 4-inch, 6-inch, and 8-inch lines in Dresden 1 was unique to that plant. In fact, General Electric scientists remarked that the trends that indicated increasing cracking rates for longer periods of time were not valid because of the corrective actions taken to minimize sensitization and cold work (Bush, 1973). The plants cited are the BWR-3s, which are discussed below. Figure 4 presents is a graphical presentation of SCC incidents by years of operation.



Note: Face being examined is parallel to pipe axis. Left figure is initial portion of crack, right figure is bottom. The bottom of the left figure and top of right are exactly contiguous.

Figure 3: *Optical micrograph of crack starting at inside surface and extending toward the outer surface.*

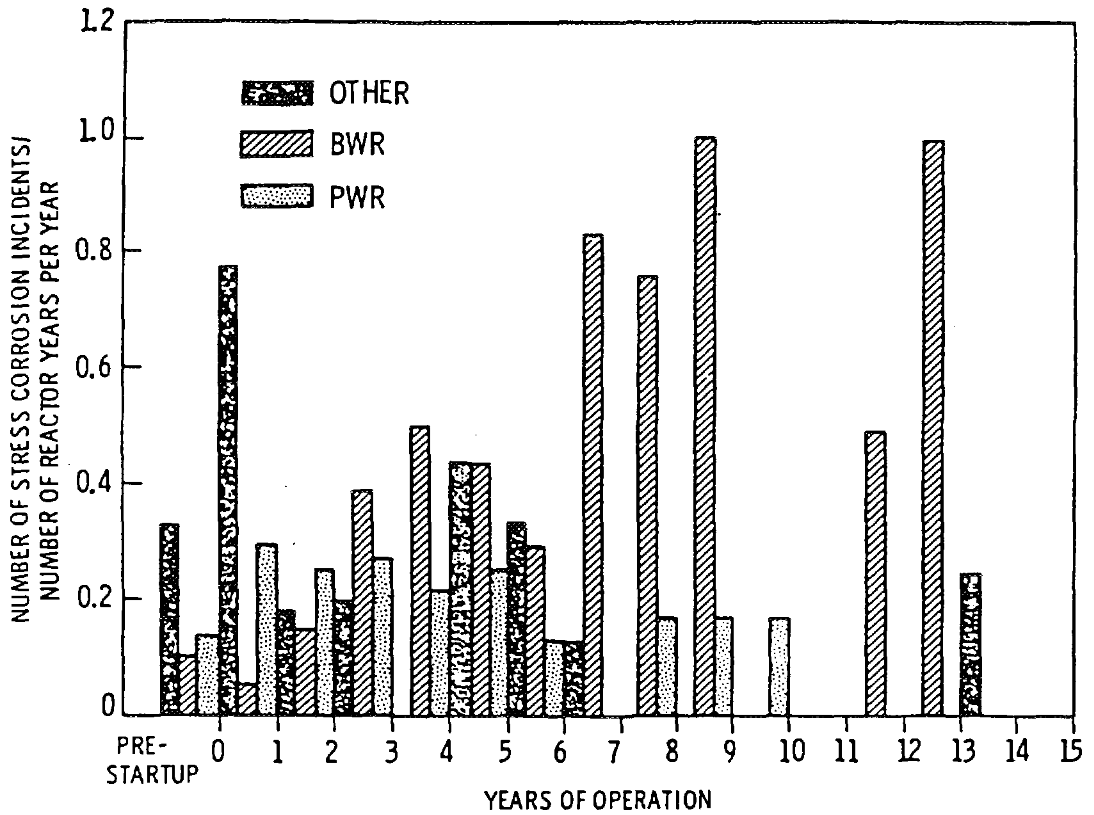
Table 5: *Summary of intergranular stress corrosion pipe failures in BWRs, through July 1975.*

Plant	Size (MWe)	Generator (Synchroized)	Material (Other than Type 304 SS)	Weld Zone Cracking Incidents	Lines	Furnace Sensitized Safe Ends
Experimental BWR	4.5	1957				
Vallecitos BWR	5	10/15/57				
Dresden 1	200	4/15/60		24	9	
Kahl	15	6/17/61	347			
Big Rock Point 1	72	12/8/62	316			
Humbold Bay 3	70	4/18/63	1-Cold work			x
Elk River	22	8/24/63				x
JPDR	11	10/26/63		1	1	x
Garigliano	150	1/23/64				x
KRB (Gundremmingen)	237	11/12/66				x
LaCrosse	52.4	4/26/68				x
Lingen KWL	240	5/20/68				x
GKN	48	10/26/68	316			x
Tarapur 1	190	4/1/69		1	1	x
Tarapur 2	190	5/7/69		1	1	
Oyster Creek 1	560	9/23/69	316			x
Nine Mile Point 1	500	11/10/69		1	1	
Tsugura 1	340	11/16/69		1	1	
Dresden 2	809	4/13/70		12	4	
Fukushima 1	440	11/17/70		8	3	
Millstone Point 1	652	11/29/70		3	2	
NUCLENOR	440	3/2/71				x
Monticello 1	545	3/5/71		3	2	
BKW (Muehleberg)	306	6/24/71				
Dresden 3	809	7/22/71				
Wurgassen	612	12/18/71				
Oskarshamn 1	440	1/72				
Quad Cities 1	809	4/12/72		4	2	
Quad Cities 2	809	5/23/72		4	2	
Pilgrim 1	652	7/19/72				
Vermont Yankee	540	9/20/72				
Browns Ferry 1	1075	10/15/73				
Ringhals 1	750	11/73				
Shimane	442	12/2/73				
Fukushima 2	760	12/24/73				
Peach Bottom 2	1075	2/16/74				
Cooper 1	778	5/10/74				
Duane Arnold	550	5/19/74				
Oskarshamn 2	580	8/74				
Hamaoka 1	513	8/1/74		1	1	
Browns Ferry 2	1075	8/28/74				
Peach Bottom 2	1075	9/1/74		1	1	
Fukushima 3	754	10/26/74				
Hatch 1	754	11/11/74				
Fitzpatrick 1	786	2/1/75				
Totals				**65**	**31**	

Table 6: *History of stress corrosion in Dresden 1 piping 304 SS coldwork, local sensitization.*

Date	System	Pipe Diameter (inches)	Detected by	Comments
12/65	Bypass line at "C" Recirculation Loop	6	Leak	
2/66	Decontamination stub on "D" Recirculation Loop	4		
4/66	Bypass line on "B" Recirculation Loop	6	UT	
4/66	Header to Demin. System	6	Leak	
	"A" Cleanup Demin. Line	4	UT	Longintudinal Indication
	Bypass line on "A" Recirculation Loop	6	UT	Circumfirential Crack
	Bypass line on "C" Recirculation Loop	6	UT	
10/67	Bypass line in "D" Recirculation Loop	6	UT	Defect Grew
12/69	Reduce RPV Head to Steam Drum Ventline	4x2	Leak	
11/70	Line connected to "A" Recirculation Loop	4	Leak	Detected by Air Sampler
2/71	Demineralizer in "A" Recirculation Loop	4	Leak	Detected by Air Sampler
1/72	Reactor Vent Line	4x2	Leak	
6/74	Steam Supply Line to Emergency Condenser	8	Leaks	

Table 7 (NUREG-75/067, 1975) covers the BWR-3s. The only plant not reporting IGSCC was Dresden 3. As can be seen, there were several cases of cracks and leaks in the 4-inch bypass lines plus other incidents in the 10-inch core spray lines. Figure 5 indicates where cracking occurred in the bypass lines, except at Monticello, where the bypass line geometry differed. Tables 8 and 9 (Bush, 1973) provide an overview of the 1965-73 period. The variety of components attacked in the various cases is listed in Table 8. Of note are the substantial incidents occurring preoperationally. Table 9 provides an early opinion of the parameters contributing to SCC. Most of the significant parameters are still considered to be in that category at this time.



Note: An incident is defined as the discovery of one or more cases of SCC at a given time in the same alloy. Fuel cladding failures are not included.

Figure 4: *Stress corrosion cracking incidents (limited to US data).*

NUREG-75/067 (1975) contains some important conclusions and recommendations, many of them are still unresolved nearly ten years later. Essentially all are included later under the Conclusions and Recommendations Section in NUREG-1061, Volume 1 (1984). One exception relates to dead legs. Items related to stagnant flow are presented in the following sections.

Recommendations: Based on Second Period Experiences

Non-flowing or low-flow branch runs of austenitic stainless steel piping are more susceptible to SCC and therefore require more frequent examination than those pipes where the coolant continuously flows during the plant operation.

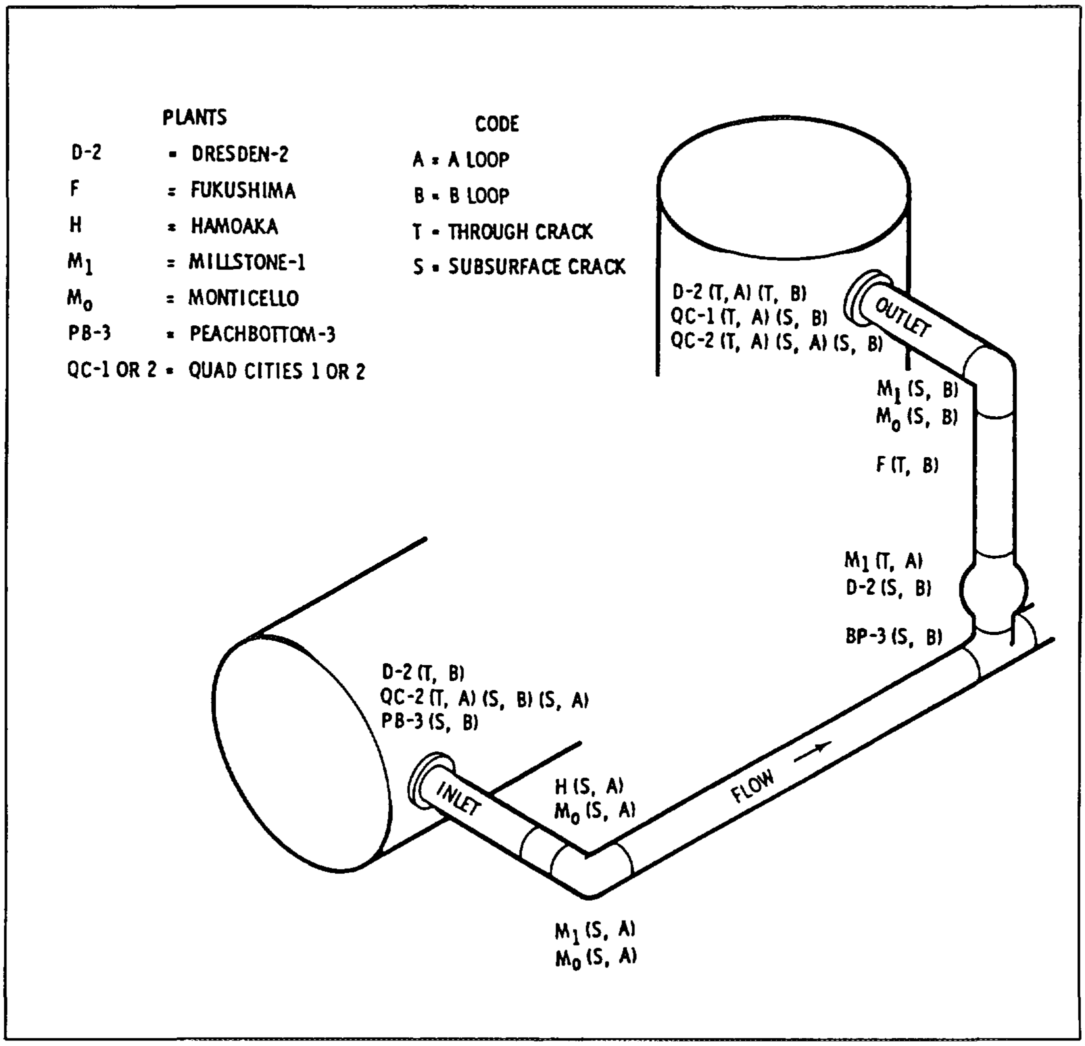


Figure 5: *Record of recirculation bypass cracks.*

Recommended Modifications of Plant Operational Practices

As part of the long-term program to ameliorate conditions of plant operation by minimizing the potential for stress corrosion cracking of austenitic stainless steel piping in BWR plants, the following recommendations should be implemented:

- Investigate revision of operating procedures to improve flushing of non-flow or low-flow branch lines in order to minimize any potential for oxygen concentration.

- Eliminate, where feasible, non-flow conditions in branch lines by operating with shutoff valves in open mode (e.g., valve in by-pass lines).

Table 7: *Summary of cracking incidents in stainless steel piping in BWR plants (late 1974 to early 1975).*

Code	Facility	Date	Crack Location Loop A	Crack Location Loop B
Bypass Lines				
D-2	Dresden 2	9/13/74	1 Thru	1 Thru
QC-2	Quad Cities 2	9/16/74		1 Subsurface
Mi-1	Millstone	9/18/74	1 Thru	
D-2	Dresden 2	12/13/74		1 Thru
QC-2	Quad Cities 2	12/23/74	2 Subsurface	1 Subsurface
PB-3	Peach Bottom 3	1/5/75		2 Subsurface
QC-1	Quad Cities 1	1/10/75	1 Thru	1 Subsurface
Mo	Monticello	1/21/75	2 Subsurface	1 Subsurface
	Number of Bypass Pipes with Cracks		10	
	Total Number - Crack Locations		15	
	Thru Wall Cracks		5 Locations	
	Surface Cracks		10 Locations	
10 inch Core Spray Pipes				
D-2	Dresden 2	1/28/75	3 Thru	2 Thru
D-2	Dresden 2	2/10/75	1 Thru	
	Number of Core Spray Pipes with Cracks		2	
	Total Number of Crack Locations		3	
	Total Number Cracks		6	

Recommended Modifications for BWR Plants Under Construction Permit Review or Being Constructed

For plants that are currently in review for a construction permit or are under construction, the following recommendations are made:

- Operate plants with bypass-line valve open in BWR 5 and BWR 6 plants, provided it does not impair effectiveness of engineered safety systems.
- Eliminate as many non-flow and low-flow lines as possible.
- Within limits of good design practice and serviceability during operation, minimize the extent of the reactor coolant pressure boundary by methods such as placing valves as close to the reactor vessel as possible. Particularly, non-isolable piping in the reactor coolant pressure boundary should be kept to an absolute minimum.

Lessons Learned

The decade 1965 to 1975 was one of substantial growth. Unfortunately, the lessons learned concerning construction procedures and their impact on the probability of SCC had limited impact. The problem of furnace-sensitized safe ends was apparent; thus they were removed from older plants and not permitted on plants under construction. With appropriate action, many preoperational problems can be eliminated and some progress made in minimizing operational problems. Table 8 indicates the variety of problems over the spectrum of reactors. Many of the problems cited had been resolved by the early 1970s; others continued to haunt the utilities. Table 9 indicates that the industry has not learned to take corrective action. Many of the significant factors leading to SCC are seen in failures reported in the period 1985 to 1990.

The Third Period: 1975 to 1985

Stress corrosion cracking continued to be a problem during the period of 1975 to 1985. This period can be divided roughly into three intervals:

1. IGSCC in furnace-sensitized safe ends of large-diameter (24-inch) pipe in a foreign reactor was the principal cause for establishing the second pipe crack group, reported in NUREG-0531 in early 1979;

2. Several incidents of cracking in the PWR secondary systems, some being IGSCC, initiated the third PCSG reported in NUREG-0691 in September 1980;

3. The occurrence of IGSCC in recirculation piping of domestic reactors in 1982 led to the formation of a task group on pipe cracking which was a part of the larger Piping Review Committee. The task group reported in NUREG-1061, Volume 1, in August 1984 as a portion of the 5-volume piping review.

Implications of NUREG-0531

The report entitled, *Investigation and Evaluation of Stress-Corrosion Cracking Piping of Light-Water Reactor Plants,* (NUREG-0531, 1979) was an extension of NUREG-75/067 (1975) with the addition of the following:

- the problem of sensitized safe ends;
- relevance of IGSCC in large-diameter (24-inch) piping in a foreign plant to domestic BWR recirculation lines;
- SCC in PWRs;
- foreign reactor IGSCC experience relevant to IGSCC;
- ability of ultrasonic techniques to detect cracks in austenitic stainless steel; and
- significance of cracking in Inconel safe ends at Duane Arnold.

Table 8: *Classes of reactors, components, and materials in which stress corrosion has occurred (Bush, 1973).*

Reactor Class	Components	Occurrence*	Materials
Boiling Water	Vessel Clad	P,O	Type 304 Wrought or Weldment
	Vessel Internals	P,O	304 SS
	Nozzle Safe Ends	P,O	304 SS
	Control Rod Drives	O	17-4PH SS
	Piping	P,O	304 SS
	Valves	O	304 SS
	Studs	O	Alloy Steel (4140, etc.)
	Fuel Cladding	O	Zircaloy
	Turbines	O	Alloy Steel
Pressurized Water Nontube LWR			
Primary	Pressurizer Clad	O	304 SS
	Flux Thimbles	O	304 SS
	Studs	O	Alloy Steel Martenistic SS
Secondary	Steam Generators	O	304, 347 SS Ni-Cr-Fe Alloy 600
	Turbines	O	Alloy Steel
Tubed LWR			
Primary	Fittings	O	17-4PH SS
	Tubes	O	Zircaloy-2
	Cladding	O	Zircaloy-2 Zircalay-4
Secondary	Valves	O	17-4PH Stem
	Steam Generators	O,P	304 SS
Superheat			
Primary	Fuel Cladding	O	304 SS
	Piping	O	304 SS
	Superheat Section	O	304 SS
Heavy Water Moderated	Low Temperature		
Light Water or Heavy Water Cooled			
	Headers	O	304 SS
	Nozzles	O	304 SS
	Heat Exchangers	O	304 SS
	Bolting	O	17-4PH SS
	Instrument Lines	O	Ni-Cr-Fe Alloy 600
	Sweep Gas Lines	O	304 SS
	High Temperature		
	Gaskets	O	304 SS
	Piping	O	304 SS
	Fittings	O	304 SS
Gas-Cooled			
Secondary	Steam Generators	P	304 SS
	Turbines	O	Alloy Steel
LMFBRs			
Secondary or Tertiary	Steam Generators	P	2-1/4 Cr-1 Mo
	Steam Generators	O	Ni-Cr-Fe Alloy 600

* Note: P = Preoperation
O = During Operation

Table 9: *Parameters contributing to stress corrosion cracking in nuclear power systems as identified from available case histories.*

Material	Low-Med Alloy Ferritic Steels	Martensitic Types 420, 440, 17-4PH	18/8 304 316	Austenitic	High Nickel Ni-Cr-Fe Alloy-600
Ferrite Percent	--	--	S	--	--
Carbon Effect	--	--	S	--	C
Fabrication					
Cold Work	--	--	S	--	C
Cleanliness	--	--	S	--	C
Heat Treatment					
Solution	--	--	-	--	--
Sensitizing	--	--	S	M	M
Precip Hardening	--	S	-	--	--
Welding Heat Input	--	--	S	--	--
Strength & Hardness	S	S		--	--
Pickling	--	--	S	--	S
Design					
Stress (Weld- Residual Thermally induced External Loads)	S	--	S	S	S
Crevices (Concentrator)	S	-	S	S	S
Galvanic Coupling	S	S	--	--	--
Cyclic Loading	--	--	S	--	--
Operational					
Temperature	--	--	M	--	S
Local Heat Flux	--	--	S	--	--
Neutron Irradiation	--	C	M	S	--
pH	--	--	M-S	--	--
Additives (Alkaline-LiOH, KOH, NaOH, Borax)	--	--	--	--	--
Acid	--	--	--	--	--
Residuals					
Oxygen	--	S	S	--	S
Chlorides	--	--	S	--	M
Fluorides	--	--	C	--	--
Lead	--	--	--	--	S
Nitrates	--	--	--	--	--

S = Significant *M = Minor* *C = Controversial*

The review was limited to piping where safe ends were defined as piping. With regard to large-diameter piping, the conclusion in NUREG-75/067 was confirmed, and the PCSG felt there was little evidence to indicate that IGSCC will not occur to some degree in large-diameter BWR stainless-steel piping in the United States. The PCSG concluded that "conditions in large diameter piping are less severe than those in core spray or recirculation bypass lines, where frequent IGSCC has been observed. Because conditions for IGSCC in large-diameter lines are less severe, IGSCC in these lines will be less frequent and may not occur for a long period." Though this statement was correct, the portion related to "less frequent and not occurring for a long period" was unduly optimistic as will be seen in the discussion of NUREG-1061, Volume 1 (1984).

NUREG-0531 was less optimistic than NUREG-075/067 with regard to the viability of ultrasonics to detect IGSCC cracks. Because this is a topic in both NUREG-0691 (1980) and NUREG-1061, Volume 1 (1984), it will not be pursued further.

The problem of cracking in safe ends was exacerbated by the fact that the IGSCC in the 24-inch diameter lines in the German reactor occurred in furnace-sensitized safe ends on the secondary side of the steam generator of the BWR-1 plant. Three domestic BWRs still had sensitized 304 SS safe ends as of 1978 (Dresden 2, Big Rock Point 1, and Nine Mile Point 1).

The potential for stress corrosion cracking in PWRs is addressed extensively in NUREG-0691 (1980) and is discussed later in this report. In NUREG-0531 the PCSG concluded that SCC of PWR primary system is improbable, but the secondary and tertiary piping systems were susceptible. Generally, impurities or chemical additives were responsible for the SCC.

The significance of cracking in the Inconel-600 safe ends at Duane Arnold merits an in-depth review. The cracking could be classified as a near miss double-ended pipe break for one or more of the safe ends. Figure 6 illustrates a typical BWR recirculation system, i.e., the configuration of the Duane Arnold Recirculation-Inlet Nozzle Safe End including location of a repair weld and a representation of IGSCC in the leaking recirculation inlet-nozzle safe end. The nozzle is approximately 10 inches in diameter. The repair weld occurred because of an error in the fabrication drawing that resulted in the machining of a groove in the outer surface of all eight safe ends. Repair consisted of filling with weld metal. Since the safe ends were attached to the vessel nozzles prior to stress relief, they underwent two 1100±25°F (595±15°C) stress relief treatments. Note the thermal sleeve that was installed to direct coolant flow into the vessel. It also resulted in a long, narrow crevice ending at the tip of the attachment weld. This crevice led to a stagnant chemical environment conducive to IGSCC. A severe resin spill produced an acid sulfate that was trapped in the crevice as confirmed by the presence of sulfur and low pH.

A combination of the chemical environment, oxygen, high residual and operating stresses, and sensitization led to severe IGSCC. Figure 6 represents the one leaking safe end. Note the 360° circumferential crack that is through-wall in one section and 50-75% through-wall over most of the remaining circumference. While Figure 6 illustrates the leaker, the other seven safe ends were cracked almost completely around the circumference. Duane Arnold and Nine Mile Point represent the two cases of SCC in nuclear power plants that could have led to a DEGB. Therefore, the lesson learned is to avoid crevices (probably by not installing thermal sleeves) and conduct inservice inspection (ISI) in conformance with NUREG-0313.

NUREG-0679 (1980) covers SCC through July 1979. Table 10 attempts to parallel Tables 3 and 7. It is apparent that the BWR-4 plants also are susceptible to IGSCC. The two pipe sizes and systems most susceptible are 4-inch and 8- or 10-inch lines in cleanup and core spray systems.

Results in NUREG-0691

Though the major emphasis of NUREG-0691 (1980) was thermal fatigue in PWR-feedwater lines, the potential for SCC in PWRs, covered previously in NUREG-0531, was reexamined, as was the applicability of ultrasonic techniques (UT) to detect cracks in austenitic stainless steel.

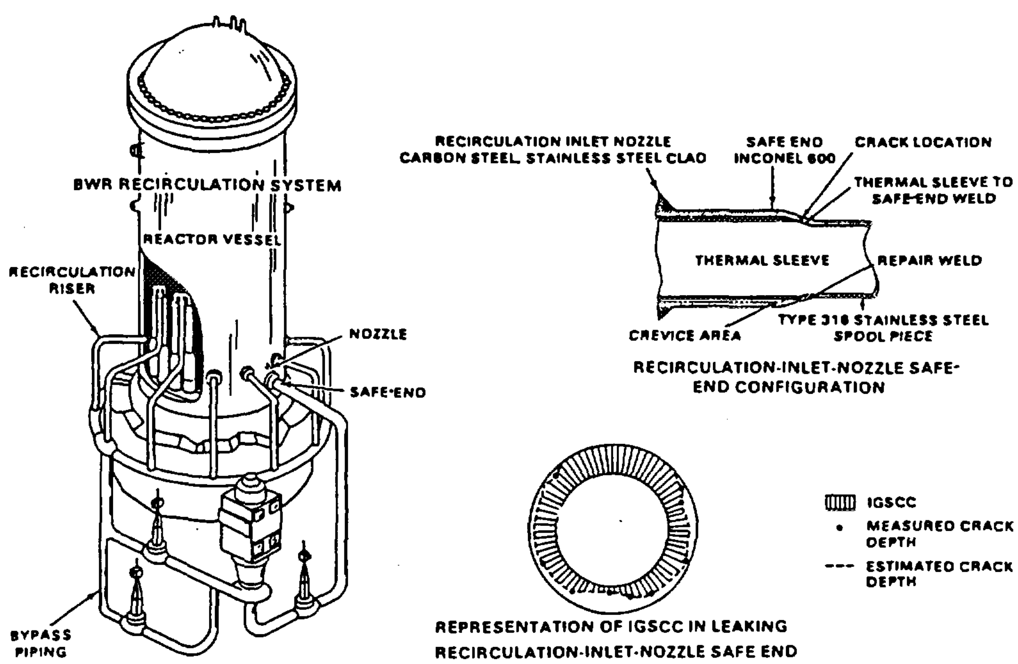


Figure 6: *Example of severe intergranular stress corrosion cracking in Inconel-600 safe end.*

Table 10: *Summary of stress corrosion cracking incidents July 1975 to July 1979.*

Category	Number of Incidents		Comments
By BWR Class			
BWR-1	4		
BWR-2	13(one reactor)		
BWR-3	23+		
BWR-4	66+*		
By System			Table 2.1 of
Instrument Lines	5		NUREG-0531
Cleanup	36	14	covers July 1975 to
Recirculation Bypass	12	12	January 1979
Core Spray	42*	17	
Recirculation Riser	7	13	
CRD	3	1	
Other	14	12	
By Pipe Size			
<4-inch	17		
4-inch	26		
6-inch	20		
8-inch	13		
10-inch	38+*		
12-inch	8		
14-inch	2		
24-inch	6		
By Year			
1975	10		
1976	23		
1977	33		
1978	10		
1979	33*		
Foreign vs. Domestic			
Foreign	24		
Domestic	85		
N.B.	25 were leaks		

* Biased by 23 incidents of IGSCC in core spray lines in one plant in 1979.

The use of UT is even more marginal in Class-2 systems as compared to Class-1 systems because the Class-2 weldments were not fabricated to permit effective UT. Also, UT was not very effective when applied to thin-walled stainless steel such as schedule-10 piping. In a broader sense, the UT procedures of ASME Section V and XI were not considered adequate for austenitic stainless steel. This is amplified in the discussion under NUREG-1061, Volume 1.

Table 11 from NUREG-0691 presents a listing of PWRs reporting IGSCC pipe cracks and leaks. Eight PWRs reported cracking in seven different systems, mostly in 10-inch or 8-inch pipe with the majority Schedule 10 but a substantial amount Schedule 40. At Three Mile Island 1, there was substantial cracking in the spent fuel and decay heat systems. The SCC was both TGSCC and IGSCC, generally caused by impurities or chemical additives, e.g. boric acid.

A specific question to the PWR PCSG was to evaluate the safety significance of pipe cracks discovered in PWR safety-related systems. The study was to include material, fracture mechanics, system and stress analyses aspects for safety-related piping 2 1/2 inch in diameter and larger.

Typically, austenitic stainless steel pipes in PWR secondary systems that have experienced SCC have low or stagnant flow conditions and contain borated solutions; the stainless steels also had relatively high carbon contents. Service experience has indicated that the SCC is usually found by small leaks with no significant coolant loss.

The nature of the cracks, the relatively low system loads, and the ability of the system to maintain function led the PCSG to conclude that SCC did not represent a substantial

Table 11: *PWR facilities reporting pipe cracks and leaks.*

Facility	System	Pipe Size
Arkansas 1	Building (containment spray)	10-inch and 8-inch Schedule 10
	Decay heat removal	10-inch Schedule 10
	Spent fuel pool cooling	3-inch to 2-inch reducer
Crystal River 3	Containment spray	8-inch Schedule 40
Ginna	Safety injection	8-inch Schedule 10
H. B. Robinson 2	Boron injection	4-inch
San Onofre 1	Containment spray	6-inch Schedule 10
	Refueling water pump suction	8-inch Schedule 10
Surry 1	Containment spray	10-inch Schedule 40
Surry 2	Containment spray	10-inch Schedule 40
Three Mile Island 1	Spent fuel pool cooling	8-inch Schedule 40
	Borated water storage tank to residual heat removal suction	10-inch Schedule 40

safety problem, although corrective action was necessary to decrease the incidence of SCC and the personnel exposure associated with repair, replacement, and inspection.

Specific recommendations related to the PWR SCC problem were as follows (NUREG-0691, 1980):

- To assist in reducing the incidence of SCC in PWR secondary systems, contaminants leading to SCC should be reduced to the extent possible by control in the fluid reservoirs, periodic flushing, and rinsing of surfaces.
- UT procedures for detection of IGSCC should be demonstrated to the satisfaction of NRC prior to use in the field on the basis of examination of pipe weldment samples containing IGSCC.
- Research should be conducted to determine the extent of repair and the number of times a weld repair can be allowed to a given pipe to minimize property degradation. In the case of austenitic stainless steels, degradation includes degree and severity of sensitization.
- Research should be conducted to define the sensitivity of acoustic emission for detecting leaks from cracks under reactor operating conditions, for improving leak location detection capability, and for discriminating leakage from cracking versus leakage from sources such as seals.

Implications of NUREG-1061 Volume 1

In NUREG-0531 (1979) IGSCC in large-diameter (>20-inch) lines was discussed, and it was concluded that there was a probability of IGSCC occurring, though it should be less frequent and might not occur for a long period.

In March 1982, a hydrotest of Nine Mile Point Nuclear Station Unit 1 revealed leaks originating in very small pinholes or cracks in the heat-affected zones of the safe end-to-pipe welds. Leaks were found in the heat affected zones (HAZs) rather than in the furnace-sensitized 316 SS safe ends. Subsequent ultrasonic examinations revealed extensive cracking at many weld joints in the type 316 SS recirculation system. The entire recirculation system was replaced with type 316 NG material, a low carbon-nitrogen strengthened alloy.

These systems were subjected to UT ISI more than once. It was concluded that the UT operators were convinced that the large lines would not crack so crack phenomena observed with UT was attributed to geometric reflectors.

Information Notice No. 82-39 was issued in September 1982, followed by IE Bulletin No. 82-03 entitled, *Stress Corrosion Cracking in Thick Wall, Large Diameter, Stainless Steel Recirculation System Piping at BWR Plants*. This Bulletin required licensees of eight BWRs with scheduled outages to conduct ISI on the recirculation system. The results of these ISIs (five of seven plants reported cracking) led to IE Bulletin 83-02 entitled, *Stress Corrosion Cracking in Large-Diameter Stainless Steel Recirculation System Piping at BWR Plants*, in March 1983, which extended ISI to all other BWRs (Table 12). Nineteen of the twenty-three BWRs ultimately reported IGSCC.

Table 12: *Summary of all inspection findings on large piping in all operating BWRs inspected according to IEB 82-03 and 83-02.*

BWR Class and Plant Name	Extent of Inspection (% of welds inspected)		Inspection Results (No. of cracked welds)		No. of Welds Overlay Repaired
	Recirculation	RHR	Recirculation	RHR	
1 Big Rock Point	20% (11/59)	--	0	--	0
4 Browns Ferry 1	98% (103/105)	90% (36/40)	33	14	42
4 Browns Ferry 2	27% (25/91)	28% (9/32)	2	0	0
4 Browns Ferry 3	98% (103/105)	28% (9/32)	0	0	0
4 Brunswick 1	25% (29/115)	75% (3/4)	3	0	3
4 Brunswick 2	100% (102/102)	100% (5/5)	15	1	8
4 Cooper	100% (108/108)	100% (7/7)	20	0	13
3 Dresden 2	47% (47/101)	10% (4/40)	10	0	7
3 Dresden 3	100% (115/115)	90% (45/50)	53(1)	11(1)	61
4 Duane Arnold	42% (49/117)	40% (2/5)	0	0	0
4 FitzPatrick	47% (49/106)	45% (5/11)	1	0	0
4 Hatch 1	47% (47/100)	100% (11/11)	5	2	6
4 Hatch 2	94% (97/103)	100% (11/11)	36	3	27
3 Millstone 1	11% (11/100)	0% (0/46)	0	0	0
3 Monticello	100% (106/106)	78% (18/23)	6	0	6
2 Nine Mile Pt. 1	82% (62/76)	--	53	0	0
2 Oyster Creek	39% (31/80)	--	0	0	0
4 Peach Bottom 2	100% (91/91)	91% (32/35)	19	7	21
4 Peach Bottom 3	91% (77/85)	92% (35/38)	10	5	15
3 Pilgrim 1(2)	--	--	--	--	--
3 Quad Cities 1	8% (9/110)	20% (9/44)	0	0	0
3 Quad Cities 2	100% (106/106)	90% (45/50)	20	2	9
4 Vermont Yankee	66% (58/88)	7% (2/30)	33	1	22

(1) 18 welds originally reported to be cracked were later reevaluated and determined not to be cracked, and thus are not included in these totals.

(2) After inspecting approximately 7 welds, and finding cracks in 4 of them, the utility decided to replace the piping with Type 316NG, and thus has not completed the examination.

The fourth PCSG was established to review piping generically. The Pipe Crack Task Group (PCTG), a subset of the PCSG, concentrated of the SCC problem. Their report (NUREG-1061, 1984) contains nearly 23 pages of conclusions and recommendations, most of which can best be seen in Figures 7 and 8 and Table 13.

Figure 7 illustrates the interdependence of material, environment, and stress factors most correctable by long-term solutions. Generally, action on two or preferably all three variables is preferred.

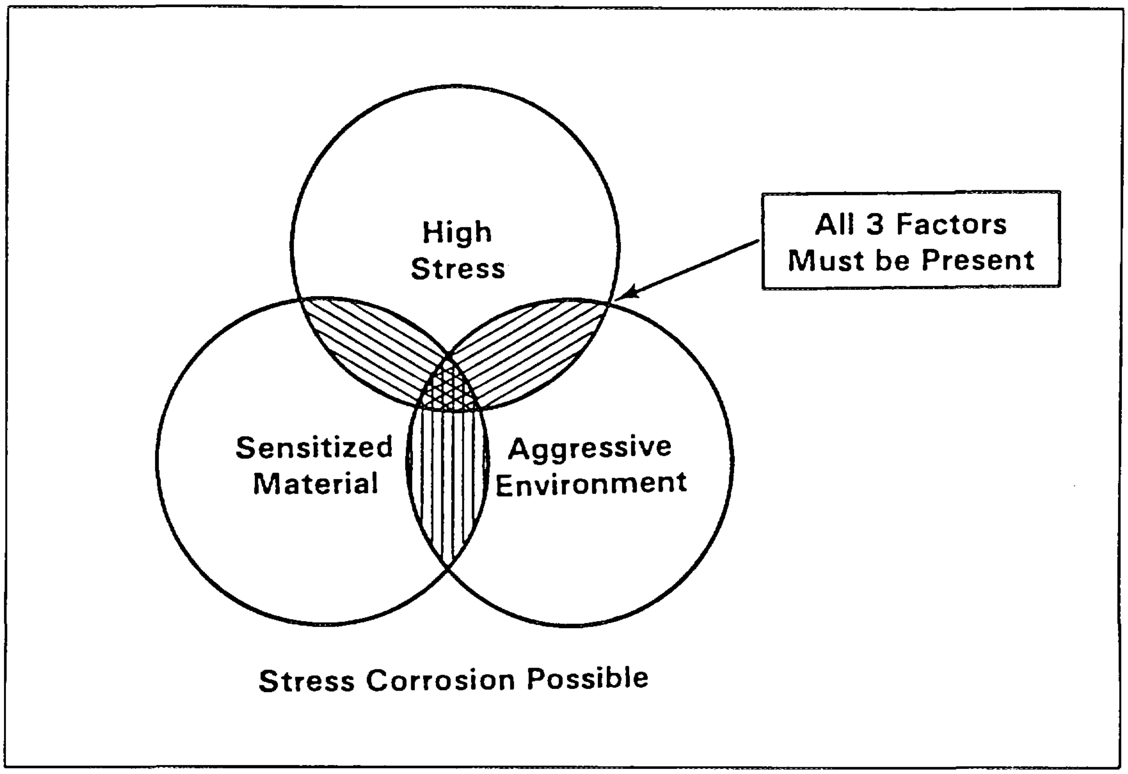


Figure 7: *Conditions leading to stress corrosion.*

Figure 8 shows both near-term and long-term solutions, as well as suggested actions related to volumetric examination and enhanced leak detection. The long-term materials solution is Type 316NG stainless steel in most countries, except Germany where a Type 347 NG is preferred.

Short-term solutions include induction heating stress improvement (IHSI), clam shell reinforcements and last pass heat sink welding (HSW). In Japan, IHSI on uncracked pipe is considered a permanent fix by reducing-reversing residual tensile stresses.

The third variable, environment, can be corrected with hydrogen water chemistry.

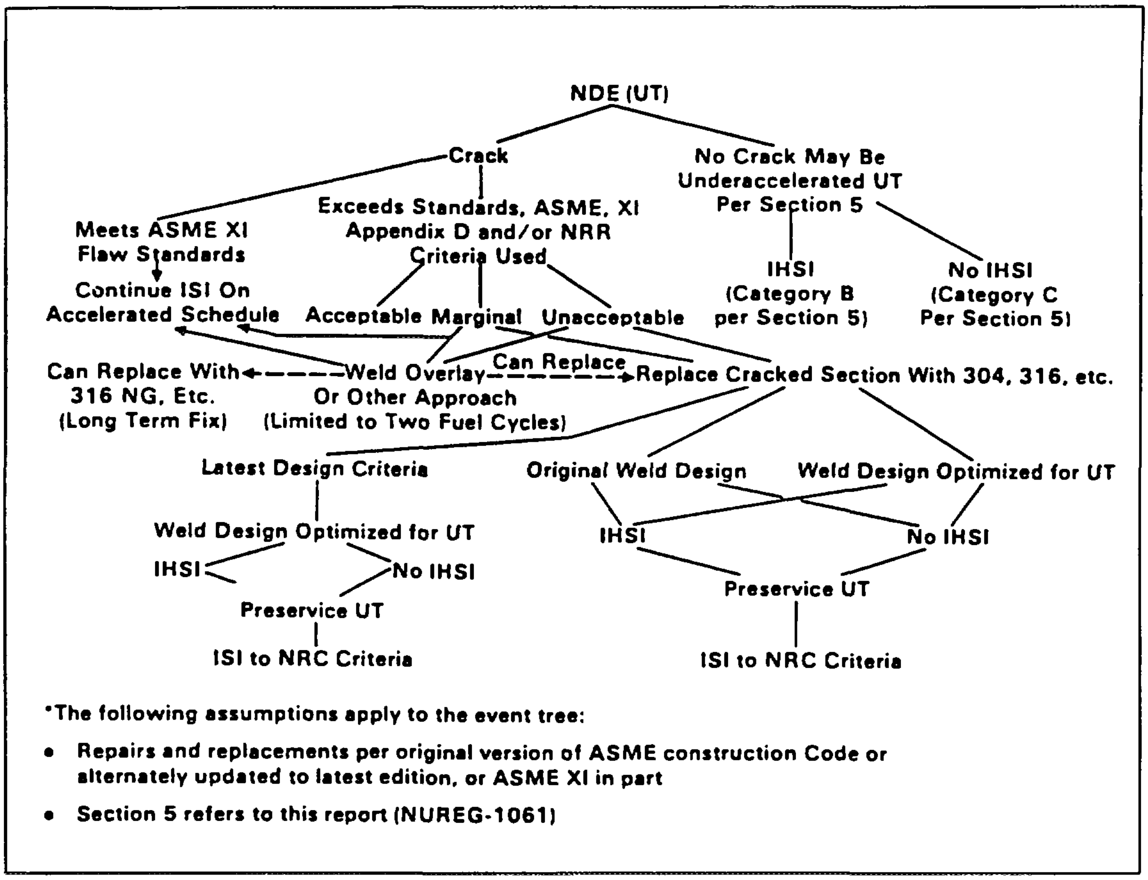


Figure 8: *Short-term solutions for replacement, repair, or continued operation without repair.*

Nondestructive Evaluation (NDE) Studies

The problem of detecting IGSCC was reviewed extensively as can be seen from the following conclusions and recommendations. The PCTG was even less optimistic than in NUREG-75/067, NUREG-0531, and NUREG-0691.

Conclusions of NDE Studies

- Code minimum UT procedures result in totally inadequate IGSCC detection. However, easily implemented modifications to these procedures can result in some improvement. These have been incorporated into Code Case N-335. Therefore, it is recommended that Code Case N-335 should immediately be made mandatory for all augmented inspections until better procedures are developed.

- Although IGSCC detection has improved to the point that it is considered acceptable under optimum conditions and procedures, the detection reliability as influenced by variability in operator procedure and equipment performance, along with field conditions, needs further study and improvement. Though length sizing of cracks is acceptable, depth sizing is currently inadequate. It is recommended that advanced techniques and procedures for crack detection and depth sizing continue to be developed and incorporated into Code requirements to provide data to reduce the need for extremely conservative fracture mechanics evaluation.

- For future plants or for replacement of existing piping systems, the material, design of pipe joints, and accessibility from both sides of the weld should be optimized for UT examinations; this requirement should be mandatory for all components with the exception of existing items such as pumps, valves, and vessels in older plants. Additionally, the joints that can not be inspected should be subjected to IHSI.

- Inspection techniques should be developed to detect and diminish flaws in pipes repaired by the weld overlay process.

- Because the UT examiner is one of the more erratic inspection variables, it is recommended that human factors research be performed to reduce the possibility of human error.

- Augmented inspections beyond those required by ASME Code Section XI should apply to all Types 304 and 316 austenitic piping systems operating over 200°F (93°C), unless they have been treated with effective countermeasures (complying with NUREG-0313 Revision 2).

- The degree of augmented inspection depends on the materials and processes used for each weld. The most frequent inspections are required for welds fabricated from nonresistant material.

Recommendations from NDE Studies

- All type 304 and 316 austenitic piping systems operating over 200°F (93°C) should receive augmented ISI, unless they have been treated with effective countermeasures.

- The extent and frequency of examinations should depend on the resistance of materials to IGSCC and the effectiveness of any processes used to prevent cracking.

- All UT examiners should attend industry-sponsored UT detection and sizing training courses and should continue to be evaluated on the cracked samples under the witness of third party personnel before participating in the field inspection.

- All BWR weld examinations should be performed in accordance with the latest version of Code Case N-335 and with the specific equipment and procedures used and personnel passed in the performance demonstration tests.

The information in Table 13 confirms that both short-term and long-term solutions are covered as is the accelerated NDE of NUREG-0313, Revision 2 (1986). Figure 8 presented an event tree approach that includes several of the solutions cited in Table 13. Note the caveat with respect to continued use of overlay clad. This has been relaxed on the basis of ability to conduct UT through the clad and no evidence of continued IGSCC growth.

Conclusions of Fracture Mechanics Studies

A few significant conclusions are cited with respect to fracture mechanics:

- The analysis of SCC growth in weldments using the techniques of linear elastic mechanics with "typical" values for the distribution of through-wall residual stresses and SCC growth rates yields results that are consistent with field and laboratory experience with welded pipes. Thus it is a reasonable approach to use this technique to analyze the remaining life of flawed weldments in-reactor and for detailed probabilistic models such as the work by Lawrence Livermore National Laboratory.

- Although conservative estimates of the through-wall growth of SCCs can be made with a high degree of confidence given an initial crack size, there is a high degree of uncertainty in the measurement of the depth of existing cracks, as discussed in Section 4 of NUREG-0313. This uncertainty must be adequately addressed in the development of acceptance criteria for flawed piping.

- Evaluation of through-wall cracks using weld metal properties shows that a surface crack breaking through the pipe wall under bounding accident loading conditions would have to be approximately 30% of the circumference in length for large-diameter pipes to result in unstable fracture of the pipe. However, leaks from cracks not large enough to cause gross failure of the pipe may be undesirably large.

- Operating experience suggests that leak-before-break is the most likely mode of failure for the vast majority of cracks occurring in service. This is a result of the asymmetry of the weld residual stresses and applied loads and the variability in material properties. Evaluations using conservative crack growth rate predictions and net section collapse analyses, applicable to cracks in very high toughness material, indicate that for the vast majority of possible crack geometries there exists significant capability to detect leakage and implement corrective actions. Evaluation using the fracture resistance properties of the weld material show substantial margins against failure, under normal and accident loading conditions for through-wall cracks that should be reliably detected by leakage.

- Flaw evaluation criteria should limit the length of the cracks accepted for continued operation without repair. The limitation on acceptable crack length is primarily a result of the lack of confidence in flaw depth sizing capability, and is intended to

Table 13: *Recommended Measures for Controlling IGSCC in BWR Piping.*

BWR Plant Status	Overlay Weld[1]	Replace with Similar Alloy	Residual Stress Improvement IHSI, HSW	Hydrogen Water Chemistry	316 NG Piping	New Baseline	Accel UT[2]	Normal UT	Enhanced Leak Detection[3]
Design Stage			X	X	X			X	
NTOL or Recent Startup			X	X		X		X	
Operating > 5 Years									
No Cracks Found			X	X		X	X	X[4]	X (maybe)
Only Limited Shallow Cracks			X	X		X	X[5]	X	X
Deeper Cracks -- Few or Many	X	X maybe	X[6]	X	X	X	X then	X (for 316 NG)	X prior to replacement

(1) Limited to two cycles unless convincing evidence is presented.
(2) Accelerated UT limited to cracked welds.
(3) Moisture tapes, AE, etc.
(4) Prior to mitigation, accelerated UT; therafter normal UT.
(5) With mitigation, accelerated UT limited to cracked welds.
(6) Also suggested after replacement with 316 NG.

ensure leak-before-break conditions. The maximum allowable through-wall crack length can be determined based on weld-joint specific loads.

- The recommendation for a limit on acceptable crack length is primarily a result of the lack of confidence in ultrasonic depth sizing capabilities. In this respect, the fracture mechanics calculations presented in this report demonstrate that there are acceptable margins against fracture for relatively long, deep surface cracks. Demonstration of reliable sizing of the part-through flaw depth would most likely allow relaxation of the limits on crack length.

- Additional fracture mechanics analyses, material properties characterization, and large-scale pipe tests should be performed to further the understanding of the implications of stainless steel weld and cast material fracture toughness properties in flawed pipe evaluations. Furthermore, it is recommended that the NRC provide active support of the ASME task group currently evaluating the concerns that have been raised regarding IWB-3640.

- Since operating experience and fracture mechanics evaluations indicate that leak-before-break is the most likely mode of piping failure, it is recommended that feasible leak detection procedures be in effect in operating plants. Current sump-pump monitoring systems are sensitive enough to provide an additional margin against leak-before-break, if more stringent requirements on surveillance intervals and unidentified leakage are imposed (see Section 4 of NUREG-0313). Therefore, it is recommended that the limits on unidentified leakage in BWRs be decreased to 3 gpm and that the surveillance interval be decreased to 4 hours or less.

The following conclusions apply to IHSI, water chemistry, and selection of 316NG SS:

- The use of additional mitigating procedures such as IHSI or hydrogen water chemistry to provide an additional margin against environmentally assisted cracking is strongly recommended, even after replacement with Type 316NG SS. Utilities should be encouraged to adopt an ALARA approach to coolant impurity levels.

- Although low-carbon stainless steels with nitrogen additions have been successfully fabricated and welded in Japan and Europe, US experience with these materials is limited. It appears that greater care must be exercised in the control of composition and fabrication variables to limit cracking during hot forming or welding.

- IHSI is considered to be a more effective mitigating action for IGSCC than HSW and LPHSW because more data are available to demonstrate that the process does produce a more favorable residual stress state. All the residual stress improvement remedies are considered to be more effective when applied to weldments with no reported cracking.

- BWR water chemistry controls should be modified to minimize IGSCC. These modifications should include both a substantial reduction in the levels of ionic species entering the primary coolant and a control of oxygen level. The current work on reduction of oxygen through hydrogen additions should be followed closely with the possibility that it may be employed to reduce further the electrochemical potential of the stainless steel to a level at which SCC, either IGSCC or TGSCC, will not occur. It appears that hydrogen water chemistry is an effective IGSCC countermeasure. However, ongoing work regarding potential adverse effects on other reactor components should be closely followed in order to confirm the acceptability of this countermeasure.

- The effects of hydrogen water chemistry on the balance of the system and on overall long-term plant operation need to be explored in greater depth before they can be recommended without reservation for adoption on a wide scale.

Conclusions and Recommendations from IGSCC Risk Studies

Conclusions from the IGSCC Risk Studies are as follows:

- Intergranular stress-corrosion cracking has caused no significant loss-of-coolant events or loss-of-coolant accidents (LOCA) in BWRs even though its presence in 304 stainless steel coolant piping has been found in many operating plants. This may be due to an aggressive ISI and repair program and/or a tendency to leak-before-break (with effective leak detection). The LOCA contribution to the core damage frequency as calculated by probabilistic risk assessment (PRA) studies for BWRs is a minor contribution (generally less than 10%) for all LOCA sizes. Because the LOCA contribution to the core damage frequency is small, it would require a considerable error, at least an order of magnitude, from IGSCC before LOCAs would be a dominant contributor to the core damage frequency.

- IGSCC is but one component of the LOCA occurrence rate. Design and construction errors, maintenance errors, cyclic and thermal fatigue, etc. all contribute to the LOCA probability. It is estimated that IGSCC may be 30% or more of the LOCA frequency in BWRs as deduced from experience, the total industrial data base, and engineering judgment. Therefore, an error in the IGSCC LOCA contribution of 20-30% would be required for an order of magnitude error in the LOCA probability.

- The piping failure data base and available studies were reviewed for evidence of IGSCC impact. To a considerable extent, uncertainties have always existed in determining piping failure rates. Largely, this uncertainty is intrinsic in attempts to measure low-frequency events from experience, and it makes the effect of trends difficult to analyze. It was statistically reasoned that an error of an order of magnitude in the large pipe failure frequencies (presumably from IGSCC) would have produced, with one chance in four, at least one failure during this interval. Therefore, it was concluded that the frequency for a large pipe failure adequately accounts for the contribution due to IGSCC.

- Overall it is concluded that the presence of IGSCC in BWRs may reduce safety margins believed to exist, but there is no apparent evidence to conclude that IGSCC has increased public risk significantly.

Piping failure rates are difficult to determine because failures of high quality piping occur infrequently and require a data base acquired over a large number of reactor years, since normal experimental procedures are impractical. This difficulty is unlikely to be overcome now or in the foreseeable future. What results is an estimated frequency with large uncertainties from aggregated events acquired from poor quality documentation. Classifying this frequency according to contributing phenomena in a meaningful way is very difficult. The following are the recommendations from IGSCC Risk Studies:

- Undertake a formal study and updating of the pipe failure data base be undertaken. Some statistics may be meaningful only when derived or assessed analytically. The NRC has developed a computer code called PRAISE (Piping Reliability Analysis Including Seismic Events) that, based on current knowledge, is currently being modified to include a model for IGSCC.

- Use the PRAISE computer code to: a) investigate the impact of IGSCC in primary piping according to pipe size; b) study the failure frequencies of components due to IGSCC, fatigue, etc.; and c) calculate the conditional probability of multiple failures associated with IGSCC to check the point value and to test the assumption of a log normal distribution.

It is obvious that NUREG-1061, Volume 1 (1984) was a major effort. In fact much of the work since the completion of that report in 1984 was to resolve questions related to the conclusions and to respond to the recommendations. For example, a major effort by ASME Subcommittee XI has effectively resolved the concerns related to UT. Also, the probabilistic study on SCC has been completed and action has proceeded on hydrogen chemistry. As cited previously, considerable work has been done on weld overlays to permit their continued use.

One objective of the PCTG was to evaluate foreign IGSCC experience. This was done. Most countries with BWRs have reported IGSCC. A specific meeting confirmed that BWRs in Germany, Italy, Japan, Spain, Sweden, and Switzerland had experienced substantial IGSCC. Prior reporting had confirmed IGSCC in BWRs in Belgium, India, and the Netherlands.

NUREG-0313 Revision 2

NUREG-0313 (1986) contains specific requirements for expanded ISI in line with the recommendations of NUREG-1061, Volume 1 (1984). The report differentiates among resistant materials (A), nonresistant materials such as Type 304 SS that were given a stress improvement (SI) such as IHSI within 2 years of operation (B), the same as B except SI occurred after 2 years of operation (C), nonresistant materials with no SI (D), cracked nonresistant material with weld overlay or SI (E), crack nonresistant material with inadequate or no repair (F), nonresistant and not inspected (G). Table 14 covers the above categories and the accelerated ISI requirements for nonresistant materials.

Table 14: *Requirements for inservice inspection of BWR welds as a function of resistance to IGSCC (NUREG-0313, 1986).*

Description of Weldments	Notes	IGSCC Category	Inspection Extent & Schedule
Resistant Materials		A	All bimetallic every 10 yrs All terminal ends every 10 yrs 25% of other welds every 10 yrs
Nonresistant Materials SI within 2 yrs of Operation	(1)	B	All bimetallic every 10 yrs All terminal ends every 10 yrs 50% others every 10 yrs
Nonresistant Materials SI after 2 yrs of Operation	(1)	C	All next 3 1/3 yrs then all next 10 yrs
Nonresistant Material No SI	(1)	D	All every 3 1/3 years
Cracked Overlayed or SI	(1), (2)	E	All next outage then All every 3 1/3 yrs
Cracked Inadequate or No Repair	(2)	F	All every outage
Nonresistant Not Inspected	(3)	G	All next outage

(1) All welds in nonresistant material should be inspected after a stress improvement process as part of the process. Schedules shown should be followed after this initial inspection.
(2) See requirements for acceptable weld overlay reinforcements and stress improvement mitigation in Section 4 of NUREG-0313 Revision 2.
(3) Welds that are not inspectable should be replaced, "sleeved," or local leak detection applied.

Other Failures

Previously, the Hinkley Point turbine failure was cited where IGSCC in a disc led to its failure and penetration of the turbine casing. Three other nuclear turbines have had disc failures initiated by IGSCC. These were at Shippingport, Rancho Seco, and Yankee Rowe. All occurred in the latter half of the 1970s. None penetrated the casing. Several nuclear turbines now operating likely have cracks due to IGSCC in the discs. These cracks are monitored routinely so that the turbines can continue to run until replaced when the cracks approach critical size.

The Fourth Period: 1985 to 1991

It is apparent that substantial corrective actions have been taken on the basis of the conclusions and recommendation of the following references: NUREG-75/067, NUREG-0531, NUREG-0691, NUREG-1061, and NUREG-0313. The period from 1985 to 1991 was reviewed to see if corrosion problems had been resolved or minimized, or whether specific studies shed more light on the IGSCC problem.

NUREG/CR-4792, Volume 3 (1986) is a probabilistic study of failure in BWR piping induced by IGSCC. The PRAISE-CC computer code was used. Figure 9 is a schematic representation of the various components that comprise PRAISE-CC applied to Type 304 SS and Type 316NG SS. Input data included constant elongation rate testing (CERT) as well as constant load testing (CL). Pipe sizes investigated were outside diameters (O.D.) of <10-inch, 10-20 inch, and >20 inch Not surprisingly, the calculated leak probabilities due to SCC in type 304 SS weldments were substantially higher than in PWR piping where fatigue was the controlling mechanism. With Type 316NG SS, the calculated leak probability decreased by a factor of 500.

Corrosion-Related Incidents in BWRs and PWRs

A review of Information Notices (IN), Nucleonics Week, etc., revealed a few corrosion-related failures, with most occurring in PWRs. These failures are examined by reactor type.

Both BWRs and PWRs

Irradiation-assisted stress corrosion cracking (IASCC) has been seen in both BWRs and PWRs. In 1985 IASCC was reported in bolting in the upper core grid of a foreign BWR. The material closely resembled 316 ELC and the bolts had been exposed to a fast neutron fluence of 2E 20n/cm^2 .

The Chooz PWR in France suffered IASCC in Inconel X-750 core barrel bolts in 1985. Other PWRs have reported cracked core barrel bolts even in annealed low strength stainless steel.

A Type 410 SS valve stem in a valve at Brunswick 2 suffered IGSCC. This was attributed to a material that was excessively hard because of improper heat treatment. Similar IGSCC in 410 SS has been observed at Farley 1 (PWR), Browns Ferry 3

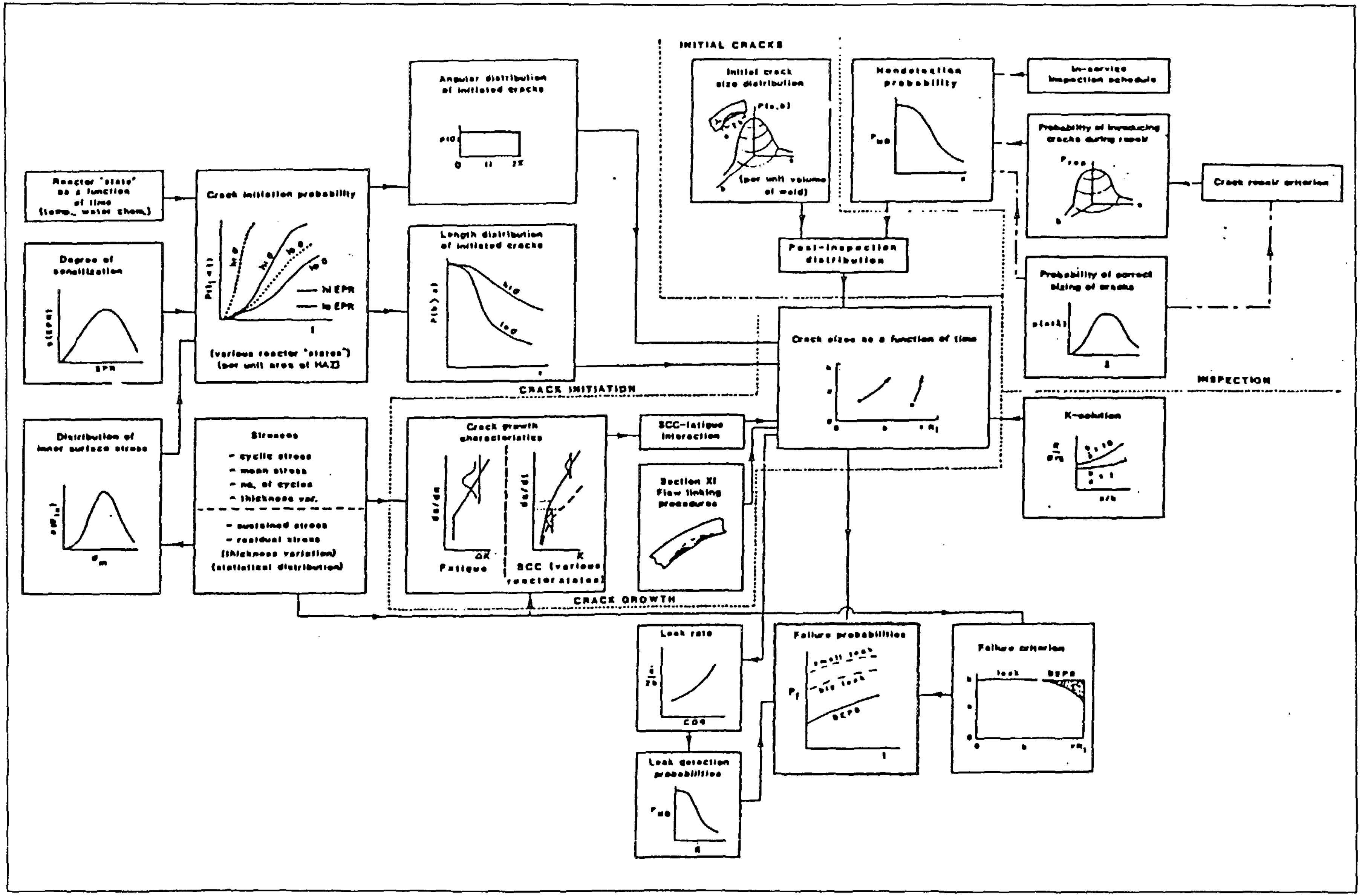


Figure 9: *Schematic Diagram of Various Components of an Expanded PRAISE Model Suitable for Application to Stress Corrosion Cracking.*

(BWR), and Oconee 1 (PWR). Corrective action was to replace the 410 SS with A-564 Grade 630 SS.

BWRs

The Dodewaard reactor in the Netherlands reported a crack like indication in a coolant purification safe-end located beneath the reactor core. It was ascribed to IGSCC. This problem represents a continuation of the safe end problem cited previously.

PWRs

Several Information Notices reported SCC in PWR components. These are grouped by component or location.

Steam generators have been subject to SCC for the past two decades; however, a discussion of such SCC is beyond the scope of this report. The two Information Notices (Ins), 89-65 and 90-49, cover SCC in the primary coolant Inconel-600 tube plugs that were rolled in. The observed cracking, believed to be SCC, was limited to specific heats of Inconel-600. At least four PWRs have reported such cracking.

Secondary side of steam generator tubes has occurred in many PWRs. IN 90-49 (1990) cited such cracking in nine PWRs. Such cracking has been observed in the support plates also. The probability is that such cracking will continue to plague PWR steam generators.

The cracking of Inconel-600 was cited previously (IN 89-65, 1989). Similar primary water SCC has been seen in Inconel-600 pressurizer heater thermal sleeves and instrument nozzles (IN 90-10, 1990). This "Corrosion Cracking" has been a recurrent problem in both domestic and foreign PWRs.

IGSCC has been observed in safety injection accumulator nozzles (NUREG/CR-7492, 1986). High stress and borated water were probable contributors in one case. At another plant the material was sensitized; however, there was no obvious corrosion environment being exposed to gaseous nitrogen. Since this occurred in more than one nozzle, and were all fabricated by one company, the possibility exists that fabrication defects played a role.

The IN 90-68 (1990) cites SCC in pump bolting of A-286 (A453 Grade 660). This has been a recurrent problem in both PWRs and BWRs, particularly with improper heat treatment and overly high hardness.

Lessons Learned

It has been over thirty years since the early prototype reactor began to operate. Several events relevant to SCC have occurred during this period. In some instances the corrective actions have resolved specific problems: in other instances the industry is still plagued with the problems. The following is a brief recapitulation of lessons learned, accomplishments, and problems remaining to haunt the reactor industry.

Common to LWRs

Irradiation-assisted stress corrosion has not been resolved; however, it is difficult to determine how severe a problem it will become in time. One might say that the "jury is still out".

PWRs

The incidents of primary water stress corrosion cracking in PWRs appear to be increasing. Cases in steam generators, in austenitic stainless steels, and Inconels have been known for years. Now it is occurring in safe-ends and piping at very low oxygen levels. Secondary side water chemistry must be controlled to prevent SCC in PWRs.

BWRs

- The corrective actions cited in this study appear to have led to a substantial reduction of IGSCC. One can be cautiously optimistic that extensive IGSCC should not occur. However, so long as the combined conditions of stress, sensitization, and oxygen exist, we should expect to see IGSCC.

- Control of carbon level, either through use of ELC grades or NG grades of austenitic stainless steel, should minimize IGSCC.

- Control of residual stresses, particularly with IHSI, greatly reduces the incidence of IGSCC.

- Hydrogen water treatment controls the oxygen and should limit IGSCC. Whether other problems arise is yet to be determined. An example is erosion-corrosion if the oxygen is low in the carbon steel lines.

- The problem with furnace-sensitized safe ends is well recognized and should not recur.

- In most cases, severe circumferential SCC should lead to detectable leakage so that leak-before-break can be identified; however, there can be special cases where the crack size approaches instability. A rule of thumb is that a through-wall crack of 30% of the circumference is stable. This should be detectable by leakage.

- IGSCC of austenitic stainless steels can occur in all pipe sizes from smallest to largest, especially when stress, sensitization, and oxygen are all present.

Detection of IGSCC

- Leak detection is a good indication of through-wall cracks before they approach instability. In particularly tight cracks, however, the process may be marginal.

- Acoustic emission will detect small leaks and may detect cracking; however, it is dependent on where transducers are located and how many are used.

- Ultrasonic detection methods have been used most often. A review of NUREG -0531, -0691, and -1061 reveals the increased uncertainty concerning the ability of UT to detect and size IGSCC. Actions taken by ASME Subcommittee XI have improved the situation. Operators will need to qualify on actual IGSCC (or other forms of cracks) before being permitted to conduct UT. This is in Appendix VIII of Section XI. A similar procedure was used at the EPRI NDE Center with considerable success. With correct use of UT, the probability of detection should be acceptable, though sizing capability still is a problem.

Although there has been considerable progress in controlling and mitigating IGSCC in nuclear plants, problems still exist.

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