Materials Degradation Matrix – First Level

Rev. 0 (8/6/04)

The first level of the DM presents an overview in terms of the reactor type and component category (below), the material used (next page) or the degradation mechanism concerned (page 3):

PWR						BWR		
PWR Reactor Pressure Vessel	**PWR Pressurizer**	**PWR SG Shell**	**PWR Reactor Internals**	**PWR Piping**	**PWR SG Tubes & Internals**	**BWR Pressure Vessel**	**BWR Reactor Internals**	**BWR Piping**

Clicking in any of the boxes of this table leads directly to the appropriate section at the second level of the DM. This consists of a set of separate tables, each of which is organized as follows:

- The X direction represents the specific degradation mechanism under consideration. Further explanations are provided as hyperlinks to a separate section containing summary information: Mechanisms.
- The Y direction represents both the component under consideration (sometimes with hyperlinks to endnotes providing specific information) and the class of materials being considered. Again, hyperlinks are provided to a separate section containing further information on material behavior: Materials.
- The body of the table represents the actual assessment as to whether or not a particular degradation mechanism is active for each class of material in the reactor type and component category under consideration. It is organized into three categories as follows:
- "Y" signifies that at least one member of the indicated class of materials is susceptible to damage by the indicated degradation mechanism under service conditions relevant to the indicated combination of component and reactor type; "N" signifies that the indicated class of materials is not susceptible to damage by the indicated degradation mechanism under service conditions relevant to the indicated combination of component and reactor type; "?" signifies that it is uncertain whether or not the indicated degradation mechanism is applicable to the indicated combination of materials, component and reactor type; and "N/A" signifies that the indicated combination of degradation mechanism, materials, component and reactor type is not relevant for some reason.
- The green color-coding signifies that the indicated combination of degradation mechanism, class of materials, component and reactor type has been well-characterized and that little, if any additional R&D should be needed; the yellow color-coding signifies that R&D is ongoing that should resolve current uncertainties in the reasonably near term and the orange color-coding signifies that insufficient R&D is ongoing to resolve current uncertainties in an acceptable time-frame. The blue color-coding applies only to cells with a question mark (see above) and indicates that insufficient R&D is ongoing to resolve the pertinent applicability issue in the reasonably near future.
- Level 3: Explanatory notes on individual issues referenced within the Level 2 tables are provided as hyperlinks (in blue within cells) to the information contained as endnotes in this document immediately after the Level 2 tables.

Materials Degradation Matrix – First Level

Rev. 0 (8/6/04)

Degradation Matrix overview in terms of the class of materials used (see hyperlinks underlined in blue for summary information).

An entry in the table below[1] implies at least one "Y" or "?" in the DM for this class of material with regard to the referenced reactor type and component category.

Carbon / Low-Alloy Steel C&LAS C&LAS Welds	Stainless Steel Wrought SS SS Welds & Clad	Cast SS CASS	Nickel-based Alloys Wrought Ni Alloys Ni-base Welds & Clad
PWR Reactor Pressure Vessel	PWR Reactor Pressure Vessel		PWR Reactor Pressure Vessel
PWR Pressurizer	PWR Pressurizer		PWR Pressurizer
PWR SG Shell	PWR SG Shell		PWR SG Shell
	PWR Reactor Internals	PWR Reactor Internals	PWR Reactor Internals
PWR Piping	PWR Piping	PWR Piping	PWR Piping
PWR SG Tubes & Internals	PWR SG Tubes & Internals		PWR SG Tubes & Internals
BWR Pressure Vessel	BWR Pressure Vessel		
	BWR Reactor Internals	BWR Reactor Internals	BWR Reactor Internals
BWR Piping	BWR Piping	BWR Piping	BWR Piping

[1] Clicking in any of the boxes of this table leads directly to the appropriate section at the second level of the DM, which consists of a set of separate tables.

Materials Degradation Matrix – First Level

Rev. 0 (8/6/04)

Degradation Matrix overview in terms of degradation mechanisms (see hyperlinks underlined in blue for summary information).

An entry in the table below[2] implies at least one "Y" or "?" in the DM for this main type of mechanism with regard to the referenced reactor type and component category.

SCC (EAC) SCC	**Corrosion** C & W	**Wear** C & W	**Fatigue** Fat.	**Reduction in Toughness** RiT	
				Thermal Aging	***Irradiation***
PWR Reactor Pressure Vessel	PWR Reactor Pressure Vessel		PWR Reactor Pressure Vessel	PWR Reactor Pressure Vessel	PWR Reactor Pressure Vessel
PWR Pressurizer	PWR Pressurizer		PWR Pressurizer	PWR Pressurizer	
PWR SG Shell	PWR SG Shell		PWR SG Shell	PWR SG Shell	
PWR Reactor Internals		PWR Reactor Internals	PWR Reactor Internals	PWR Reactor Internals	PWR Reactor Internals
PWR Piping	PWR Piping		PWR Piping	PWR Piping	
PWR SG Tubes & Internals	PWR SG Tubes & Internals	PWR SG Tubes & Internals	PWR SG Tubes & Internals	PWR SG Tubes & Internals	
BWR Pressure Vessel	BWR Pressure Vessel	BWR Pressure Vessel	BWR Pressure Vessel	BWR Pressure Vessel	BWR Pressure Vessel
BWR Reactor Internals			BWR Reactor Internals	BWR Reactor Internals	BWR Reactor Internals
BWR Piping	BWR Piping	BWR Piping	BWR Piping	BWR Piping	

[2] Clicking in any of the boxes of this table leads directly to the appropriate section at the second level of the DM. This consists of a set of separate tables.

Materials Degradation Matrix – Second Level

Rev. 0 (8/6/04)

PWR **Component**	**Material**	**SCC** **SCC**					**Corrosion/Wear** **C & W**				**Fatigue** **Fat.**			**Reduction in Toughness** **RiT**					
														Aging	***Irradiation***				
	Subdivision[3] →	**IG**	**IA**	**TG**	**LTCP**	**PW**	**Wstg**	**Pit**	**Wear**	**FAC**	**HC**	**LC/Th**	**Env**	**Th**	**Emb**	**VS**	**SR**	**Th_n**	**Fl**
PWR Reactor Pressure Vessel e001	**C&LAS**	? e002	N	? e002	N	? e003	Y e004	N	N	Y e005	N	Y e006	Y e007	Y e008	Y e009	N	N	N	Y e010
	C&LAS Welds	? e002	N	? e002	N	? e003	Y e004	N	N	Y e005	N	Y e006	Y e007	Y e008	Y e011	N	N	N	Y e010
	Wrought SS	? e012	N	? e012	? e013	? e012	N	N	N	N	N	Y e014	Y e015	N	N	N	N	N	N
	SS Welds & Clad	Y e016	? e017	Y e018	? e013	? e019	N	N	? e020	N	N	? e021	Y e015	Y e022	Y e022	N	N	N	N
	Wrought Ni Alloys	N	N	N	? e023	Y e023	N	N	N	N	Y e014	Y e014	Y e015	N	N	N	N	N	N
	Ni-base Welds & Clad	N	? e024	N	Y e023	Y e025	N	N	N	N	N	Y e014	Y e015	N	N	N	N	N	N
PWR Pressurizer **(Including Shell, Surge and Spray Nozzles, Heater Sleeves and Sheaths, Instrument Penetrations)**	**C&LAS**	? e002	N	? e002	N	? e003	Y e004	N	N	Y e005	N	Y e006	Y e007	Y e008	N/A	N/A	N/A	N/A	N/A
	C&LAS Welds	? e002	N	? e002	N	? e003	Y e004	N	N	Y e005	N	Y e006	Y e007	Y e008	N/A	N/A	N/A	N/A	N/A
	Wrought SS	? e012	N	? e012	? e013	? e012	N	N	N	N	N	Y e014	Y e015	N	N/A	N/A	N/A	N/A	N/A
	SS Welds & Clad	Y e016	? e017	Y e018	? e013	? e019	N	N	? e020	N	N	? e014	Y e015	Y e022	N/A	N/A	N/A	N/A	N/A
	Wrought Ni Alloys	N	N	N	? e023	Y e023	N	N	N	N	Y e014	Y e014	Y e015	N	N/A	N/A	N/A	N/A	N/A
	Ni-base Welds & Clad	N	? e024	N	Y e023	Y e025	N	N	N	N	N	Y e014	Y e015	N	N/A	N/A	N/A	N/A	N/A

[3] IG = intergranular; IA = irradiation assisted; TG = transgranular; LTCP = low temperature crack propagation; PW = primary water; Wstg = wastage; FAC = flow accelerated corrosion; HC = high cycle; LC = low cycle; Th = thermal; Env = environmental; Emb = embrittlement (from fluence); VS = void swelling; SR = stress relaxation; Fl = flux effect

Materials Degradation Matrix – Second Level

Rev. 0 (8/6/04)

PWR	**Material**	**SCC** **SCC**					**Corrosion/Wear** **C & W**				**Fatigue** **Fat.**			**Reduction in Toughness** **RiT**					
														Aging	***Irradiation***				
	[4]Subdivision→	**IG**	**IA**	**TG**	**LTCP**	**PW**	**Wstg**	**Pit**	**Wear**	**FAC**	**HC**	**LC/Th**	**Env**	**Th**	**Emb**	**VS**	**SR**	**Th_n**	**Fl**
PWR SG Shell (Including Feedwater and Main Steam Nozzles)	**C&LAS**	? e026	N	Y e136	N	? e003	Y e027	Y e028	N	Y e005	N	Y e006	Y e030	Y e008	N/A	N/A	N/A	N/A	N/A
	C&LAS Welds	? e026	N	Y e136	N	? e003	Y e027	Y e028	N	Y e005	N	Y e006	Y e030	Y e008	N/A	N/A	N/A	N/A	N/A
	Wrought SS	? e012	N	? e012	? e013	? e012	N	N	N	N	N	Y e006	Y e015	N	N/A	N/A	N/A	N/A	N/A
	SS Welds & Clad	Y e016	? e017	Y e018	? e013	? e019	N	N	? e020	N	N	? e006	Y e015	Y e022	N/A	N/A	N/A	N/A	N/A
	Wrought Ni Alloys	N	N	N	? e031	Y e023	N	N	N	N	Y e032	Y e006	Y e015	N	N/A	N/A	N/A	N/A	N/A
	Ni-base Welds & Clad	N	? e024	N	Y e023	Y e025	N	N	N	N	N	Y e006	Y e015	N	N/A	N/A	N/A	N/A	N/A
PWR Reactor Internals e033 e034 e035	**Wrought SS**	N	Y e036	N	? e013	Y e037	N	N	Y e038	N	Y e039	N	Y e040	N	Y e041	Y e042	Y e042	Y e043	? e044
	CASS	N	Y e045	N	? e013	Y e045	N	N	Y e038	N	Y e035	N	Y e035	Y e046	Y e046	N e047	N e047	N e047	? e044
	SS Welds & Clad	N	Y e045	N	? e013	Y e045	N	N	N	N	N	N	N	Y e022	Y e048	Y e042	Y e042	Y e042	? e049
	Wrought Ni Alloys	N	? e050	N	Y e051	Y e052	N	N	? e020	N	N	N	N	N	Y e053	? e053	? e054	N	? e053

[4] IG = intergranular; IA = irradiation assisted; TG = transgranular; LTCP = low temperature crack propagation; PW = primary water; Wstg = wastage; FAC = flow accelerated corrosion; HC = high cycle; LC = low cycle; Th = thermal; Env = environmental; Emb = embrittlement (from fluence); VS = void swelling; SR = stress relaxation; Fl = flux effect

Materials Degradation Matrix – Second Level

Rev. 0 (8/6/04)

PWR	Material	SCC SCC					Corrosion/Wear C & W				Fatigue Fat.			Reduction in Toughness RiT *Thermal Aging*
Component	[5]*Subdivision*→	IG	IA	TG	LTCP	PW	Wstg	Pit	Wear	FAC	HC	LC/Th	Env	
PWR Piping **(Including Safe Ends, Feedwater and Main Steam Piping)**	**Wrought SS**	N	N/A	Y e055	? e013	Y e019	N	N	N	N	Y e014	Y e021	Y e056	N
	CASS	N	N/A	? e055	? e013	? e019	N	N	N	N	Y e014	Y e006	Y e056	Y e057
	C&LAS	? e002	N/A	? e002	N	? e003	Y e004	Y e058	N	Y e058	Y e014	Y e006	Y e007	Y e008
	C&LAS Welds	? e002	N/A	? e002	N	? e003	Y e004	Y e058	N	Y e058	Y e014	Y e006	Y e007	Y e008
	Wrought Ni Alloys e133	N	N/A	N	? e031	Y e023	N	N	N	N	N	? e073	? e073	N
	SS Welds & Clad	N	N/A	Y e018	? e013	? e019	N	N	N	N	Y e014	Y e014	Y e014	Y e022
	Ni-base Welds & Clad	N	N/A	N	Y e031	Y e059	N	N	N	N	Y e014	Y e014	Y e014	Y e057

[5] IG = intergranular; IA = irradiation assisted; TG = transgranular; LTCP = low temperature crack propagation; PW = primary water; Wstg = wastage; FAC = flow accelerated corrosion; HC = high cycle; LC = low cycle; Th = thermal; Env = environmental; Emb = embrittlement (from fluence); VS = void swelling; SR = stress relaxation; Fl = flux effect

Materials Degradation Matrix – Second Level

Rev. 0 (8/6/04)

PWR SG Tubes & Internals	Material	SCC SCC					Corrosion/Wear C & W				Fatigue Fat.			Reduction in Toughness RiT *Thermal Aging*
	[6]Subdivision→	IG	IA	TG	LTCP	PW	Wstg	Pit	Wear	FAC	HC	LC/Th	Env	
TUBES	**Wrought Ni Alloys 600ma**	Y e060	N/A	Y e061	? e013	Y e062	Y e063	Y e064	Y e065	N	Y e065	N	Y e066	? e067
	Wrought Ni Alloys 600tt	Y e068	N/A	Y e061	? e013	Y e069	Y e070	Y e071	Y e072	N	Y e072	N	Y e073	N/A
	Wrought Ni Alloys 690tt	Y e074	N/A	Y e061	Y e075	? e076	? e077	Y e073	Y e078	N	Y e078	N	Y e073	? e079
	Wrought Ni Alloys 600sen (OTSG)	Y e065	N/A	Y e080	? e013	Y e082	N	Y e083	Y e084	Y e085	Y e072	N	Y e086	N/A
SEAL WELDS e089	**Ni-base Welds & Clad**	N/A	N/A	N	Y e075	Y e081	N	N	Y e087	N	N	? e088	N	N/A
INTERNALS	**C&LAS Wrought SS**	? e019	N/A	Y e090	? e013	N/A	Y e091	Y e091	N	Y e092	N	N	N	? e093
TUBE PLUGS/SLEEVES	**Wrought Ni Alloys**	N	N/A	N	? e013	Y e094	N	N	N	N	N	N	N	N/A
DIVIDER PLATE	**Wrought Ni Alloys**	N/A	N/A	N	? e013	Y e095	N	N	N	N	N	N	N	N/A

[6] IG = intergranular; IA = irradiation assisted; TG = transgranular; LTCP = low temperature crack propagation; PW = primary water; Wstg = wastage; FAC = flow accelerated corrosion; HC = high cycle; LC = low cycle; Th = thermal; Env = environmental

Materials Degradation Matrix – Second Level

Rev. 0 (8/6/04)

BWR Component	Material	SCC SCC				Corrosion/Wear C & W				Fatigue Fat.			Reduction in Toughness RiT					
													Aging	*Irradiation*				
	Subdivision→[7]	IG	IA	TG	LTCP	Wstg	Pit	Wear	FAC	HC	LC/Th	Env	Th	Emb	VS	SR	Th	Fl
BWR Pressure Vessel	**C&LAS**	? e096	? e097	Y e098	N	Y e099	Y e099	N	N	N	Y e100	Y e101	Y e102	Y e009	N	N	N	Y e010
	C&LAS Welds	? e096	? e097	Y e098	N	Y e099	Y e099	N	N	N	Y e100	Y e101	Y e102	Y e009	N	N	N	Y e010
(Including stainless steel and Ni-base penetrations)	**Wrought SS**	Y e103	N	Y e104	? e013	N	N	N	N	Y e032	Y e105	Y e007	N	N	N	N	N	N
	SS Welds & Clad	Y e106	Y e107	Y e108	? e013	N	N	N	N	Y e032	Y e109	Y e109	Y e022	Y e022	N	N	N	N
	Wrought Ni Alloys	Y e110	Y e050	N	Y e031	N	N	N	N	Y e032	N	Y e053	Y e057	Y e057	N	N	N	Y e057
	Ni-base Welds & Clad	Y e110	Y e050	N	Y e031	N	N	N	N	Y e032	N	Y e053	Y e057	Y e057	N	N	N	Y e057
BWR Reactor Internals	**Wrought SS**	Y e103	Y e045	Y e111 e104	? e013	? e112	N	Y e113	N	Y e114	N	Y e014	N	Y e045	N	Y e115	Y e116	? e117
e135	**SS Welds & Clad**	Y e103	Y e045	Y e111 e104	? e013	? e112	N	Y e113	N	Y e114	N	Y e014	Y e118	Y e045	N	Y e115	Y e116	? e117
	CASS	Y e119	Y e045	N	? e013	N	N	N	N	N	N	N	Y e057	Y e045	N	N	N	N
	Ni-base Welds & Clad	Y e120	Y e045	N	Y e031	N	N	N	N	Y e121	N	Y e040	? e045	Y e122	N	Y e123	N	N
	Wrought Ni Alloys	Y e124	Y e053	N	Y e051	N	N	N	N	Y e040	N	Y e040	? e045	Y e122	N	Y e123	N	N

[7] IG = intergranular; IA = irradiation assisted; TG = transgranular; LTCP = low temperature crack propagation; PW = primary water; Wstg = wastage; FAC = flow accelerated corrosion; HC = high cycle; LC = low cycle; Th = thermal; Env = environmental; Emb = embrittlement (from fluence); VS = void swelling; SR = stress relaxation; Fl = flux effect

Materials Degradation Matrix – Second Level

Rev. 0 (8/6/04)

BWR	Material	SCC SCC				Corrosion/Wear C & W				Fatigue Fat.			Reduction in Toughness RiT					
													Aging		*Irradiation*			
Component	[8]*Subdivision*→	IG	IA	TG	LTCP	Wstg	Pit	Wear	FAC	HC	LC/Th	Env	Th	Emb	VS	SR	Th	Fl
BWR Piping	Wrought SS	Y e103	N/A	Y e104	? e013	N	N	N	N	Y e032	Y e105	Y e007	N	N/A	N/A	N/A	N/A	N/A
	SS Welds & Clad	Y e103	N/A	Y e104	? e013	N	N	N	N	Y e032	Y e105	Y e007	N	N/A	N/A	N/A	N/A	N/A
	CASS	Y e119	N/A	N	? e013	N	N	N	N	N	N	? e125	Y e057	N/A	N/A	N/A	N/A	N/A
	C&LAS	? e096	N/A	Y e126	N	N	Y e127	N	Y e128	Y e014	Y e105	Y e007	N	N/A	N/A	N/A	N/A	N/A
	C&LAS Welds	? e096	N/A	Y e126	N	N	Y e127	N	Y e128	Y e014	Y e105	Y e007	N	N/A	N/A	N/A	N/A	N/A
	Ni-base Welds & Clad	Y e129	N/A	Y/ e130	Y e031	N	Y e127	N	N	Y e032	Y e105	Y e131	? e132	N/A	N/A	N/A	N/A	N/A
	Wrought Ni Alloys	Y e129	N/A	Y e130	Y e031	N	N	N	N	Y e032	Y e105	Y e131	N	N/A	N/A	N/A	NA	N/A

[8] IG = intergranular; IA = irradiation assisted; TG = transgranular; LTCP = low temperature crack propagation; PW = primary water; Wstg = wastage; FAC = flow accelerated corrosion; HC = high cycle; LC = low cycle; Th = thermal; Env = environmental; Emb = embrittlement (from fluence); VS = void swelling; SR = stress relaxation; Fl = flux effect

Level 3: Explanatory notes on individual issues referenced within the Level 2 tables

e001	Includes stainless steel and nickel-base penetrations such as stub tubes, BMIs, and pressurizer penetrations.
e002	Possibility of IGSCC or TGSCC at external surfaces following concentration of boric acid (Czech data) requires clarification.
e003	Consensus opinion is that this is not an issue in normal primary water coolant, even if LAS is exposed as a result of a "half-nozzle" repair (see, e.g., NRC SER on CE NPSD 1198-P from Feb. 2002), but final clarification required. Only evidence for this possibility is thought to be CANDU feeder cracking in carbon steel exposed to heavy water coolant, where chemistry is different from PWR.
e004	BAC following leakage into annulus or onto external surfaces. BAC rates provided in Boric Acid Handbook, and additional research is in progress within MRP Alloy 600 ITG to clarify. External corrosion, e.g. Oyster Creek sand bed. Corrosion of low alloy steel exposed to normal primary coolant (e.g., in context of half nozzle repairs) not considered to be an issue.
e005	Not an issue within primary circuit, but unsure of FAC role in D-B RPVH corrosion incident following leakage into annulus. In the case of small leaks, high-velocity steam jets can cause steam cutting. Research is in progress within MRP Alloy 600 ITG to clarify.
e006	Current research in MRP Fatigue ITG primarily addresses piping, not vessels. This is a potential gap for non-piping components. (For wrought stainless steel piping, welds and clad piping also see e021.)
e007	Some environmental fatigue research is ongoing within MRP Fatigue ITG (stainless steel fatigue tests in Germany). Current efforts are aimed primarily at license renewal issues.
e008	Currently being researched by MRP RPV ITG. BWOG data to date suggests that effects up to 200,000h are insignificant.
e009	Being addressed by MRP RPV ITG.
e010	NRC contractor believes there is a flux effect – being addressed by MRP RPV ITG. Spectral effect not considered significant.
e011	Being addressed by MRP RPV ITG. Different weld types must be considered (e.g. SAW vs. SMAW).
e012	Pressurizer heater sheath issue (EDF input sought) – unclear if this is IG, TG or PWSCC. Possible emerging issue (including potential effect of long-term aging; related to e019).

e013	Possibility that LTCP could affect other classes of materials (e.g., SS) is reflected by inclusion of ? in the DM.
e014	Potential mechanism – has not been fully evaluated. Current research in MRP Fatigue ITG primarily addresses piping, not vessels. This is a potential gap for non-piping components.
e015	Not observed in-service to date; however, based on lab data, this is a potential vulnerability. Need to ascertain potential for thermal fatigue of nickel-based welds in CRDM housing.
e016	Only primary-side clad cracking is considered. Davis-Besse had SCC on exposed clad surface after low alloy steel wastage (not considered relevant for further action).
e017	BWR evidence at low fluence (Oyster Creek, etc.) raises the issue of possible IASCC at lower than expected fluences for BWRs and PWRs.
e018	Canopy seals have experienced TGSCC, also CRDM housing welds (Palisades). Incorrect clad welding might lead to TG cracking.
e019	Emerging issue – actively being pursued by EdF. Being considered for wrought material – applicability to clad and weld metal unclear.
e020	Rubbing marks/wear observed during ISI of vessel flange and core support lug areas-not considered major problem.
e021	The widespread adoption of low leakage cores has increased the temperature differential between primary water exiting the core from its center compared to the water from the periphery to possibly around 30°C. The different temperature stream lines are known to persist into the pressure vessel outlet nozzle and into the hot leg piping. While the consequences for thermal fatigue are negligible for components in the intended design condition, this may not be the case in components with poor surface finishes. In addition, such cycling may be important for aiding stress corrosion crack initiation and growth. (Applicable to welds and clad piping also.)
e022	Embrittlement will occur, consequences expected to be low.
e023	Vessel head penetration and bottom-mounted instrumentation (need additional input from A600 ITG). Also applicable to Ni piping. (see Note e133).
e024	Primarily Inconel attachment welds to the vessel.
e025	Vessel head penetration and bottom-mounted instrumentation – pertains primarily to Ni-based alloys.

e026	SG shell has a potential vulnerability in off-chemistry conditions on secondary side, but field experience suggests TGSCC would be more likely than IGSCC (see e136). See also e002 regarding possibility of SCC at external surface in the event of a primary water leakage.
e027	BAC following leakage into annulus or onto external surfaces, c.f. e004. Wastage can occur during secondary side chemical cleaning of steam generator shell.
e028	Restricted to steam generator shell issues on the secondary side and associated with off-chemistry conditions.
e029	
e030	Corrosion-assisted fatigue is a known phenomenon on secondary side (e.g., in the vicinity of girth welds in steam generator shells and in the region of feedwater nozzles) and is not like environmental fatigue described in other areas of this DM. Environmental fatigue research relevant to this specific phenomenon is not ongoing within MRP Fatigue ITG, and is a potential gap.
e031	Applicability not determined – track with work identified for PWR vessel head penetrations and bottom-mounted instrumentation (cf. e023) – pertains primarily to Ni-based welds.
e032	Potential problem – MRP Fatigue ITG to consider.
e033	Other than the observed cracking of baffle/former bolts, other potential issues are believed to be primarily license renewal issues.
e034	Vessel attachment welds including the weld pads are considered part of RPV.
e035	The internals of CE plants are assembled by welding and are not bolted together as in other designs. Thus, the response to fatigue of welded stainless steels experiencing high fluences needs to be considered.
e036	Observed in baffle former bolts.
e037	EDF example: Thermocouple attachment hardware ('staples') cracking is PWSCC as it occurred in nominal PWSCC conditions (but cold work SS). Also an issue for reactor internals in license renewal.
e038	Possibility of wear in PWR internals.
e039	One example – DC Cook baffle-barrel bolt failure. Could be of more generic significance because of stress relaxation issues.

e040	No current activity within MRP Fatigue ITG – this is a potential gap.
e041	A consequence of significantly increased helium gas bubble generation: the helium gas content cannot, therefore, be simply estimated from knowledge of the fast neutron fluence. Helium may play a role in embrittlement, particularly if, for any reason, the bubbles agglomerate preferentially on grain boundaries.
e042	Fast reactor testing is underway and future work is planned through the MRP Reactor Internals ITG.
e043	The thermal neutron flux exhibits a peak just above and below the core, and the thermal to fast neutron ratio increases with radial distance from the periphery of the core. A consequence is significantly increased helium gas bubble generation; the helium gas content cannot, therefore, be simply estimated from a knowledge of the fast neutron fluence. Helium may play a role in embrittlement particularly if, for any reason, the bubbles agglomerate preferentially on grain boundaries.
e044	Relevance of fast reactor data to PWR conditions needs to be evaluated. Additional expert comment: Intergranular cracking has been observed by several groups while fatigue pre-cracking fracture mechanics specimens of highly irradiated stainless steels (up to ~30dpa, in general from the field) in air at room temperature. This unusual observation suggests an involvement of hydrogen embrittlement, the hydrogen being trapped in the metal during prior exposure to PWR coolant.
e045	Limited data – additional work needed.
e046	MRP Reactor Internals ITG is looking at synergistic effect of thermal aging and irradiation embrittlement.
e047	Not an issue because CASS is outside high-fluence area.
e048	Primary relevance is to CE plants where welds are used instead of baffle bolts. See also e041.
e049	Relevance of fast reactor data to PWR conditions needs to be evaluated.
e050	Very limited data on IASCC of Ni-based material – mitigative potential limited.
e051	Limited data on X-750, cf e023.
e052	X-750 has cracked in-service.
e053	No specific data for LWR environments.

e054	No specific data for LWR environments (fuel alignment pin).
e055	Potential emerging issue for piping (also Palisades CRDM housings) **See Note e096.** Potential for ODSCC due to contaminants.
e056	MRP Fatigue ITG addressing surface finish effects may be important. However, research is very limited.
e057	License renewal issue for cast materials and potentially Ni-based welds – not resolved w/NRC.
e058	Is only thought to be an issue for secondary plant piping (e.g., steam and feedwater piping).
e059	Pertains presently to Ni-based alloys. Various mitigation methods under active consideration.
e060	Applies to secondary side for both low- and high-temp annealed tubing.
e061	Applies to secondary side, associated with Pb contamination.
e062	Extensive experience – mitigation measures provide some improvement.
e063	Wastage was experienced by many early units that operated on phosphate water chemistry. Significant wastage has also been experienced by several plants using AVT water chemistry, such s Millstone 2 at the hot leg TTS, Point Beach 2 at the cold leg TTS, and Prairie Island at cold leg TSPs (called cold leg fretting). Use of AVT with tight water chemistry control has kept this problem insignificant in modern PWRs.
e064	Limited experience.
e065	Extensive experience, applies to secondary side – mitigation measures provide some improvement.
e066	Limited experience with OTSGs.
e067	Large concentrations of reduced sulfur species have been shown in autoclave and model boiler tests to cause IGA/SCC of 600MA similar to that observed in plants. Reduced sulfur in the form of nickel sulfides is routinely found on tube and crack surfaces during pulled tube examinations. Plants typically have 1 to 2 ppb sulfate in their blowdown. Tests indicate that sulfate in the bulk water can be reduced by hydrazine and that this can result in reduced sulfur species in the crevices.
e068	Some experience on secondary side.

e069	Some limited mitigation experience (kiss-rolled expansions), Vogtle plant experience.
e070	As discussed in e063, SGs with 600MA tubing have experienced significant wastage. Since 600TT has the same chemical composition as 600MA, it is expected that it would be equally susceptible to this mode of attack as 600MA if aggressive chemistry conditions should occur. However, use of AVT with tight water chemistry control and the improved design features of SGs with 600TT tubes (e.g., smaller crevices) are expected to keep this problem at an insignificant level in 600TT SGs.
e071	Korean experience (secondary side).
e072	Extensive experience, support modifications mitigate.
e073	Lab data (ANL) indicates susceptibility.
e074	Potential secondary side vulnerability.
e075	Potential vulnerability (Bettis results).
e076	Industry believes not an issue, under discussion with NRC – Navy Program (?).
e077	Tests in acidic sulfate-chloride environments indicate that 690TT is as susceptible to wastage, or a little more susceptible, as 600TT.[9] Experience has shown that the chemical composition of 600 makes it susceptible to wastage in aggressive chemistry conditions. However, tight AVT water chemistry has been successful at keeping wastage at an insignificant level in modern plants with 600MA and 600TT tubes. For this reason, while 690TT has some susceptibility to wastage, it is expected that tight water chemistry control and the improved design features of SGs with 690TT tubes (e.g., smaller crevices) will keep this problem at an insignificant level in 690TT SGs.
e078	Limited experience in field.
e079	Unclear if threshold iron content adequately defined.
e080	Potential issue – associated with Pb contamination.
e081	Ni-based, extensive field experience (applies to seal welds).

[9] W. H. Cullen, "Review of IGA, IGSCC, and Wastage of Alloys 600 and 690 in High-Temperature Acidified Solutions," *Control of Corrosion on the Secondary Side of Steam Generators*, p273, NACE, 1996.

e082	Extensive field experience with PWSCC, also thiosulfate contamination incident at TMI-1.
e083	Limited experience, mitigation provides some improvement.
e084	Potential problem from loose-parts.
e085	Some limited erosion experience.
e086	Extensive field experience.
e087	Several plants have experienced severe wear of welds at the tube to clad weld at the primary face of the tube sheet as the result of loose part impact. This is a recurring problem. Recent incidents have been the result of loose parts from broken control rod guide tube alignment pins. This problem is mitigated by actions to prevent the generation of loose parts (such use of more resistant materials and designs for the alignment pins) and by use of loose part monitors at the SG tube sheet.
e088	Applicable to replacement OTSG designs.
e089	This row is limited to seal-welds at primary TS interface with tube bundle.
e090	Top of tube sheet (TTS) crevices between the tubes and the tube sheet are potential locations for denting to occur, even in new steam generators. In the long term, this could increase the probability of SCC at this location. Typical crevices are about 1/8 in. deep and have a 10 mil radial gap. The crevice is generally made more severe by development of a collar of deposits above the crevice, resulting in an effective depth in the neighborhood of ¼ inch. Several plants with full depth expansion have experienced significant denting and resulting SCC at this location
e091	TSP denting and cracking – extensive experience, mitigation available; also feed-ring cracking. Potential also for problems originating from lay-up conditions.
e092	Mechanism originally identified in heavily fouled CE steam generators at 4 units. The egg crates adjacent to the support wedges experienced mild to severe corrosion. Some egg crates were actually missing.
e093	Need to evaluate actual material category used for certain components (e.g., tie rods).
e094	Plugs have failed and leaked – consequences of this failure have resulted in wear damage to other tubes.

e095	Some EDF experience. SG partition plates are not inspected in the U.S. License renewal issue?
e096	Although SCC in (oxygenated) BWR water is generally considered to be transgranular in nature, intergranular facets may appear under certain circumstances (e.g., as a result of dynamic strain aging). See also CANDU experience referred to in e003.
e097	Limited data available within CIR Program (from VGB contribution) do not show an effect, but may be inconclusive.
e098	Addressed in BWRVIP-60. Additional information needed on intermediate temperature, ripple loading and chloride susceptibility (Seifert's SCC data). Full mitigation unclear.
e099	Experience to date indicates insignificant degradation; issue was examined in the context of license renewal.
e100	Resolved via NUREG-0619.
e101	Seifert's fatigue data being evaluated by BWRVIP & MRP Fatigue ITG.
e102	PWR analysis should be bounding for BWR, see e008.
e103	High local strain at weld fusion line in stainless steels, including unsensitized and 316NG leads to intergranular crack propagation. Lab experience indicates cracking not fully mitigated by hydrogen. Applicability to U.S. plants not known. Need to understand Japanese and German experience.
e104	Occurred for example in Japanese plants (316L) – associated with a cold-worked surface, lab evidence suggests HWC mitigation not effective. The transition to IGSCC is not understood.
e105	BWR operating conditions generally do not promote thermal fatigue, but MRP Fatigue ITG to address.
e106	Cracking observed in early BWRs, addressed in BWRVIP-05, HWC would mitigate in some areas.
e107	Potentially exacerbates IGSCC for clad at beltline of vessel.
e108	Incorrect clad welding has led to TG cracking thru H-mechanism.
e109	Applies only to Reactor Pressure Vessel.

e110	Cracking to date limited to H-9 welds (Tsuruga 1 and NMP 1) – all 182 weld metal potentially susceptible, HWC can mitigate.
e111	This is an impurity problem, good chemistry control mitigates.
e112	White deposits were observed on the RPV stainless steel clad walls, steam dryer and steam separator in the RPV steam region of a BWR-6. In the area of the RPV steam line nozzles, this deposit appears heavy and stucco like in appearance. The heaviest deposit was observed in the areas of highest flow and only on stainless steel; no deposit was observed on carbon steel. Factors that are implicated in this deposition are low iron and application of NobleChem before instituting H2 injection. Although the material was very hard and difficult to remove, sample analysis on a very small volume of deposits indicated that the deposit was an iron chromium oxide with traces of hematite, also including smaller amounts of silica, calcium, nickel, aluminum, zirconium, and manganese.
e113	Experience limited to jet pumps and steam dryer.
e114	High Cycle Fatigue has recently occurred on steam dryers in several older BWRS after power uprates were implemented. GE, in coordination with the BWROG and BWRVIP, are currently working to understand the source and magnitude of the acoustic loads believed to be responsible for the steam dryer damage.
e115	Potential issue for core-plate bolts.
e116	Significant impact on the ability to perform weld repairs because of He.
e117	Relevance of fast reactor data to BWR conditions needs to be evaluated.
e118	Limited data – should be worked in conjunction with studies of CASS.
e119	Limited European experience – additional information to be developed.
e120	All 182 weld materials considered susceptible. Japanese experience indicates stainless weld metals have some level of susceptibility.
e121	Includes experience to date relating to Jet Pump riser brace. Orange shading because of insufficient understanding of how to deal with SD welds.
e122	Limited to JP beam and shroud repair hardware.
e123	Data re. shroud repair hardware.

e124	Alloy 600 access hole covers – can be mitigated. X-750 JP beams – no known mitigation.
e125	License renewal issue – BWRVIP needs to review and resolve.
e126	Some European experience, HWC will mitigate.
e127	Service water and lay-up issues.
e128	Addressed by BOP corrosion, except for feedwater piping.
e129	Primarily Ni-based weld material has cracked – HWC will mitigate.
e130	Has not been observed-HWC should mitigate.
e131	Weld material fatigue issues need to be addressed w/piping – cf. e007.
e132	Non-ferritic welds may be subject – need additional study.
e133	Applies to B&W design flow-meters in hot leg piping.
e134	No known failure due to fatigue loading for the material identified; however, there may be some plants with wrought safe-end components on nozzles subject to thermal fatigue.
e135	Unless otherwise noted, potential damage mechanisms for BWR internals have been or are being addressed by BWRVIP.
e136	Secondary side steam generator shell cracking has been observed and linked to off-chemistry conditions (e.g. Indian Point). See also e002 regarding possibility of SCC at external surface in the event of a primary water leakage.

Mechanism Information Summaries

CONTENTS

Mechanism Information Summaries

1. Introduction

1.1 Objective

The objective of this document is to provide summaries of relevant information regarding materials degradation mechanisms for use in conjunction with the Degradation Matrix being developed by EPRI. The degradation mechanisms covered in this document are those known (or believed to be) pertinent to the materials used in the reactor coolant system and internals of current light water reactors in the U.S.

1.2 Acronyms Used in This Document

Acronym	Meaning
ANL	Argonne National Laboratory
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing Materials
BWR	Boiling water reactor
BWRVIP	Boiling Water Reactor Vessel Internals Program
CASS	Cast austenitic stainless steel
CGR	Crack growth rate
EAC	Environmentally Assisted Cracking
ECP	Electrochemical potential
EPR	Electrochemical potentiokinetic reactivation
FAC	Flow-accelerated corrosion
HAZ	Heat affected zone
HWC	Hydrogen water chemistry
IASCC	Irradiation assisted stress corrosion cracking
IHSI	Induction heating stress improvement
LAS	Low-alloy steel
LTCP	Low-temperature crack propagation
LWR	Light water reactor
MSIP	Mechanical stress improvement process
NMCA	Noble metal chemical addition
NWC	Normal water chemistry
PWR	Pressurized water reactor
PWSCC	Primary water stress corrosion cracking

RT	Radiographic test
RT_{NDT}	Reference temperature/nil ductility temperature
SCC	Stress corrosion cracking
USE	Upper shelf energy
UT	Ultrasonic test
VCD	Vacuum carbon deoxidation

2. Stress Corrosion Cracking (SCC)[1]

This section addresses materials degradation due to stress corrosion cracking (SCC) which, in one form or another, is the key aging degradation mechanism for a number of major components throughout the primary pressure boundary of both PWRs and BWRs and is also the primary concern for vessel internals.

2.1 Equipment

Major components potentially affected by SCC include:

1. Pressure vessels (RPV, pressurizer and SG shell, including cladding, attachment welds and vessel nozzles)
2. RPV internals
3. Piping
4. SG tubing

2.2 Materials

A large variety of materials are used in the components mentioned above, including:

1. Pressure vessels – carbon steel (CS), low-ally steel (LAS), stainless steel (SS) cladding, nickel-based alloys and various weld metals, depending upon the parent material used.
2. Internals – SS, cast austenitic stainless steel (CASS), nickel-based alloys, and various weld metals, depending upon the parent material used.
3. Piping – SS, CASS, CS, LAS, and various weld metals, depending upon the parent material used.
4. SG tubing – nickel-based alloys and associated seal-weld metals.

All of the above materials are potentially susceptible to one or more of the forms of SCC degradation described below.

2.3 SCC Degradation Mechanisms and Mitigation Options

Fundamentally, SCC involves a complex interaction between mechanical loading of a susceptible material in an environment capable of causing cracking. Each of these three areas involves many subsets of critical parameters.

The scheme chosen to differentiate between key SCC mechanisms in the present context of the materials degradation matrix is as follows, although it should be noted that there are many areas of overlap.

[1] Crevice corrosion is inherently part of SCC and will be appropriately addressed in an expanded treatment of the degradation mechanism.

Mechanism Information Summaries

2.3.1 Intergranular SCC (IGSCC) Except for Primary Water SCC (PWSCC)

IGSCC is associated particularly with the cracking of austenitic stainless steels that has been experienced in high-purity water in BWR piping and internals, although it is also a relevant mechanism for SS vessel cladding (in both PWR and BWR), PWR internals and SG tubing on the secondary side. IGSCC has been extensively researched and is considered to proceed primarily by a slip oxidation (dissolution) mechanism, which has been successfully modeled in terms of key parameters such as crack tip strain rate (from applied/residual stresses), corrosion potential and conductivity (from surface chemistry/bulk water composition) and material composition/microstructure (e.g., degree of sensitization). Key areas where further work is required include the effects of cold work (including locally in weld heat-affected zones) and the behavior of CASS and nickel-based weld metals, as well as the influence of specific, deleterious coolant impurities (e.g., lead, residues from ion-exchanger resins). IGSCC of CS and LAS does not normally occur in LWR media, but limited cracking of this type is known to have been observed in CANDU reactors, and it should also be considered a possible degradation mechanism in concentrated boric acid environments, such as might form on external surfaces following leakage of PWR primary coolant.

Mitigation of IGSCC is focused primarily on improved coolant chemistry (e.g., hydrogen water chemistry (HWC), impurity reduction and zinc addition in BWRs, optimized secondary-side chemistry in PWRs), sometimes together with component surface modification (e.g., Noble Metal Chemical Addition (NMCA) or zirconia coating in BWRs). However, stress reduction has also been used extensively (e.g., weld overlays for piping and clamps for internals in BWRs, improved tube support plate structures in PWR SGs, etc.). The primary emphasis has been on avoiding crack growth (or at least reducing rates), since minor intergranular attack (IGA) of austenitic alloys is often present from fabrication and/or cannot be prevented in operation.

2.3.2 Irradiation Assisted SCC (IASCC)

The SCC behavior of irradiated SS is a natural extension of IGSCC of un-irradiated SS, but the critical fluence level above which irradiation effects begin to dominate material behavior is complex. A lower value of ~5 x 10^{20} n/cm^2 is often quoted for BWR internals, with saturation of the effects beginning at around 3 x 10^{21} n/cm^2, i.e., shortly before the expected end of life (EOL) fluence of ~8 x 10^{21} n/cm^2 . In contrast, IASCC in PWRs has only been observed (e.g., in baffle/former bolts) to start after reaching a fluence of ~2 x 10^{21} n/cm^2, and little information is available about expected behavior near the much higher EOL fluence values typical of PWRs.

The fundamental mechanism of IASCC in PWR primary water is currently unclear, with no evidence that locally oxidizing conditions, grain boundary segregation, helium formation, or hydrogen embrittlement play a major role, although high strength from irradiation hardening does appear to be important (possibly analogous to the effects of cold work in SCC without irradiation). Mitigation measures are not yet available.

Apart from its concurrent role in reducing fracture toughness, irradiation in BWRs is best viewed as a SCC accelerant through its effect on grain boundary segregation (including material sensitization), hardening, differential swelling, and elevation of corrosion potential. It can sometimes have a beneficial effect via creep relaxation. Mitigation of IASCC in BWRs is focused primarily on reductions in corrosion potential through the use of HWC/NMCA, i.e., an extension of the approach already taken for IGSCC, but the benefits are less well quantified.

Research on IASCC has been extensive in recent years, much of it organized through the international cooperative CIR Program, managed by EPRI. However, more work remains to be done, both in terms of understanding mechanisms and application of the data generated to actual, long-term LWR operation.

2.3.3 Transgranular SCC (TGSCC)

TGSCC is a classic form of material degradation, usually associated with the ingress of aggressive impurities (e.g., chlorides) and oxidants into reactor systems, where cracks (e.g., in SS) often then originate from corrosion pits. Such incidents are avoided primarily by attention to water chemistry and cleanliness of external surfaces, although system dead legs may pose challenges here (e.g., for CRDMs and canopy seals in PWRs). However, there is increasing evidence to suggest that TGSCC can also occur in austenitic materials in nominal LWR water chemistry as a result of excessive cold work.

Although CS and LAS are regarded as highly resistant to SCC under LWR conditions, limited TGSCC has been observed, e.g., in SG shells exposed to faulted secondary water, and in BWR components subjected to high, local loads while operating with normal (oxygenated) water chemistry. Recent research suggests that occasional susceptibility may also be related to changes in the deformation behavior of particular steels associated with the dynamic strain aging that can occur at intermediate operating temperatures.

Mitigation of all forms of TGSCC can usually be achieved by the avoidance of high corrosion potentials, often in conjunction with the elimination of detrimental impurities in the operating medium. In specific cases, other options, e.g., involving surface treatment, stress reduction, etc., may also be available.

2.3.4 Low-Temperature Crack Propagation (LTCP)

LTCP refers both to high sub-critical crack growth rates (i.e., SCC, most likely from hydrogen-assisted cracking) and to reduced fracture toughness. While the largest concerns are for higher-strength Ni alloys (e.g., Alloy X750 and Alloy 182/82 weld metals), there are known concerns and/or reasonable bases for concerns for base metals, particularly (but not only) if the yield strength is elevated (e.g., from cold work or irradiation).

Initial studies by Grove and Petznold in the 1980s showed very rapid crack propagation in the temperature range 70-140ºC in moderate to high-strength Ni base alloys once IG SCC cracks had formed in high-temperature water. The highest rates were observed in Alloy X750, although large effects were also observed for Alloy 182 and 82 weld metals and other Ni base alloys (e.g., aged Alloy 625, Alloy 718, and Alloy 690). The observations occurred in constant displacement (wedge/bolt loaded) CT specimens, in actively loaded CT specimens, and also in specimens exposed only to gaseous hydrogen in this temperature regime (leading to the reasonable conclusion that it's a hydrogen related phenomenon). More recently, Mills and others have observed reduction in fracture toughness (e.g., in J-R tests) in the same temperature regime.

The possibility that LTCP could affect other classes of materials (e.g., SS) has not yet been explored and this uncertainty is reflected by the inclusion of question marks in the pertinent cells in the DM table.

2.3.5 Primary Water SCC (PWSCC)

- As used in the degradation matrix, PWSCC refers to intergranular cracking of any material, but particularly of Ni- base alloys such as Alloy 600 and its weld metals, in PWR primary coolant (i.e., containing lithium, boric acid and hydrogen) of correct nominal chemistry.
- Over the last 30 years, intergranular stress corrosion cracking in PWR primary water (PWSCC) has been observed in numerous components made of Alloy 600 and its associated weld metals (Alloys 82/182), sometimes after relatively long incubation times. In stark contrast to IGSCC of Ni-base alloys in other media (e.g., on the PWR secondary side) and to IASCC of these and other austenitic alloys, sensitization of the material through intergranular precipitation of chromium-containing carbides is <u>beneficial</u> to the PWSCC resistance of Alloy 600, which justifies its consideration in a separate category. However, large variations exist in the susceptibility of individual heats of material, even of nominally similar composition and thermomechanical processing history, so that prediction of service behavior is difficult. Cold work is highly detrimental.
- Cracking, which can also occur in pure hydrogenated water or steam, is highly temperature-dependent and appears to be associated with environmental conditions under which the surface films are in the transition region of Ni/NiO stability. Despite intensive research, there is no general agreement on the mechanism of PWSCC. Candidate theories include hydrogen-assisted cracking, slip oxidation, thermally activated dislocation creep and internal oxidation. The latter has a particular attraction, since it could explain the very long times (>100,000 hours) sometimes necessary for cracking to initiate, even under conditions where subsequent crack propagation is relatively rapid. PWSCC of weld metals (and its possible interaction with fabrication defects such as hot cracking) is currently a high-profile topic that has been insufficiently studied and is not well understood.
- To date, mitigation of PWSCC has usually involved repair and replacement actions using more resistant materials (such as Alloy 690). However, increased attention is now being paid to possible mitigation measures involving surface treatment (e.g. water-jet peening), chemistry optimization (e.g., adjustment of hydrogen levels and/or addition of potentially inhibiting species such as zinc), and various mechanical options to achieve a reduction in tensile stress levels.

2.4 Areas for Further Research

- Given the extreme variety and complexity of SCC degradation in LWRs, and the fact that individual aspects of many of the abovementioned problems are already being addressed in the separate issue programs such as BWRVIP, MRP and SGMP, it would make sense to focus additional work within the Corrosion Research Program into four main areas:
  - Aspects where a commonality of response has been observed throughout the separate SCC categories discussed above. One obvious example of this is the effect of cold work (particularly as it may occur in highly localized form in the HAZ of welds). Others

would include the key transition between short- and long-crack behavior relevant to concepts of crack initiation versus growth, as well as the dominant influence of corrosion potential in many systems.

- Situations where recent, major advances in the techniques used to study SCC offer the possibility of resolving key questions (such as the mechanism of PWSCC and IASCC) that have remained unanswered to date. Examples here include the use of high-resolution ATEM studies of stress corrosion crack tips in both specimens and components (pioneered by Bruemmer & Thomas at PNNL) and in-situ Raman spectroscopy of surface films during autoclave studies.

- Work with alloys where aspects of physical metallurgy (such as dynamic strain aging, ductility-dip cracking, etc.) have recently been recognized as having a possible link with their SCC behavior in LWR environments.

- Renewed efforts to develop monitoring techniques so as to provide early warning of the likely occurrence of SCC in LWR systems. Initial efforts here (e.g., using acoustic emission or electrochemical noise) did not prove easy to transfer into practical devices, but the potential rewards for success here justify renewed attention to this topic.

- A number of other R&D topics on which further work is needed were identified in the course of a review of this version of the Degradation Matrix by teams of subject-matter experts. These topics will be included in the next revision of the Degradation Matrix supporting material.

2.5 References

Key references providing further details on materials degradation due to stress corrosion cracking will be added to this section during the next revision of the Degradation Matrix.

3. Corrosion & Wear (C&W)

This section addresses degradation mechanisms which result in loss of material including: general corrosion; galvanic, crevice, and pitting corrosion; erosion and erosion-corrosion; and mechanical wear. All these mechanisms are age-related effects that must be managed throughout the current license term, continuing through any license renewal term for primary pressure boundary components of both PWRs and BWRs.

3.1 Equipment

In general, the potential for loss of material is a consideration for all major components of the PWR and BWR primary pressure boundary as well as many internal components. However, as discussed below, many components are either not affected or not affected significantly, because of environment (e.g., temperature, coolant chemistry) or material selection. Loss of material, through a variety of degradation mechanisms – including general corrosion; galvanic, crevice, and pitting corrosion; erosion and erosion-corrosion; and wear, is an age-related effect.

3.2 Materials

A large variety of materials are used in the fabrication of PWR and BWR primary pressure boundary components, including:

1. Pressure vessels – carbon steel (CS), low-alloy steel (LAS), stainless steel (SS) cladding, nickel-based alloys and associated weld metals.
2. Internals – SS, cast austenitic stainless steel (CASS), nickel-based alloys, and associated weld metals.
3. Piping – SS, CASS, CS, LAS, and associated weld metals.
4. Steam Generator tubing – nickel-based alloys.

All of the above materials are potentially susceptible to one or more mechanisms of loss of material, depending upon the combinations of material and service conditions.

3.3 Loss of Material Degradation Mechanisms & Mitigation Options

3.3.1 General Corrosion

General corrosion is the loss of material, generally measured as a ***uniform*** rate of loss of surface material, caused by a chemical or electrochemical reaction between the surface of a metal component and an aggressive environment in contact with that surface. General corrosion is characterized by ***uniform*** surface loss through material dissolution, often accompanied by the presence of corrosion products in the coolant. General corrosion requires both a susceptible material and an aggressive environment.

General corrosion is an electrolytic reaction and, regardless of the fluid or gas in contact with the metal surface, depends on the presence of oxygen and moisture. When oxygen is controlled, general corrosion is not a consideration. Therefore, the internal surfaces of primary pressure boundary components are not susceptible. The external surfaces of primary pressure boundary components could be subject to a general corrosion environment, provided that oxygen and moisture are present.

Wrought austenitic stainless steel, CASS, and nickel-based alloys are not susceptible to general corrosion when exposed to water environments with controlled chemistries (e.g., pH), and are only slightly susceptible under normal external environments in nuclear power plants. However, CS and LAS are both potentially susceptible to general corrosion.

The effects of general corrosion are mitigated by the design basis for PWR and BWR primary pressure boundary CS and LAS components, which includes a corrosion allowance to account for the loss of material over the component service life. For example, BWR CS piping design involves the specification of corrosion allowances for wall thickness. Actual corrosion rates have been found to be significantly below the design-basis allowable values. The specified corrosion allowance is conservatively linear based upon the design life. The design specification[i] [1] provides a design general corrosion allowance of 120 mils for the main steam system. The actual general corrosion rate for CS piping in a steam environment is less than 0.16 mils per year[ii] [2]. This general corrosion rate is based upon relevant available information obtained from laboratory and in-reactor investigations and the open literature. For a 40-year life, the general corrosion total would be less than 7 mils. Extrapolation for an additional license term shows that generous corrosion margins would still exist even after an additional 20 years of service life.

Similarly, the corrosion allowance for SS piping operating in the 500-600°F range is 2.4 mils. The actual general corrosion rate for stainless steel in this temperature range is 0.01mils/year of service life. These rates are conservative since corrosion rates generally decrease with time. CASS (e.g., recirculation pump bowl, valve bodies) components also have a low susceptibility to general corrosion.

3.3.2 Galvanic Corrosion

Galvanic corrosion is the loss of material, generally measured as a ***local*** rate of loss of surface material, caused when two materials with substantially different electrochemical potentials are in contact in the presence of a corrosive environment. The effects of galvanic corrosion are typically precluded through design – separation of materials with different electrochemical potentials to prevent electrolytic connection. CS and LAS have substantially lower electrochemical potentials relative to SS, and would be preferentially attacked in a galvanic couple. The severity of galvanic corrosion is a function of the type and amount of moisture, and will not occur when the metal surfaces are completely dry (i.e., no electrolyte to carry the galvanic current).

Galvanic corrosion is mitigated through material selection or through control of the corrosive environment (e.g., elimination of the electrolyte). The internal surfaces of primary pressure boundary components are not susceptible, because water chemistry controls limit the species that contribute to the electrolytic connection. The external surfaces are potentially susceptible; however, materials with substantially different electrochemical potentials are generally physically separated, so that the electrolyte path is interrupted.

3.3.3 Crevice and Pitting Corrosion

Crevice and pitting corrosion are two forms of loss of material, generally measured as a ***local*** rate of loss of surface material, that are caused by corrosive conditions within the crevice or pit. Crevice corrosion is observed in joints and connections, such as gaskets, lap joints, and under bolt heads, where the aggressive chemical species concentrate in the crevice itself, promoting auto-catalytic behavior between the crevice and the surrounding metal. Pitting corrosion is similar, except that the geometric feature that leads to auto-catalytic behavior is self-created after the long incubation period required for the initial pits to form.

Austenitic SSs resist corrosive attack in a PWR environment by quickly oxidizing to form a protective film. However, even for the internal surfaces of PWR reactor coolant system components, creviced locations have the potential to cause localized corrosion, even for film-forming materials. For example, SS piping and Alloy 600 safe ends have exhibited damage from crevice corrosion[iii] [3]. However, hydrogen plays an important role in the control of crevice corrosion by minimizing the adverse effects of oxygen. The hydrogen overpressure in a PWR reactor coolant system provides adequate protection against crevice corrosion for the internal surfaces of reactor coolant system components.

Crevice corrosion is also mitigated by designs that avoid most localized crevice geometries, or by controlling harmful species that can concentrate and attack the material surfaces in the crevices that remain. Crevices generally contain stagnant fluid that permits the concentration of contaminants even under system fluid flow conditions. Halides and sulfates are two of the most common aggressive species; however, dissolved oxygen is often sufficient in itself to promote crevice corrosion. Provided that the oxygen content is well controlled (e.g., PWR water chemistry controls or BWR hydrogen water chemistry), crevice corrosion is highly unlikely. Normal BWR water chemistry and a creviced geometry are sufficient to cause crevice corrosion.

All nuclear power plant materials are susceptible to pitting corrosion in the presence of a sufficiently aggressive environment to cause the initial formation of pits whose growth can then be auto-catalytically driven. Pitting requires either low flow or stagnant flow, in order to sustain the corrosion reactions and to provide for the concentration of contaminants. Therefore, maintaining an adequate flow rate will help to mitigate pitting corrosion. In addition, water chemistry controls on the initiating chemical species – which include halide and sulfate ions – will generally increase the initial pit incubation time sufficiently to avoid this phenomenon.

3.3.4 Boric Acid Wastage

Boric acid wastage is a form of loss of material that represents an example of very aggressive pitting corrosion. In this case, borated water may leak from the PWR reactor coolant system onto the external surface of a metallic component, such as a reactor coolant pump. This leakage of PWR primary coolant through a bolted closure, and the subsequent evaporation and re-wetting cycles, can lead to the presence of a concentrated boric acid slurry on the external surfaces of adjacent reactor coolant system components[iv,v,vi] These alternate wetting and drying cycles produce a low pH that, in combination with an air atmosphere, can cause very high corrosion rates (approximately 1 inch per year). The corrosion rate is greatest at temperatures between 200 and 350°F, but potentially significant corrosion rates are possible even at higher temperatures. Evaporative cooling of exposed components, associated with the flashing of leaking coolant into steam, can increase the corrosion rate of component external surfaces that are normally at temperatures where boric acid corrosion rates would be much lower.

Loss of material from boric acid wastage is mitigated by control of borated coolant leakage, through in-service inspection and corrective action programs.

3.3.5 Microbiologically Influenced Corrosion

Microbiologically influenced corrosion (MIC) is the loss of material, generally measured as a local rate of loss of surface material, that is caused by corrosive attack accelerated by micro-biological activity. MIC usually occurs at temperatures between 50 and 120ºF; however, the microbes are known to survive over a temperature range from 15 to 210ºF. The organisms have been observed in media with pH ranging from 0 to 10.5, and under pressures up to 15,000 psi. The features of MIC are similar to pitting corrosion.

Water chemistry controls mitigate MIC. In particular, sulfates in the primary coolant should be controlled to <100 ppb. However, some treated water systems have shown evidence of MIC, with a common example being torus damage for BWR Mark I containments. Other treated water systems, such as PWR borated emergency core cooling systems, have operated for many years with no evidence of MIC. Typically, treated water systems are low in the nutrients required to sustain microorganism activity. However, stagnant or low flow regions can allow corrosion products and contaminants to accumulate and settle.

3.3.6 Erosion and Erosion-Corrosion

Erosion is the loss of material, generally measured as a reduction in component wall thickness, caused by the action of fluids or fluid-suspended particulate matter on a metal surface. Erosion is a function of the fluid velocity and any fluid turbulence – generally a function of component geometry and local flow conditions. Impingement and cavitation are forms of erosion. Erosion-corrosion occurs when erosion is enhanced by the corrosive nature of the fluid.

In both PWR and BWR environments, CS and LAS are considered potentially susceptible to erosion-corrosion. SS, CASS, and nickel-based alloys are considered resistant to erosion-corrosion.

Erosion-corrosion in CS and LAS is a complex phenomenon that involves the electrochemical aspects of general corrosion, mass transfer and momentum transfer. Studies[vii,viii] on single-phase erosion have shown that the erosion-corrosion of CS under single-phase flow conditions depends on water chemistry, temperature, flow path, material composition and geometry. For wet steam (two-phase flow), the percentage of moisture provides an additional functional dependency. Erosion-corrosion occurs at temperatures ranging from 50 to 200°C, with a maximum loss of material at approximately 130°C.

Single-phase erosion-corrosion is mitigated by increasing the pH and the dissolved oxygen content in the coolant. Reactor feedwater dissolved oxygen content is regulated to the range of 20 to 50 ppb during power operation. Other primary coolant pressure boundary systems are maintained in the range of 20 to 200 ppb. For oxygen concentrations below 20 ppb, data indicate an increase in the erosion-corrosion rate for CS.[ix]

3.3.7 Wear

Wear is the loss of material, generally measured as the rate of removal of surface material, caused by the relative motion between adjacent metal surfaces or by the action of hard, abrasive particles in contact with a metal surface. Mechanical wear is observed in bolted or clamped joints where relative motion is not intended, but which occurs due to the reduction or loss of pre-load. Flow-induced vibration is also known to be a cause of wear through intermittent contact of adjacent metal surfaces.

A limited number of the PWR and BWR primary pressure boundary components are subject to relative motion and none are subject to the action of hard, abrasive particles under normal conditions of operation. However, safety and relief valve seats and disks are subject to intermittent relative motion due to operation and testing, and reactor coolant pump and safety/relief valve closure parts, such as the cover and bonnet flanges, the casing and body flanges, and the closure bolting, are subject to some degree of relative motion, especially when pre-load is lost or during infrequent disassembly and reassembly operations. Therefore, mechanical wear could be an issue for some primary pressure boundary components, such as reactor coolant pump and safety/relief valve closure elements and the safety/relief valve seats and disks for PWRs, and for BWR primary coolant pressure boundary recirculation pump seal flange and valve closure flanges.

3.4 Areas for Further Research

The amount of research into the various material degradation mechanisms that lead to loss of material is relatively minimal. The two areas where active research has been underway and continues are: (1) boric acid wastage, where the phenomenon was studied in order to support regulatory requirements; and (2) erosion-corrosion, where the industry studied both single-phase and two-phase flow manifestations of the phenomenon, in order to develop guidelines for utility inspection of CS piping.

A number of additional R&D topics on which further work is needed were identified in the course of a review of this version of the Degradation matrix by teams of subject-matter experts. These topics will be included in the next revision of the Degradation Matrix supporting material.

3.5 References

[i] GE-NE Document 22A1495, Rev. 3, “Pressure Integrity of Piping and Equipment Pressure Parts,” February, 1974.

[ii] GE-NE F&PMT Transmittal No. 89-178-006, Rev. 1, dated September, 1989.

[iii] “Component Life Estimation: LWR Structural Materials Degradation Mechanisms,” EPRI Report NP-5461, September 1987.

[iv] Howells, E and L. H. Vaughn, “Corrosion of Reactor Materials in Boric Acid Solutions,” RDE-1086, Babcock & Wilcox, Alliance, Ohio, August 1960.

[v] IE Information Notice 8b-108, “Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion,” U.5. Nuclear Regulatory Commission, December 29, 1986.

[vi] IE Information Notice 86-108, Supplement 1, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Add Corrosion," U.5. Nuclear Regulatory Commission, April 20, 1987.
[vii] G. Cragnolino, C. Czajkowski, W.J. Shack, "Review of Erosion-Corrosion in Single-Phase Flows," NUREG/CR-5156 ANL-88-25, April, 1988.
[viii] B.M. Gordon, "Corrosion Issues in the BWR and their Mitigation for Plant Life Extension," PMWG-GE-529, Modification 0, issued January 3,1989.
[ix] "Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants," NUREG-1344, April,1989.

4. Fatigue

This section addresses materials degradation due to fatigue, an aging degradation mechanism that can affect a number of major components throughout the primary pressure boundary of both PWRs and BWRs.

4.1 Equipment

Major components potentially affected by fatigue include:

- Reactor coolant system (RCS) piping, fittings and valves and RCS-attached branch line piping/fittings/valves
- Reactor pressure vessel (RPV)
- Pressurizer
- Steam generator shell, tubes, and internals
- RPV internals components
- PWR reactor coolant pumps and BWR recirculation pumps

4.2 Materials

A large variety of materials are used in the above components, including:

1. RCS piping and fittings – carbon steel (CS), low-alloy steel (LAS), stainless steel (SS), and cast austenitic stainless steel (CASS).
2. Reactor pressure vessels (applicable to both PWR and BWRs) – low-alloy steel (LAS), wrought stainless steel (SS) cladding, wrought nickel-based penetrations and various weld materials depending on the parent material used.
3. Pressurizer – same as reactor pressure vessels.
4. Steam generator shell, tubes, and internals – same as reactor pressure vessels plus (steam generator tubes).
5. RPV internals (applicable to both PWR and BWRs) – SS, cast austenitic stainless steel (CASS), Inconel, and various weld materials depending on the parent material used.
6. Pumps – SS and CASS for pressure boundary materials; various high-alloy steels for bolting and austenitic or martensitic SS for pump shafts and other internal components.

4.3 Fatigue Degradation Mechanisms and Mitigation Options

Fatigue is the structural deterioration that can occur as the result of repeated stress/strain cycles caused by fluctuating loads or temperatures. After repeated cyclic loading, if sufficient localized micro-structural damage has been accumulated, crack initiation can occur at the most highly affected locations. Subsequent cyclic loading and/or thermal stress can cause crack growth.

A brief description of the relevant fatigue-related degradation mechanisms is provided below.

4.3.1 High-Cycle Fatigue

The most 'classical' fatigue-related degradation mechanism is high-cycle (HC) fatigue. HC fatigue involves a high number of cycles at a relatively low stress amplitude (typically below the material's yield strength but above the fatigue endurance limit of the material). High cycle fatigue may be:

- Mechanical in nature, i.e., vibration or pressure pulsation or due to flow-induced vibration (FIV). FIV can induce HC fatigue in otherwise normally passive components merely through interaction of flow adjacent to the component or within the system, establishing a cyclic stress response in the component. Additionally, power uprates are also of concern here as an increase in flow may change the acoustical characteristics of the system and excite a HC mode where a resonant frequency is achieved.
- Thermally induced due to mixing of cold and hot fluids where local instabilities of mixing lead to low-amplitude thermal stresses at the component surface exposed to the fluid.
- Due to combinations of thermal and high-cycle mechanical loads such as might occur on pump shafts in the thermal barrier region.

4.3.2 Low-Cycle Fatigue

Low-cycle fatigue is due to relatively high stress range cycling where the number of cycles is less than about 10^4 to 10^5. To induce cracking at this number of cycles, the stress/strain range causes plastic strains that exceed the yield strength of the material, and the cycling causes local plasticity leading to more rapid material fatigue degradation. In reactor coolant system components, the cumulative combined effects of reactor coolant system pressure and temperature changes are considered in the component design analysis. The stress cycling that contributes to low-cycle fatigue is generally due to the combined effects of pressure, piping moments and local thermal stresses that result during normal operation.

4.3.3 Thermal Fatigue

Thermal fatigue is due to cyclic stresses that result due to changing temperature conditions in a component or in the piping attached to the component. Thermal fatigue may involve a relatively low number of cycles at a higher stress (e.g., plant operational cycles or injection of cold water into a hot nozzle) or due to a high number of cycles at low-stress amplitude (e.g., local leakage effects or cyclic stratification).

4.3.4 Environmental Fatigue

Environmental fatigue concerns the reduction in fatigue "life" in reactor water environment compared to "room temperature air" environments. Environmental fatigue involves two primary elements; the effects of a reactor water environment on the overall fatigue life of reactor components (as represented by either multiplying the location fatigue usage factor by a 'factor' to account for environment or use of an environment-adjusted fatigue design curve), and the potential accelerated growth of an identified defect due to reactor water environments. With regard to the evaluation of fatigue for component aging management, consideration of the effects of a particular reactor water environment on the overall fatigue life is more relevant. Environmental acceleration of fatigue crack growth is important in dispositioning detected/postulated flaws in a component to permit continued operation.

Fatigue crack initiation and growth resistance is governed by a number of material, structural and environmental factors, such as stress range, temperature, fluid oxygen content, mean stress, loading frequency (strain rate), surface roughness and number of cycles. Cracks typically initiate at local geometric stress concentrations, such as welds, notches, other surface defects, and structural discontinuities. The presence of an oxidizing environment or other deleterious chemical species can accelerate the fatigue crack initiation and propagation processes. For example, oxidation can produce pits in the surface of some alloys that act as stress concentrators and potential fatigue crack initiation sites.

4.4 ASME Code Rules on Fatigue

Design against fatigue damage is based on fatigue curves in Section III, Appendix I (e.g., Figures I-9.1 and I-9.2) of the ASME Code. These curves indicate the number of stress cycles of given amplitude of stress intensity that is required to reach a usage factor of 1.0. The fatigue curves are based on test data taken in air at room temperature reduced by a factor of 2 on stress range or 20 cycles to failure, whichever is most conservative, to account for scatter of data, size effects, roughness, and service environment. For carbon and low-alloy steel materials, the most adverse conditions of mean stress are used to correct the test data prior to applying these factors. The ASME Code includes analytical approaches and criteria for determining usage factors for Class 1 components. For Class 1 code components, the usage factor must be shown to be less than 1.0 for the component life. However, a fatigue usage factor of unity does not imply crack initiation because of the safety factors applied to the stress amplitude or number of allowed cycles for the Code fatigue curves.

The crack growth that follows fatigue crack initiation can be predicted if the crack can be characterized and if the cyclic stress field is known. Procedures for performing crack growth analyses are contained in Section XI of the ASME Code.

4.5 Service Experience of Fatigue

Mitigation of fatigue damage for existing components is accomplished by reducing the magnitude of the applied loads or thermal conditions or reducing the number of cycles of loading. For thermal transients, reduction in the rate of temperature change for extreme temperature cycles can be effective. However, the normal operating cycles are not generally the source of significant fatigue damage in nuclear plants. The observed fatigue cracking has generally been due to high-cycle fatigue as a result of conditions not anticipated at the time of original plant design.

Major areas where fatigue failures and leakage have occurred are as follows:

4.5.1 RCS Piping

A number of fatigue issues have been identified, as described below.

a. The major occurrence of leakage has been due to mechanical vibration-induced cracking of small attached lines (primarily socket welded instrument lines). Power uprate has contributed to a number of recent incidences.

b. Thermal fatigue has also caused cracking in normal flowing lines where relatively colder water is injected into flowing RCS lines.

c. Thermal fatigue has also occurred in a number of normally stagnant branch lines attached to flowing RCS lines. The source has been thermal stratification/cycling due to valve in-leakage in up-horizontal running safety injection line configurations and swirl-penetration thermal cycling in down-horizontal drain/excess letdown lines. This is being addressed by the MRP Fatigue ITG, where guidelines are planned in mid-2005. An interim guideline was issued in 2001.

d. Although no occurrences of leakage have been identified, an issue related to surge line stratification was identified in 1988. The issue was resolved by analysis; however, the computed usage factors were quite high. Environmental fatigue effects may be significant for these lines.

e. Other potentially susceptible locations include PWR charging nozzles and BWR RHR tees, where significant thermal transients can occur in some plants.

4.5.2 Reactor Pressure Vessels

The effects of fatigue are adequately managed by adherence to the plant design basis, where thermal transients were considered in the original plant designs. The notable exception was BWR feedwater nozzles and control rod drive nozzles, where the effects of cold water injection caused cracking early in the life of some plants. Mitigating actions and continued monitoring have been implemented and have proved to be effective.

4.5.3 Pressurizers

There have been no known fatigue failures in pressurizers. However, recent considerations of cold water insurge to pressurizers have been identified that may be a contributing factor to leakage that has been observed in pressurizer heater sleeve welds. The pressurizer spray nozzle is also affected by some significant thermal transients. Pressurizer surge nozzles can be affected by thermal stratification conditions in the surge line.

4.5.4 Steam Generator Shell, Tubes, and Internals

Steam generator feedwater nozzles have exhibited cracking as a result of thermal stratification and cycling, but high oxygen content of the feedwater for low-power conditions may have also increased environmental effects. Girth weld cracking of the steam generator shells and feed-water nozzle blend radii have also been observed, where both hot/cold water thermal fatigue and an environmental contribution were identified.

4.5.5 RPV Internals Components

The major issue identified has been that due to flow-induced vibration of BWR steam dryers. This has led to cracking of the vessel-attached support brackets at several plants.

The area of environmental fatigue is still evolving and is under considerable discussion in the technical community, Code bodies, and regulatory agencies. Laboratory data indicate that, for many materials, the fatigue resistance is lower in reactor service environments than in room-temperature air. Code safety factors may bound this difference in some cases. In addition, the effects of flow adjacent to affected components may reduce some of the environmental effects.

Work continues to better quantify these effects and to determine how these effects may be factored into Code acceptance criteria for design.

4.6 Areas for Further Research

Although fatigue is not perceived to be in issue of safety consequence based on studies reported in NUREG/CR-6674, the combined effects of adverse loadings and environmental effects may lead to more cracking in the future than has been observed in the past. In addition, the effects of power uprate have increased the occurrences of flow induced vibration failures and related damage to component supports. Thus, research in the following areas is recommended:

- Develop a better understanding of the relationship between laboratory environmental testing and actual reactor water conditions. The conditions in laboratory testing are significantly different than those observed in actual flowing reactor water. In addition, material conditioning between the extremes of actual cyclic conditions may be beneficial in reducing environmental effects. Although this has been primarily identified as a license renewal issue, the laboratory effects are real and indicate that the fatigue resistance in a water environment is not as good as previously thought.
- Investigate high-cycle fatigue effects due to hot and cold water mixing. Several incidences of cracking in France have led to EDF embarking on research programs in this area. Participation in these efforts and determination of applicability to all regions of mixing in U.S. plants should be considered.
- Improve methods for predicting and quantifying flow-induced vibration and acoustic loadings. A number of cases have been identified that have resulted in piping and component wear and failure.
- Past attention to fatigue issues has related primarily to pressure-retaining components. Additional more detailed evaluations are probably needed to determine flow-induced fatigue effects and safety consequences for reactor internals and possibly other support components.

4.7 References

Key references providing further details on materials degradation due to fatigue will be added to this section during the next revision of the Degradation Matrix.

5. Reduction in Toughness (RiT)

This section addresses material degradation mechanisms that lead to a reduction in the fracture toughness of the material with increasing time. Because a high level of fracture toughness is a design assumption for most LWR pressure boundary and internal components, degradation mechanisms that lead to reductions of toughness are of high significance for many components. Two distinct degradation mechanisms are of concern, thermal aging and radiation embrittlement.

5.1 Equipment

Reduction in fracture toughness is one key aging degradation mechanism for a number of major components throughout the primary pressure boundary of both PWRs and BWRs and is also the primary concern for vessel internals. Major components potentially affected by thermal aging induced fracture toughness reduction, include:

1. Pressure vessels (RPV, pressurizer and SG shell, including cladding, attachment welds and vessel nozzles)
2. RPV internals
3. Piping
4. SG tubing

Major components potentially affected by both thermal aging and irradiation induced fracture toughness reduction include:

Pressure vessels
RPV internals

For the cast austenitic stainless steel (CASS) parts in the PWR reactor coolant system (RCS), the synergistic effect of thermal aging and radiation embrittlement, due to lack of data, has been identified by NRC in licensing renewal SERs as requiring special attention. The RCS components in PWRs having specific parts fabricated from cast austenitic stainless steel are:

1. Reactor Coolant Pumps – pump casings and covers.
2. Pressurizer – surge nozzles (on CE and Westinghouse pressurizers).
3. Safety and Relief Valves – valve bodies and bonnets.
4. Reactor Coolant Piping – fittings, nozzle safe-ends and connected piping of CE and Westinghouse primary coolant and surge line systems.
5. Auxiliary System Piping – fittings, nozzle safe-ends and connected piping.

The BWR primary coolant pressure boundary and internal components that are fabricated from cast austenitic stainless steel include:

1. Recirculation pump bowls and covers
2. Flow control and gate valve bodies
3. Some internals including some portions of the jet pumps.

However, no specific issue has been identified by NRC regarding BWR CASS parts.

5.2 Materials

Numerous materials are used in the components mentioned above, including:

1. Pressure vessels (RPV, pressurizer and SG shell, including cladding, attachment welds and vessel nozzles) – carbon steel (CS), low-alloy steel (LAS), stainless steel (SS) cladding, and various weld metals, depending upon the parent material used.
2. RPV Internals – SS, cast austenitic stainless steel (CASS), nickel-based alloys, and various weld metals, depending upon the parent material used.
3. Piping – SS, CASS, CS, LAS, nickel-based alloys, and various weld metals, depending upon the parent material used.
4. SG tubing – CS and LAS.

All of the above materials are potentially susceptible to fracture toughness reduction either because of thermal aging or radiation embrittlement or both.

5.3 Fracture Toughness Reduction Mechanisms & Mitigation Options

Loss of fracture toughness in various reactor components cited above can occur through thermal aging or radiation embrittlement. Although no data have been gathered to evaluate its significance, the potential synergistic effect of thermal aging and radiation embrittlement requires attention.

5.3.1 Thermal Aging

Thermal aging has been shown to cause precipitation of additional phases in the ferrite such as formation of an α phase by spinoidal decomposition, nucleation and growth of an α phase, or nucleation and growth of carbides at the ferrite/austenite phase boundaries. Development of these additional phases results in an increase in hardness and yield strength of the casting, with a corresponding reduction in fracture toughness properties. As a result, the component becomes more susceptible to brittle fracture when sufficient tensile loadings are present to drive crack growth. A brittle fracture occurs when the ferrite phase becomes continuous or the ferrite/austenite phase boundary provides an easy path for crack propagation in the presence of an existing flaw and sufficient stresses. This type of failure is due to cleavage of the ferrite or separation of the ferrite/austenite boundary and is termed "channel fracture."

The effects of thermal aging on casting fracture toughness have been shown to saturate once conditions leading to predominantly brittle fracture occur. This saturation effect is associated with development of channel fracture conditions. While the extent of reductions in casting fracture toughness due to thermal aging is related to operating temperature, time at temperature, casting method (static vs. centrifugal), and material composition (molybdenum and ferrite content), available research results indicate that the saturation fracture toughness (Cvsat) can be correlated to casting chemical composition, material properties and the casting method. The actual casting toughness decreases logarithmically with increased operating time toward this "infinite-time" saturation value so the use of Cvsat as a measure of casting fracture toughness is conservative.

Thermal aging embrittlement of materials other than CASS used in reactor components includes (1) temper embrittlement and (2) strain aging embrittlement. Ferritic and low-alloy steels are subject to both of these degradation mechanisms, but wrought stainless steels are not affected by either mechanism.

Temper embrittlement of low-alloy steels is caused by the diffusion and segregation of impurity elements, such as phosphorous, tin, antimony and arsenic, into the grain boundaries after prolonged exposure to temperatures in the range 662°F (350°C) to 1067°F (575°C). At temperatures above this range, the impurities tend toward solution in the ferrite matrix. For example, little or no grain boundary segregation is observed at temperatures above 1157°F (625°C). At temperatures below this range, very long exposure times are necessary for the impurities to diffuse to, and segregate in, the grain boundaries. The presence of carbon tends to accelerate the embrittlement process, due to preferential segregation of the impurities at the interface between grain boundary carbides and ferrite grains. The role of other alloying elements, such as chromium, nickel, magnesium, and molybdenum, in the acceleration or retardation of the temper embrittlement process has been studied extensively. The principal manifestation of temper embrittlement in low-alloy steels is an increase in ductile-to-brittle transition temperature, due to the change from predominantly cleavage fracture (before temper embrittlement) to predominantly intergranular fracture along impurity segregation paths (after temper embrittlement).

Strain aging embrittlement occurs in cold-worked ferritic steels when they are subjected to temperatures in the range of 500 to 700°F, and is caused by the pinning of dislocations by interstitial impurities (nitrogen, carbon, etc.). Post-weld heat treatment of reactor vessel components following cold working during fabrication mitigates, but does not eliminate, the effects of strain aging embrittlement. However, following post-weld beat treatment, residual strain aging embrittlement has only a slight effect on the ductility and fracture toughness of LWR vessel component materials under the environmental and loading conditions of interest.

5.3.2 Radiation Embrittlement

Radiation embrittlement results in an increase in the material's yield and ultimate strengths, with a corresponding decrease in material ductility and resistance to flaw propagation (fracture toughness). Radiation embrittlement in ferritic steels is measured by an increase in the ductile-to-brittle transition temperature (RT_{NDT}) and a drop in the Charpy upper shelf energy. Embrittlement in ferritic steels is primarily caused by the formation of copper-rich precipitates that harden the matrix and reduce toughness. Neutron irradiation enhances the formation of these precipitates.

Extensive databases exist for evaluating and predicting embrittlement in reactor vessel steels. These data are obtained from vessel material surveillance capsules in both PWR and BWR vessels, and from test reactors. Embrittlement trend curve models such as Reg. Guide 1.99, Rev. 2 are used to predict the shift in RT_{NDT} and drop-in upper -shelf energy as a function of copper, nickel, and fluence.

Significant variations in radiation embrittlement have also been observed between different types of steel (CS, LAS, etc.) and even between different heats of the same steel. These differences are caused by variations in metallurgical structure and composition. Improved empirical trend models have recently been developed to describe the combined effects of copper, nickel, phosphorus, irradiation temperature, and neutron flux and fluence on the embrittlement of pressure vessel steels. Steels with a very low copper content show little embrittlement in spite of high radiation doses. The effect of irradiation exposure at low temperatures (below 525°F) increases the rate of embrittlement damage. Weld metal is generally more sensitive to radiation embrittlement than base metal. Impurity chemistry, chemistry variability, and different microstructure are responsible for the greater sensitivity of the weld metal. In 2002, this improved trend curve model was approved in a revision to ASTM Standard Guide E900.

Stainless steels are also affected by irradiation exposure, but do not exhibit a ductile-to-brittle transition. In stainless steels, reduction in the ductile fracture toughness properties is associated with microstructure changes resulting from the effects of neutron interactions. Neutrons interact with atoms in the crystal lattice, both directly and indirectly, to displace atoms in the lattice and alter material properties through formation of dislocations, interstitials, and vacancies. Segregation of material impurities also occurs.

Data are available from austenitic stainless steel components exposed to neutron irradiation in experimental and thermal reactors. They show that significant reductions in material J-integral values and tearing modulus values appear at approximately one displacement per atom (dpa). Reductions in these fracture toughness properties appear to saturate at fast neutron exposures greater than 10 dpa.

Currently, there is a lack of substantive fracture toughness data for austenitic stainless steels exposed to a neutron fluence exceeding ~10^{21} n/cm^2 in an LWR environment. The bulk of existing data are developed from materials irradiated in experimental reactors. Differences in neutron spectra of experimental reactors and light water reactors could result in actual material property changes. Specific data regarding irradiation exposure of cast stainless steels in an LWR environment are particularly limited.

5.3.3 Synergistic Effects

The NRC Staff, in section XI.M13 of NUREG 1801, has proposed the existence of potential "synergistic" effects of combined thermal aging and radiation embrittlement in cast austenitic stainless steel components. To date, no data have been presented to prove or disprove the existence of such synergistic effects.

5.3.3 Void Swelling Effects

Void formation is a mechanism in which radiation-induced vacancies accumulate in metal to form microscopic voids. If a large number of voids form, termed void swelling, dimensional changes can occur and loads at connection points (for example, at bolted or welded joints of structural members) may also be altered. Thus void swelling could potentially affect the intended functionality of certain component(s). Based on available fast-reactor data, significant fracture toughness reduction of SS materials can also occur if void swelling is large (i.e., greater than several percent).

5.4 Areas for Further Research

- The fracture toughness reduction mechanisms described above require a combined consideration of materials and operating conditions. Consequently, their effects on the actual service performance of reactor components could be minimal either because the materials are more resistant or the environment is less aggressive.

- The EPRI BWR Vessel and Internals Program (BWRVIP) and Reactor Internals Issue Task Group (RI-ITG) of the EPRI Materials Reliability Program (MRP) have ongoing and planned programs to test irradiated SS materials to determine fracture toughness reduction due to irradiation. In the MRP RI-ITG program, CASS samples were thermally aged and then irradiated in a fast reactor. These samples will be tested to determine both separate effects and synergistic effects on fracture toughness reduction. Efforts are also underway in the RPV Integrity ITG to investigate the potential effects of a thermal aging-induced reduction in the fracture toughness of RPV materials. Data up to 200,000 hours indicates no significant reduction in material fracture toughness.

- The following additional work is needed to address more quantitatively the generic fracture toughness reduction issues discussed earlier in this section:

    1. PWR-specific data to address CASS materials and weld metals
    2. PWR-specific data to address void swelling effects
    3. Correlating PWR and BWR data
    4. Correlating LWR data and fast/test reactor data
    5. Evaluating mitigation options if required

- A number of other R&D topics on which further R&D is needed were identified in the course of a review of this version of the Degradation Matrix by teams of subject-matter experts. These topics will be included in the next revision of the Degradation Matrix supporting material.

5.5 References

Key references providing further details on materials degradation mechanisms that lead to a reduction in fracture toughness will be added to this section during the next revision of the Degradation Matrix.

Material Information Summaries

CONTENTS

Material Information Summaries

Material Information Summaries

Material Information Summaries

1. Introduction

1.1 Objective

The objective of this document is to provide summaries of relevant information regarding materials for use in conjunction with the Degradation Matrix being developed by EPRI. The applications of the materials covered in this document are service in the reactor coolant system and in reactor internals.

Most of the information in these summaries is taken from the Materials Handbook for Nuclear Plant Pressure Boundary Applications, EPRI report 1002792, December 2002, hereafter referred to as the "Materials Handbook (Report 1002792, Dec. 2002)." The Materials Handbook should be consulted for further details, and for additional references to supporting information.

1.2 Acronyms Used in This Document

Acronym	Meaning
ANL	Argonne National Laboratory
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing Materials
BWR	Boiling water reactor
BWRVIP	Boiling Water Reactor Vessel Internals Program
CASS	Cast austenitic stainless steel
CGR	Crack growth rate
EAC	Environmentally Assisted Cracking
ECP	Electrochemical potential
EPR	Electrochemical potentiokinetic reactivation
FAC	Flow-accelerated corrosion
HAZ	Heat affected zone
HWC	Hydrogen water chemistry
IASCC	Irradiation assisted stress corrosion cracking
IHSI	Induction heating stress improvement
LAS	Low-alloy steel
LTCP	Low-temperature crack propagation
LWR	Light water reactor
MSIP	Mechanical stress improvement process
NMCA	Noble metal chemical addition

Material Information Summaries

NWC	Normal water chemistry
PWR	Pressurized water reactor
PWSCC	Primary water stress corrosion cracking
RT	Radiographic test
RT_{NDT}	Reference temperature / nil ductility temperature
SCC	Stress corrosion cracking
USE	Upper shelf energy
UT	Ultrasonic test
VCD	Vacuum carbon deoxidation

2. Carbon and Low-Alloy Steels

This section covers carbon and low-alloy steels that are used as pressure boundary materials in pressure vessels, piping, and similar applications in reactor coolant systems of LWRs. The reasons for use of carbon and low-alloy steels for these applications are their combination of relatively low cost, good mechanical properties in thick sections, good weldability, and high resistance to SCC. With regard to reactor vessels, the grades of low-alloy steels that are used also have acceptably low rates of embrittlement when subjected to neutron flux for long periods of time. In many reactor coolant applications, the carbon and low-alloy steels have been clad on the inside wetted surface with corrosion-resistant materials such as austenitic stainless steels or nickel-base alloys.

Welds in these carbon and low-alloy steels are covered in Section 3 rather than in this section. The term “weld” is intended to cover both the weld metal and the heat-affected zone (HAZ) of the base material. For example, flaws that develop in the HAZ are mainly covered in Section 3 (carbon and low-alloy steel weld metal) and not in this section.

2.1 Service Experience

Carbon and low-alloy steel pressure vessel steels are widely used in nuclear power plants. Service experience has generally been good. However, there have been some problems, as summarized below. Note that problems related to welds and cladding are mainly covered in Section 3 rather than here.

- The most significant service-induced flaws have been cracks at nozzles associated with mixing of lower-temperature water with hot water in a vessel, i.e., thermal fatigue cracks in BWR reactor vessel feedwater nozzles and control rod drive return line nozzles.[i,ii,iii,iv,v,vi] Significant inspections and repairs were required in the late 1970s and early 1980s to address this problem. The design and procedure changes made at that time seen to have been effective as there have been no further reported occurrences.

- A through-wall crack developed in the LAS wall of an early BWR (Garigliano) secondary steam generator channel head.[vii,viii,ix] The crack appeared to have grown due to SCC and was attributed to the presence of cracks in the Alloy 400 type cladding (Alloy 190 weld metal) that acted as initiating sites for the SCC in the base material, combined with high residual stresses due to an ineffective post weld heat treatment.

- A few flaw indications have been detected in vessel base materials by UT performed for baseline or in-service inspections, e.g., due to laminations or inclusions in the steel plates or forgings. The base material flaws have rarely if ever required repair. There appear to be no reported cases of service-induced growth of flaws present in the base plates or forgings.

- Corrosion fatigue has been identified as a potential concern in piping applications, but in practice has not been a serious problem except at feedwater connections to some PWR steam generators and at some German BWR feedwater nozzles.

- Steels in the reactor vessel core belt line region are subject to embrittlement due to neutron irradiation. Embrittlement of the base materials of western design LWRs has generally not been a serious problem. However, some welds in early generation PWRs have been found to be especially sensitive to embrittlement, and have required significant programs to address the resulting embrittlement concerns, as discussed in Section 3.

- Significant numbers of cracks have developed in the cladding of BWR reactor vessel heads, as discussed in Section 5. In some cases, the cracks have penetrated short distances into the low-alloy steel base material. This cracking has required significant inspection and analysis to demonstrate the continued safe condition of the affected parts. In a few cases, it has been concluded that the cladding cracks may have penetrated into the base material as the result of service, but it appears more likely that such penetration occurred during fabrication.

2.2 Material Compositions & Properties

Most of the carbon and low-alloy steel pressure boundary material used in reactor coolant service is in the wrought or forged form, although cast material is occasionally used, e.g., for channel heads and piping elbows. Pressure vessel shells have often been fabricated using plates rolled to the correct curvature and then welded. Flanges and nozzles typically are seamless forgings. The trend has been to eliminate as many welds as possible by use of ring forgings for vessel shells and by use of integrally forged nozzles. The materials used are covered by applicable ASME/ ASTM specifications. Typical reactor coolant system applications and specifications of carbon and low-alloy steels include:

- Reactor vessel plates, e.g., low-alloy steel to SA-533, Type B, Class 1, with internal cladding except in some BWR applications.

- Reactor vessel forgings, e.g., low-alloy steel to SA 508, Grade 2, Class1 (formerly Class 2) or Grade 3, Class 1 (formerly Class 3), with internal cladding except in some BWR applications.

- Steam generator shell plates, e.g., low-alloy steel to SA-533, Type A, Class 1 or Class 2.

- Steam generator tube sheets (e.g., low-alloy steel to SA-508, Grade 2, Class 1 (formerly Class 2), or SA-508, Grade 2, Class 2 (formerly Class 2a), with cladding on primary face.

- Steam generator channel heads, e.g., carbon steel to SA-216, Grade WCC, with internal cladding.

- Pressurizer shell, e.g., carbon steel to SA-516, Grade 70, or low-alloy steel to SA-533, Type B, Class 1, with internal cladding.

- Reactor coolant piping, e.g., carbon steel to SA-516, Grade 70, with internal cladding.

The compositions and mechanical properties of some of the typical carbon and low-alloy steels used in LWRs reactor coolant system service are shown in Tables 2-1 and 2-2.

Material Information Summaries

Table 2-1
Specified Compositions of Some Typical Carbon and Low-Alloy Steels Used for Pressure Vessels (wt %)

ASME/ASTM Spec. Type Grade, UNS No.	C	Mn	P	S	Si	Cu	Ni	Cr	Mo	V	Nb
SA/A 105 CS Forgings K03504	.035 max	.60 - 1.05	0.035 max	.040 max	0.10 - 0.35	0.40 max (1)	0.40 max (1)	0.30 max (1)	0.12 max (1)	0.05 max	0.02 max
SA/A 106 Seamless CS Pipe Grade B, K03006	.30 max	0.29 - 1.06	0.035 max	0.035 max	0.10 min	0.40 max (2)	0.40 max (2)	0.40 max (2)	0.15 max (2)	0.08 max (2)	--
SA/A 216 Casting Grade WCB, J03002	0.30 max	1.00 max	0.04 max	0.045 max	0.60 max	0.30 max (3)	0.50 max (3)	0.50 max (3)	0.20 max (3)	0.03 max (3)	--
SA/A 302 Pr. Vessel Plates, Mg-Mo Grade B, >2", K12022	0.25 max	1.15 - 1.50	0.035 max	0.035 max	0.15 - 0.40	--	--	--	0.45 - 0.60	--	--
SA/A 508 CS & AS Forgings Gr 2-Cl. 1 & 2, K12766 (4)	0.27 max	0.50 - 1.00	0.025 max	0.025 max	0.15 - 0.40	--	0.50 - 1.00	0.25 - 0.45	0.55 - 0.70	0.05 max	--
SA/A 508 CS & AS Forgings Gr 3-Class 1, K12042 (5)	0.25 max	1.20 - 1.50	0.025 max	0.025 max	0.15 - 0.40	--	0.40 - 1.00	0.25 max	0.45 - 0.60	0.05 max	--
SA/A 516 CS Plates Gr. 70, K02700	(6)	0.85 - 1.20	0.035 max	0.035 max	0.15 - 0.40	--	--	--	--	--	--
SA/A 533 Pr. Vessel Plates Type A, Cl. 1 & 2, K12521	0.25 max	1.15 - 1.50	0.035 max (7)	0.035 max (7)	0.15 - 0.40	(7)	--	--	0.45 - 0.60	(7)	--
SA/A 533 Pr. Vessel Plates Type B, Cl. 1, K12539	0.25 max	1.15 - 1.50	0.035 max (7)	0.035 max (7)	0.15 - 0.40	(7)	0.40 - 0.70	--	0.45 - 0.60	(7)	--

(1) The sum of Cu, Ni, Cr & Mo shall be ≤ 1.00%, and the sum of Cr and Mo shall not exceed 0.32%.
(2) By agreement, limits for vanadium and niobium may be increased to 0.10% and 0.05% respectively.
(3) The sum of Cr and Mo shall not exceed 0.32%.
(4) Grade 2, Class 1 was formerly known as Class 2
(5) Grade 3, Class 1 was formerly known as Class 3
(6) Carbon max. varies with thickness: 0.5-2": 0.28% max; 2-4": 0.30% max, 4-8": 0.31% max
(7) SA533 suggests, for reactor core belt line applications, the following limits: Cu ≤ 0.10%, P ≤ 0.012%, S ≤ 0.015%, and V ≤ 0.05%

Table 2-2
Specified Room Temperature Mechanical Properties for Some Typical Carbon and Low-Alloy Steels Used for Pressure Vessels

ASME/ASTM Spec. Type Grade , UNS No.	Tensile Strength min, (ksi)	Yield Strength min, (ksi)	El. in 2" min (%)	Reduction in Area (%)	Hardness max	Charpy V-Notch min. ave. ft-lbs
SA/A 105 CS Forgings K03504	70	36	22 (1)	30	187 HB	--
SA/A 106 Seamless CS Pipe Grade B, K03006	60	35	30 long. 16.5 trans.	--	--	--
SA/A 216 Casting Grade WCB, J03002	70 - 95 min-max	36	22	35	--	--
SA/A 302 Pr. Vessel Plates, Mg-Mo Grade B, K12022	80 - 100 min-max	50	18	--	--	--
SA/A 508 CS & AS Forgings Gr 2-Cl. 1 & 2, K12766 (4)	Cl1:80-105 Cl2:90-115 min-max	Cl1: 50 Cl2: 65	Cl1: 18 Cl2: 16	Cl1: 38 Cl2: 35	--	Cl1: 30 at 40°F Cl2: 35 at 70°F
SA/A 508 CS & AS Forgings Gr 3-Class 1, K12042 (5)	80 - 105 min-max	50	18	38	--	30 at 40°F
SA/A 516 CS Plates Gr. 70, K02700	70 - 90 min-max	38	21	--	--	--
SA/A 533 Pr. Vessel Plates Type A, Cl. 1 or 2, K12521	Cl1:80-100 Cl2:90-115 min-max	Cl1: 50 Cl2: 70	Cl1: 18 Cl2: 16	--	--	--
SA/A 533 Pr. Vessel Plates Type B, Cl.1, K12539	80 - 100 min-max	50	18	--	--	--

(1) The ASTM/ASME specifications provide alternate rules for elongation that may be used.
(2) 70 - 90 for thickness ≤ 2.5", 65 - 85 for thickness > 2.5" and ≤ 4"
(3) 50 for thickness ≤ 2.5", 45 for thickness > 2.5" and ≤ 4"

2.3 Main Limitations

The main limitations with regard to use of carbon and low alloy steels are as follows:

- Radiation-induced embrittlement of core beltline materials has been found to be sensitive to the chemistry of the materials. This applies to both base materials and welds, but embrittlement has been more of a problem with welds than base materials. It is important to control the amounts of deleterious materials, especially copper, phosphorous and nickel, in both the base materials and weld materials. The following guidance in NRC Regulatory Guide 1.99, Revision 2, should be considered: "For beltline materials in the reactor vessel for a new plant, the content of residual elements such as copper, phosphorous, sulfur, and vanadium should be controlled to low levels. (For more information, see the Appendix to ASTM Standard Specification A 533.) The copper content should be such that the calculated adjusted reference temperature at the 1/4T position in the vessel wall at the end of life is less than 200°F. In selecting the optimum amount of nickel to be used, its deleterious effect on radiation embrittlement should be balanced against its beneficial metallurgical effects and its tendency to lower the initial RT_{NDT}."

- Significant numbers of low-alloy steel parts that were clad with stainless steel or nickel chromium iron weld deposits experienced underclad cracking in the base material. Since this problem is related to cladding, it is covered in Section 3.

- Austenitic stainless steel cladding is subject to sensitization during vessel heat treatment and is then susceptible to IGSCC in service in BWRs. This IGSCC sometimes penetrates into the base material, i.e., into the carbon or alloy steel plate or forging. Since this problem is related to cladding, it is covered in Section 3.

2.4 Welding and Heat Treatment

The carbon and low-alloy steels covered in this section are readily weldable. Standard procedures for carbon and low-alloy steels can be used. ASME Code requirements need to be observed with regard to preheat temperatures and post weld heat treatment. ASME Code fracture toughness requirements need to be met by both base materials and weldments.

2.5 Research & Development Results

2.5.1 Radiation Embrittlement

Since radiation embrittlement has generally been most severe at welds, this topic is covered in Section 3 rather than here.

2.5.2 Stress Corrosion Cracking in LWR RCS Environments

In normal or hydrogen BWR water chemistry with low conductivity and low impurity levels, it is difficult to sustain SCC in LAS.[x] However, several conditions can increase the likelihood of SCC, including higher potentials (such as due to higher oxygen levels or the presence of copper ions), higher sulfates and conductivity (such as due to resin ingress transients), cyclic stresses, and low flow velocities. Considering these factors and service experience, SCC is unlikely in LAS parts in BWRs but could occur under unusual circumstances.

Tests indicate that SCC of pressure boundary steels does not occur under normal PWR reactor coolant conditions, i.e., under conditions with normal PWR reactor coolant chemistry characterized by fully deoxygenated conditions with low ECP (near the hydrogen line on a Pourbaix diagram). However, a limited amount of testing indicates SCC of weld HAZ materials could possibly occur at high stress intensities under PWR operating conditions.

Tests indicate that SCC of pressure boundary steels and weldments can occur if oxidizing conditions develop. This result is consistent with the occurrence of SCC that was a factor (along with corrosion fatigue) in PWR steam generator shell cracking at girth welds that has been observed at several plants. This SCC is attributed to oxidizing conditions that were probably present in the feedwater inlet area of these older PWR steam generators during early years of operation, and to the presence of hardened areas associated with welds, although some non-weld areas (such as inspection holes) have also cracked. The possible occurrence of SCC under oxidizing conditions is also supported by the BWR secondary steam generator shell cracking observed at Garigliano. While tests indicate that SCC of reactor coolant wetted steels is unlikely, the probability of its occurrence would increase if oxidizing conditions were allowed to develop, with the likelihood of SCC increasing as the oxidizing potential increases, and as the amount of manganese sulfide inclusions in the steel increases.

Under the abnormal, somewhat oxidizing, conditions required for SCC of pressure vessel steels, tests indicate that the following trends apply:

- Susceptibility to SCC increases as the amount of sulfides in the steel increases.
- Susceptibility is greater for steel strained in a direction perpendicular to the rolling direction, and thus perpendicular to elongated sulfides.
- Higher flow rates tend to decrease susceptibility.
- Weld HAZs are the most susceptible areas.

Many of the above observations are considered to be the result of manganese sulfide inclusions in the steel having a strong influence on the SCC behavior. This is because the concentration of dissolved sulfur species at the crack tip is considered to be a controlling factor in the rate of SCC, with dissolution of manganese sulfides at the crack tip being an important source of sulfur for the cracking process.

2.5.3 Corrosion Fatigue in LWR RCS Environments

An extensive amount of work has been done over the past 30 or more years to characterize both the crack initiation and crack propagation behavior of pressure vessel steels in LWR environments. The main focus of this work has been to quantify the effects of variables such as cyclic frequency (strain rate), stress (R ratio, the ratio of minimum to maximum stress or stress intensity), water quality (oxygen and conductivity), sulfur content of the steel, and temperature on crack initiation and growth rate. The results of the work on crack propagation (crack growth rate) have been reflected in the crack growth rate curves in Section XI of the ASME Code. However, the results of work on the effect of environment on crack initiation (S-N curves) are not covered in the Code. Thus, dealing with environmental effects in the fatigue design of new vessels is left to the discretion of the owner and designer. The NRC has sponsored work at the Argonne National Laboratory (ANL) that provides guidance in regard to environmental effects on fatigue crack initiation.

Several recent EPRI reports provide extensive information on corrosion fatigue and should be consulted for more details.[xi] For example, EPRI report TR-106696 provides a recent comprehensive summary of service experience and research results.

The main trends that have been identified by corrosion fatigue research regarding crack initiation, i.e., fatigue life or S-N behavior, are as follows:

- Fatigue life tends to decrease as strain rate decreases, i.e., for a given number of cycles, the cycles are more damaging if they occur at low frequencies with low strain rates.

- Fatigue life decreases as temperature increases in the range from 150°F (66°C) to 610°F (321°C), but is insensitive to temperature below 150°F (66°C).

- Fatigue life decreases as oxygen increases above the 50 - 100 ppb range.

- Fatigue life decreases as the sulfur content of the steel increases, up to about 0.015%, but is insensitive to sulfur above that level.

The main trends that have been identified by corrosion fatigue research regarding crack growth rates are as follows:

- Crack growth rates in water environments are increased as compared to growth rates in air. In general, the increase is moderate (median increase of about 1.7) and is no more than a factor of three. However, under some circumstances, crack growth rates can be increased by a factor of ten or more. This type of enhanced crack growth rate is known as "environmentally assisted cracking" or EAC.

- For a given stress intensity range, crack growth rates are increased by increased R ratio.

- Crack growth rates, in terms of crack growth per cycle, increase with decreasing strain rate, increasing oxygen content, increasing conductivity, and increasing sulfur content of the steel. Crack growth rates tend to decrease as the flow rate increases.

- Crack growth rates at low stress intensity, e.g. 15 ksi√in. (17 MPa√m), are higher at 350°F (177°C) than at 550°F (288°C), and the stress intensity threshold for crack growth to occur is lower at the lower temperature. However, peak crack growth rates, at high stress intensity, are higher at 550°F (288°C) than at 350°F (177°C).

2.5.4 Crack Growth Rate Model and Crack Tip Chemistry

A significant amount of work has been done on the development of a model for crack growth in stainless and carbon/low-alloy steels that takes into account the effects of crack tip chemistry as well as other variables, and on related investigations of conditions that develop in the crack tip region. The crack tip model and crack tip chemistry evaluations have been found useful for understanding and evaluating the effects of variables such as corrosion potential, oxygen content, water conductivity, pH, flow rate, and steel sulfur content on crack growth rate. However, the crack growth rate model and related crack tip experimental investigations are mainly directed at understanding the development of local environments and their influence on crack growth, rather than on materials, and thus are outside the scope of this section.

2.5.5 Improved Initial Properties

Research continues at several organizations to develop improved materials for reactor vessels and other vessels. This work seems to be concentrated on material to SA508, Grade3, Class 1 or 2 (formerly Class 3 or 3A). Some of the most noteworthy results of this work include:

- The trend in Japan is to use integral forgings for steam generator primary heads, steam drum heads, and shell segments with no longitudinal welds and with integral nozzles and manways. The trend is to use higher-strength Class 2 material (65 ksi yield strength) rather than lower-strength Class 1 material (50 ksi yield strength), and to use low silicon with aluminum to improve impact properties.

- A Korean supplier investigated three steel making practices for reactor vessel shell forgings of SA508 Grade 3 Class 1 (formerly Class 3) material: vacuum carbon deoxidation (VCD), modified VCD with aluminum, and silicon killing. It found that the modified VCD and silicon killing processes provided a significant improvement in fracture toughness, as compared to the plain VCD process.

- A U.S. company investigated the effects of composition and heat treatment on the toughness of SA508 Grade 3 Class 2 material for pressure vessels. It found that additions of both Al and N (relative to mid range values) provided the best strength/toughness combination.

2.5.6 Warm Pre-Stressing

Experiments have shown that if a flaw is stressed at elevated temperature and then loaded to fracture at a lower temperature, the apparent fracture toughness will be higher than if no pre-stressing at the elevated temperature had taken place.[xii] The increased toughness is attributed to development of a plastic zone surrounding the border of the flaw at higher temperatures that is locked in place due to elevation of strength at lower temperatures. The warm pre-stress effect is often significant, e.g., a factor of two or more increase in fracture toughness can result.

2.5.7 General Corrosion & Corrosion Product Release

General corrosion rates and corrosion product release rates of carbon and low-alloy steel in the controlled conditions of primary and secondary coolant systems have been extensively studied. This work has shown that there are two effective chemistry approaches for limiting the rates of general corrosion and corrosion product release (assuming material, temperature, flow rate, etc., are held fixed): (1) maintaining the oxygen concentration over about 15-20 ppb, and (2) maintaining a high pH. Maintaining oxygen over about 15-20 ppb is the strategy adopted by BWRs, as discussed in the BWR Water Chemistry Guidelines and many supporting documents. Maintaining high pH is the strategy adopted by PWRs for both primary and secondary systems, as discussed in the PWR Primary and Secondary Water Chemistry Guidelines and many supporting documents.

2.5.8 Hydrogen Water Chemistry Effects on Carbon & Low-Alloy Steel Piping

Hydrogen water chemistry (HWC) is being used in lieu of normal water chemistry (NWC) at many BWRs as a countermeasure against IGSCC of stainless steel piping and internals. Tests of the effects of HWC on the structural behavior of carbon and low-alloy steel indicate that HWC increases margins against fatigue and SCC of these materials. However, the reduction in the oxygen concentration associated with HWC can increase the rate of FAC/erosion-corrosion in some parts of the plant. This concern, and approaches for dealing with it, are covered in the BWR Water Chemistry Guidelines.

2.5.9 Boric Acid Corrosion

Tests and plant experience demonstrate that corrosion rates of carbon and low-alloy steels can be high, up to 10 inches/year (25 cm/year), if exposed to flowing aerated hot boric acid. Exclusion of oxygen reduces corrosion rates to low levels, a few mils/year or less. Results of investigations of the effects of boric acid solutions on the corrosion of carbon and low-alloy steel are reviewed and summarized in EPRI's Boric Acid Corrosion Guidebook.[xiii] Some key points with regard to these investigations include the following:

1. When carbon and low-alloy steels are immersed in boric acid solutions in the temperature range from room temperature up to 590°F (310°C), corrosion rates are less than 1 mil/ year (25 μm/year) for cases where the oxygen level is low for long periods of time, and in the range of 6-17 mils/year (0.15 – 0.41 mm/year) for cases where oxygen is present at the start of a 70-hour or longer test, but no new oxygen is added during the test.

2. Exposure to aerated borated water results in corrosion rates up to a high of about 10 in./year (25 cm/year). There are several situations of interest:

3. Leaking water at ambient temperature on to a steel part at 70°F-100°F (21-38°C). In this case, the corrosion rate ranges between 2 and 7 mils/year (0.05 and 0.18 mm/year).

4. Leakage of borated water (e.g., drips) on to moderate temperature steel parts that are not so hot that the metal stays dry. Maximum corrosion rates occur for parts at about 200-220°F (93-104°C), which can corrode at rates up to 10 in./year (25 cm/year).

5. Corrosion in tight crevices into which boric acid leaks at low rates. In this situation, where the parts exposed to oxygen stay at high temperature and are mostly dry, the corrosion rate is low in the crevice itself because oxygen is largely excluded from the wetted area. However, corrosion at the exit of the crevice can be high because of access to oxygen.

6. Corrosion in tight crevices into which boric acid leaks at high rates. In this situation, if the leak rate becomes sufficient that the area stays continuously wetted, the corrosion rate can be high, in the range of inches/year (cm/year), as seen at Davis Besse.

2.6 References

[i]. R. Snaider, BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking: Resolution of Generic Technical Activity A-10 (Technical Report), NUREG-0619-REV-1, Nov. 1980.

[ii]. B. M. Gordon and G. M. Gordon, “Corrosion in Boiling Water Reactors,” Metals Handbook, Volume 13, Corrosion, p927-937, ASM International, 1987.

Material Information Summaries

[iii]. B. M. Gordon, D. E. Delwiche, and G. M. Gordon, "Service Experience of BWR Pressure Vessels," Performance and Evaluation of Light Water Reactor Pressure Vessels, PVP-Vol. 119, p9-17, ASME, 1987.

[iv]. N. G. Cofie, et al., Evaluation of Stress Corrosion Crack Growth in Low Alloy Steel Vessel Materials in the BWR Environment (BWRVIP-60), EPRI TR-108709, March 1999.

[v]. R. Snaider, BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking: Resolution of Generic Technical Activity A-10 (Technical Report), NUREG-0619-REV-1, Nov. 1980.

[vi]. Vol. BWR-2, III. Reactor Vessel, items 19, 20, 23, 25, 34, and 38, Nuclear Power Experience, RCG/Hagler, Bailly, Boulder, Colorado, 1995.

[vii]. B. M. Gordon, et al., "Service Experience of BWR Pressure Vessels," Performance and Evaluation of Light Water Reactor Pressure Vessels, ASME PVP-Vol. 119, p9-17, 1987.

[viii]. G. D'Anna and G. F. Sica, "Plans for Repair of Garigliano BWR, Proceedings: Seminar on Countermeasures for Pipe Cracking in BWRs, Volume 4, EPRI WS-79-174, May 1980.

[ix]. Nuclear Power Experience, Vol. BWR-1, Garigliano, A. Plant Description and History of Operation, pages dated 1973 - 1985, RCG/Hagler, Bailly, Boulder, Colorado, 1995..

[x]. Evaluation of Stress Corrosion Crack Growth in Low Alloy Steel Vessel Materials in the BWR Environment (BWRVIP-60), EPRI TR-108709, March 1999.

[xi]. Recent EPRI reports regarding corrosion fatigue of pressure vessel steels include: Evaluation of Environmental Fatigue Effects for a Westinghouse Nuclear Power Plant, TR-110043, April 1998; Environmentally-Assisted Fatigue Crack Initiation in Low Alloy Steels, TR-109051, Nov. 1997; Corrosion Fatigue of Water-Touched Pressure Retaining Components in Power Plants, TR-106696, Nov. 1997; An Environmental Factor Approach to Account for Reactor Water Effects in Light Water Reactor Pressure Vessel and Piping Fatigue Evaluations, TR-105759, Dec. 1995; Environmental Acceleration of Fatigue Crack Growth in Reactor Pressure Vessel Materials: Volumes 1 and 2, TR-102796, Aug. 1993, and Analysis of Fatigue Crack Growth Rate Data for A508 and A533B Steels in LWR Environments, TR-102793, August 1993.

[xii] B. D. Macdonald, et al., "Analysis of Warm Prestress Data," Fatigue and Fracture Mechanics: 27th Volume, p243-263, ASTM STP 1296, 1997.

[xiii]. E. S. Hunt, Boric Acid Corrosion Guidebook, Revision 1: Managing Boric Acid Corrosion Issues at PWR Power Stations, EPRI, Palo Alto, CA: 2001. Report 1000975.

3. Carbon and Low-Alloy Steel Welds

This section covers welds in carbon and low-alloy steel (LAS) base materials in pressure vessels, piping, and similar applications in reactor coolant systems of nuclear power plants. The welds are mainly used to join pressure boundary material segments to each other to form a continuous pressure boundary, but are also used to repair material defects and to attach non-pressure boundary parts (such as external support lugs) to the pressure boundary.

3.1 Service Experience

Experience with welds in carbon and low-alloy steels in reactor coolant system service has generally been good, with relatively few service induced problems. Adverse experience is summarized in the following paragraphs.

Welds in the core beltline region are subject to embrittlement due to neutron irradiation. Experience has shown that the rates of embrittlement are not a serious concern at most western design LWRs. However, some welds in early generation PWRs have been found to be especially sensitive to radiation embrittlement, and have required significant programs to address the resulting embrittlement concerns.[i] A small lead PWR, Yankee Rowe, was decommisioned because of issues related to reactor vessel embrittlement.[ii,iii] Shutdown for radiation embrittlement concerns is judged unlikely to occur at any of the currently operating plants, but addressing radiation embrittlement issues is likely to continue to require significant research efforts, especially for plant life extension. A review of this topic is provided in Section 10.1 of Chapter (I)1 in the Materials Handbook (Report 1002792, Dec. 2002).

- A significant number of flaw indications have been detected in pressure vessels by ultrasonic testing (UT) performed for baseline or in-service inspections. Most of these flaws have been associated with welding or cladding. Some of the weld- and clad-related flaws have led to repairs being made, especially when the flaws were detected before operation. However, there appear to be no reported cases of service-induced growth of weld flaws present since initial construction.

- There have been a few cases of crack initiation and growth in PWR steam generator shells at transition cone girth welds.[iv] These cracks appear to have been initiated as a result of weld damage, thermal stress cycles, and the occasional presence of oxidizing conditions. No new cases of this type of cracking have been reported since about 1991, and it appears that current water chemistry controls minimize the likelihood of serious cracking of this type in the future.

- A through-wall crack developed in the LAS wall of an early BWR (Garigliano) secondary steam generator channel head. The crack appeared to have grown due to SCC and was attributed to the presence of cracks in the Alloy 400 type cladding (Alloy 190 weld metal) that acted as initiating sites for the SCC in the base material, combined with high-residual stresses due to an ineffective post weld heat treatment.[v,vi,vii]

3.2 Material Compositions and Properties

A large variety of welding materials and welding processes are used to join carbon and low-alloy steels, and it is not practical to show typical material compositions and material specifications. Section NB-2431.1 of Section III, Division I of the ASME Code requires that weld materials have tensile strength, ductility and impact properties that match those of either of the base materials being welded, as demonstrated by tests using the selected weld material and the same or similar base materials. Section NB-2432.2 of Section III, Division I of the ASME Code requires that the chemical composition of the welding material be in accordance with an appropriate ASME Code welding specification (in Section II.C of the Code), but leaves the choice of the specific material up to the manufacturer.

The most common weld processes used to join carbon steel and LAS parts include submerged arc welding, shielded metal arc welding (SMAW), and gas tungsten arc welding (GTAW). Post-weld heat treatment is generally required per ASME Code rules after welding of the carbon and low-alloy steels used for reactor coolant system service.

3.3 Main Limitations

The main limitations with regard to welds in carbon and low-alloy steel pressure boundary materials are as follows:

Radiation-induced embrittlement of core beltline welds has been found to be sensitive to the chemistry of the materials. Accordingly, it is important to control the amounts of deleterious materials, especially copper, phosphorous and nickel. The following guidance in NRC Regulatory Guide 1.99, Revision 2, should be considered: "For beltline materials in the reactor vessel for a new plant, the content of residual elements such as copper, phosphorous, sulfur, and vanadium should be controlled to low levels. (For more information, see the Appendix to ASTM Standard Specification A 533.) The copper content should be such that the calculated adjusted reference temperature at the 1/4T position in the vessel wall at the end of life is less than 200°F. In selecting the optimum amount of nickel to be used, its deleterious effect on radiation embrittlement should be balanced against its beneficial metallurgical effects and its tendency to lower the initial RT_{NDT}."

Radiographic testing (RT) is typically required by the ASME Code for acceptance of welds. However, in-service inspections typically are performed using ultrasonic testing (UT). It has been found that UT detects some flaws that are not detected by RT, and this has led to the need to evaluate many flaws detected by in-service UT, and sometimes to the need to repair welds after installation and service. To avoid this type of problem, welds that will be UT inspected during in-service inspections should be inspected during fabrication using similar UT methods and acceptance standards as those to be used for in-service inspections.

3.4 Welding and Heat Treatment

ASME Code requirements need to be observed with regard to pre-heat temperatures and post-weld heat treatment. ASME Code fracture toughness requirements need to be met by both base materials and weldments.

When repairs are made to pressure boundary base materials or welds after start of plant operation, it is often not practical to perform post-weld heat treatment. In such cases, temper bead methods are used in compliance with the ASME Code guidance (e.g., Paragraph NB-4622.9, "Temper Bead Weld Repair," in Section III of the ASME Code and Section IWA-4600, "Alternative Welding Methods," in Section XI of the ASME Code).

3.5 Research and Development Results

3.5.1 Radiation Embrittlement

Radiation embrittlement is covered here, rather than in the section on carbon and low-alloy steel base materials because experience has shown that welds are the most critical area of reactor vessels from an embrittlement standpoint. Because of the importance of ensuring reactor vessel integrity, there has been extensive work done to characterize the effects of irradiation on vessel steels, especially at welds, and to develop ways to assess the impact of irradiation-induced changes on margins of safety against rapid fracture. This work has developed approaches for addressing the two main concerns associated with radiation embrittlement: (1) large shifts in the brittle to ductile transition temperature indicating that the material might behave in a brittle manner at unacceptably high temperatures, and (2) lowered upper shelf energy (USE) values indicating that resistance to ductile tearing might be reduced below acceptable levels, even at high temperatures where the material behaves in a ductile manner. The Master Curve approach, which uses fracture toughness measurements to determine the temperature shift in toughness and to determine a statistically based lower-bound fracture toughness, promises to largely resolve the first concern.[viii,ix] Analytical methods using elastic plastic fracture mechanics appear to have resolved the low USE concern.[x,xi,xii,xiii]

Research is continuing to improve the level of understanding in topics such as the effects of material composition, temperature and flux spectrum on embrittlement; the benefits of thermal annealing; use of small punch specimens to monitor embrittlement; and mechanisms and microstructure of embrittlement.

3.5.2 Stress Corrosion Cracking (SCC) in LWR Environments

The treatment of this topic in Section 2.5.2 for carbon and low-alloy steel base materials is equally applicable to welds in these materials and should be consulted for information on this topic. In this regard, most flaws in carbon and low-allow steels that could initiate SCC are likely to be located at welds, and the high residual stresses that could cause SCC initiation and growth are more likely to be present in and around welds than elsewhere in the parts.

3.5.3 Corrosion Fatigue in LWR RCS Environments

The treatment of this topic in Section 2.5.3 for carbon and low-alloy steel base materials is equally applicable to welds in these materials and should be consulted for information on this topic. In this regard, most flaws in carbon and low-alloy steels that could grow due to corrosion fatigue are likely to be located at welds, and the high residual stresses that could aggravate the initiation and growth of flaws due to corrosion fatigue are more likely to be present in welds than elsewhere in the parts.

3.5.4 Other Topics

The research for carbon and low-alloy steels summarized in Section 2.5 is generally as applicable to the welds as to the base materials, e.g., for warm pre-stressing, effects of hydrogen water chemistry, boric acid corrosion, etc. Accordingly, Section 2.5 should be consulted for information regarding research areas that apply to base materials as well as welds.

3.6 References

[i]. R. B. Jenkins, "Chapter 7 - The Utility Industry and Reactor Surveillance," Status of USA Nuclear Reactor Pressure Vessel Surveillance for Radiation Effects, ASTM STP 784, L. E. Steele, Ed., ASTM STP 784, 1983.

[ii]. "Yankee Rowe to be Permanently Shut Down," Nuclear Safety, v33, n3, p467.

[iii]. "The NRC Reverses Earlier Decision, Orders Yankee Rowe to Shut Down," Nuclear Safety, v33, n2, p276-280.

[iv]. Review of Steam Generator Girth Weld Cracking, EPRI, Palo Alto, CA: 1993. TR-103498.

[v]. B. M. Gordon, et al., "Service Experience of BWR Pressure Vessels," Performance and Evaluation of Light Water Reactor Pressure Vessels, ASME PVP-Vol. 119, p9-17, 1987.

[vi]. G. D'Anna and G. F. Sica, "Plans for Repair of Garigliano BWR, Proceedings: Seminar on Countermeasures for Pipe Cracking in BWRs, Volume 4, EPRI WS-79-174, May 1980.

[vii]. Nuclear Power Experience, Vol. BWR-1, Garigliano, A. Plant Description and History of Operation, pages dated 1973 - 1985, RCG/Hagler, Bailly, Boulder, Colorado, 1995..

[viii]. S. T. Rosinski and W. L. Server, "Application of the Master Curve in the ASME Code," International Journal of Pressure Vessels and Piping, v77n, p591-598, 2000.

[ix]. W. Server, S. Rosinski, et al., Application of Master Curve Fracture toughness Methodology for Ferritic Steels (PWRMRP-01), EPRI TR-108390-R1, May 1999.

[x]. "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less than 50 Ft-Lb.," NRC, June 1995.

[xi]. Evaluation of Upper-Shelf Toughness Requirements for Reactor Pressure Vessels, EPRI NP-6790-SL, April 1990.

[xii]. Evaluation Procedure for Service Level C and D Upper shelf Toughness Criteria for Linde 80 Weld Material, EPRI TR-102851, Oct. 1993.

[xiii]. Evaluation of the Toughness Properties and ASME Service Level A and B Upper Shelf toughness Criteria for A 302-B Vessel Steel, EPRI report TR-103434, Aug. 1994.

4. Wrought Stainless Steel

This section covers wrought austenitic stainless steel base materials that are used for piping and other pressure boundary applications and for reactor internals. Wrought austenitic stainless steel has been widely used in BWR and PWR reactor coolant system pressure boundary applications and as a structural material in reactor internals. Pressure boundary applications include use in piping, valve bodies, and pump casings. In reactor internals, wrought stainless steel has been used for applications such as core shrouds, baffle plates, former plates, and core support plates.

4.1 Service Experience

Service experience with wrought stainless steel has mostly been satisfactory, although there have been some significant problems. Highlights of service experience with wrought austenitic stainless steels include:

- In high-temperature BWR applications such as the reactor coolant system (RCS), conventional wrought austenitic stainless steels (Types 304 and 316) have been subject to intergranular stress corrosion cracking (IGSCC) in areas where the material was sensitized during fabrication, such as furnace sensitized safe ends and at weld joint heat affected zones (HAZs). The term "sensitized" refers to a process of chromium carbide precipitation at grain boundaries that can occur when the material is held in the temperature range of 800-1600°F. The precipitation of chromium carbide reduces the chromium concentration at the grain boundaries and makes them susceptible to corrosive attack in BWR reactor coolant environments with normal water chemistry (NWC). In this regard, BWR reactor coolant with NWC contains about 200 ppb oxygen, which raises the electrochemical potential (ECP) to a level at which the coolant is aggressive towards sensitized material. Countermeasures have been developed based on residual stress and ECP reduction and one or more remedial actions have been implemented for all BWRs in the US. In addition, "nuclear grades" of Types 304 and 316 have been developed that are resistant to this type of attack, and these have been used successfully in piping replacements and new applications. These countermeasures have proved to be very effective and incidents of IGSCC in stainless steel piping in BWRs are now rare.

- While IGSCC in BWR piping is no longer a serious problem, it is a continuing issue for BWR reactor vessel internals. The BWRVIP has had an extensive program stating in the 1990s that is addressing topics such as how to ameliorate the problem, how best to inspect affected parts, how to disposition detected flaws, and how to make repairs. The methods for ameliorating IGSCC of reactor internals are similar to those developed for pipe cracking and include use of hydrogen water chemistry (HWC) and noble metal chemical addition (NMCA). The effects of high levels of radiation exposure on the effectiveness of these countermeasures is an area of uncertainty which is the subject of current R&D.

- Wrought stainless steels have also been widely used in PWR reactor coolant systems and reactor internals and have provided relatively trouble free service in PWR applications. The absence of systematic IGSCC problems of the types that have affected BWRs is attributed to the low oxygen content and hydrogen overpressure in PWR reactor coolant, which keeps the ECP well below the range in which IGSCC occurs in pure water environments. The relatively limited number of problems that have occurred in stainless steel parts in PWRs are generally due to either mechanical or thermal fatigue or, in a few cases, to the development in stagnant areas of aggressive environments with chlorides, concentrated boric acid and trapped oxygen.

- While stainless steel piping in PWR reactor coolant service has performed without significant corrosion-induced problems, it has exhibited ID corrosion problems in other PWR applications, including IGSCC at weld joints in stagnant borated water systems, TGSCC in control rod drive mechanism and control element drive mechanism applications where high levels of oxygen are trapped during filling of the RCS,[i] and IGSCC and fatigue cracking in dead legs off of the RCS.[ii,iii] These problems have required remedial actions to prevent recurrence, such as tighter water chemistry control for stagnant systems, use of vacuum filling to minimize oxygen in CRDM and CEDM high points, and avoidance of two-phase conditions in dead legs.

- There have been many fatigue failures of small diameter stainless steel piping in both BWRs and PWRs. These failures are not considered to be caused by the material, but rather to be due to mechanical factors such as high imposed vibratory stresses. Remedial actions have included provision of improved supports, and elimination of sources of vibration. Remediation seems to have been largely successful, based on the current low rate of such problems.

- Thermal fatigue cracking has been a problem in both BWRs and PWRs at locations where colder water is introduced into locations with hotter water. In BWRs, the problem has mainly affected low-alloy steel vessels, e.g., at feedwater nozzles and control rod drive return nozzles. It PWRs, stainless steel nozzles have been susceptible to thermal-fatigue cracking at locations where auxiliary systems connect to the RCS. These problems are considered to be thermal-hydraulic in origin, and not to be indicative of problems with the materials used.

- In both BWRs and PWRs, stainless steel piping has occasionally experienced ODSCC, generally at locations where the piping has been both wetted and contaminated by chlorides. This problem has been addressed by preventing contaminants and leaks from contacting stainless steel piping.

- There have been a few cases of SCC of non-sensitized wrought stainless steels in BWRs, including cracking of non-sensitized 12-inch diameter jet pump inlet riser safe ends made of Type 316L stainless steel,[iv] and some cases in reactor internals.[v] Research to better understand the causes of this type of SCC is ongoing.

4.2 Material Compositions and Properties

The compositions of some of the stainless steels that have been or could be used for reactor coolant system piping and for reactor internals are shown in Table 4-1. Typical specified mechanical properties are shown in Table 4-2.

Table 4-1
Specified Compositions of Some Typical Stainless Steels (wt %)[vi,vii] (maximum values except where otherwise indicated)

Name, UNS No.	C	Mn	Si	Cr	Ni	P	S	Mo	N	Cu	Other
Type 304, S30400	0.08	2.0	1.00	18.0-20.0	8.0-10.5	0.045	0.03	--	--	--	--
Type 304L, S30403	0.03	2.0	1.00	18.0-20.0	8.0-12.0	0.045	0.03	--	--	--	--
Type 304NG	0.020	2.00	0.75	18.00-20.00	8.00-11.00	0.030	0.005	0.5	0.060-0.10	0.50	(1)
Type 316, S31600	0.08	2.0	1.00	16.0-18.0	10.0-14.0	0.045	0.03	2.0-3.0	--	--	--
Type 316L, S31603	0.03	2.0	1.00	16.0-18.0	10.0-14.0	0.045	0.03	2.0-3.0	--	--	--
Type 316NG	0.020	2.00	0.75	16.00-18.00	11.00-14.00	0.030	0.005	2.00-3.00	0.060-0.10	--	(1)
Type 347	0.08	2.0	1.0	17.0-19.0	9.0-13.0	0.045	0.03	--	--	--	Nb≥10xC
Type 347NG	0.030	2.00	1.0	17.00-19.00	9.00-12.00	0.035	0.020	--	--	--	0.2 Co, Nb≥10xC

(1) Co max 0.25, Ta + Nb max 0.05, B max 0.001, Al max 0.04, V max 0.1, Bi + Sn + As + Pb + Sb + Se max 0.02

Table 4-2
Specified Room Temperature Mechanical Properties for Typical Stainless Steels[vi,vii]

Grade	Thermo-Mechanical Treatment	Tensile Strength min (ksi)	Yield Strength min (ksi)	Elong. min (%)	Hardness R_B max
Type 304	annealed	75	30	40	92
Type 304L	annealed	70	25	40	88
Type 304NG	annealed	75	30	40	92
Type 316	annealed	75	30	40	95
Type 316L	annealed	70	25	40	95
Type 316NG	annealed	75	30	40	92
Type 347	annealed	75	30	40	92
Type 347NG	annealed	75	30	40	92

Several types of stainless steel with an “NG” designation are listed in the tables, i.e., Types 304NG, 316NG, and 347NG. The requirements for these grades were developed by the BWR Owners Group and EPRI for use in BWRs. The chemical compositions of these grades are tighter than, but nevertheless meet, those of the normal types (Types 304, 316, and 347, respectively). The mechanical properties also meet those of the normal grades. In other words, the nuclear-grade materials are bought and designed to normal Type 304, 316, and 347 requirements and, in addition, meet the tighter chemistry requirements of the nuclear grade specification.

4.3 Main Limitations

The main limitations with regard to use of wrought austenitic stainless steel for reactor coolant system and reactor internals applications are as follows:

- Wrought stainless steel in reactor coolant system or reactor internals service are susceptible to IGSCC when sensitized. Sensitization occurs when the material is heated within the range of 800-1600°F (427-871°C) as the result of heat treatment or welding, and becomes more significant as the time in this temperature range increases and as the carbon content of the material increases. Sensitization has resulted in major problems in BWR applications, and preventing or dealing with such problems requires strict control of the selection of materials, manufacturing and fabrication, as well careful control of water chemistry during plant operation. Sensitization has generally not been found to be a problem in PWR reactor coolant applications; this is considered to be the result of low oxygen levels and ECPs under PWR conditions.

- As noted above, stainless steel in BWR reactor coolant applications is susceptible to IGSCC in areas where the material was sensitized by welding or other fabrication or manufacturing processes. For these reasons, materials resistant to weld-induced sensitization should be used, such as 304NG or 316NG, and manufacturing and fabrication processes should be controlled so as to ensure that deleterious levels of sensitization are not developed by the processes. In this regard, guidelines prepared by the BWR Owners Group and EPRI for procurement, manufacturing and fabrication should be followed.

- Wrought stainless steel used in reactor internals applications in both BWRs and PWRs is susceptible to irradiation assisted stress corrosion cracking (IASCC) after it is exposed to significant fast neutron fluence. Development of methods to cope with and minimize problems due to IASCC is underway for both BWRs and PWRs.

- Stainless steel in BWR reactor coolant system applications is susceptible to IGSCC in areas where it is cold worked. For this reason, cold working of stainless steels for use in BWR reactor coolant applications should be avoided. For background information on the effects of cold work, reference should be made to Chapter (II) 5, of the Materials Handbook (Report 1002792).

- Stainless steels can be attacked by pitting and SCC at dry-out zones during dry lay-up of systems if the residual water in the system has impurities that become aggressive as their concentration is increased by evaporation. Protection against this type of problem requires that lay-up procedures include steps to clean up and dry the system under controlled conditions.

4.4 Welding and Heat Treatment

Wrought austenitic stainless steel materials are readily weldable using conventional welding processes. Discussion of topics related to welding is contained in Section 5, "Stainless Steel Welds and Clad."

Stress relief heat treatment after welding of normal carbon grades of austenitic stainless steels is generally not performed since it can lead to sensitization of the material. However, low-carbon grades generally can be stress relieved without excessive sensitization.

4.5 Research and Development Results

4.5.1 General Corrosion and Corrosion Product Release in High Temperature and Purity Water

General corrosion rates of stainless steels in the types of environments encountered in high-temperature reactor coolant applications are typically very low and do not need to be considered from a structural standpoint. However, corrosion product release is important. This is because the corrosion products accumulate in the reactor core and can cause problems with heat transfer there, and also can be activated and then transported around the reactor coolant system, resulting in increased after-shutdown dose rates. The science and technology for predicting and controlling corrosion product release and transport are described in the BWR and PWR water chemistry guidelines.[viii]

4.5.2 BWR Piping IGSCC Causes and Remedies

Because IGSCC of BWR piping was an early and widespread problem, a large amount of research was performed by the industry to understand the problem and to develop remedies. In summary, the main causes of the IGSCC were determined to be (1) susceptibility of normal carbon -grade stainless steels at weld joint HAZs to IGSCC in BWR reactor coolant due to sensitization caused by welding, (2) the high-residual stresses at weld joints caused by welding, and (3) the relatively oxidizing environment of BWR normal-chemistry water. The presence of high-residual stresses and surface damage due to grinding were found to be aggravating factors.

Based on the research performed from about 1975 to 1988, remedial measures were developed and applied at all operating domestic BWRs. The remedial measures have included (1) replacement of piping using materials with greater resistance to IGSCC, and using fabrication methods that result in low susceptibility to IGSCC, (2) repairs of existing piping by means such as induction heating stress improvement (IHSI), weld overlays, mechanical stress improvement process (MSIP), and last pass heat sink welding, and (3) use of hydrogen water chemistry (HWC) and imposition of tighter controls on water purity

4.5.3 SCC Crack Growth Rates in BWR Environments

Several different groups have reviewed crack growth rate data and developed conservative models of crack growth rate that bound relevant data. These models can be used for plant assessments. Test data and models indicate that the crack growth rate increases with increasing stress intensity, sensitization, coolant conductivity, and potential. It appears that crack growth in non-sensitized stainless steel can be suppressed by keeping the conductivity at 0.1 μS/cm or less.

The BWR Water Chemistry Guidelines review and summarize the crack growth model developed by the BWRVIP.[ix,x]

A 1997 article reviews knowledge regarding effects of flow velocity on crack growth rate.[xi] Some models and data indicate that higher velocity results in flushing of impurities from cracks and, therefore, reduces the crack growth rate. However, other models indicate that higher velocity can increase the potential within the crack or at the crack mouth, and thus, increase crack growth rates. Studies reported in the BWR Water Chemistry Guidelines (Section 4.2) showed that higher flow rate decreased crack growth rate.

A 1991 article discussed results of tests comparing crack growth rates of non-sensitized 316L stainless steels in solutions of different conductivity with that of sensitized 304 stainless steel.[xii] In water with a conductivity of 0.5 μS/cm, the crack growth rate of non-sensitized 316L stainless steel was about a tenth of that of sensitized Type 304 stainless steel, and in water with 0.1 μS/cm conductivity, no crack growth was observed in non-sensitized 316L stainless steel.

4.5.4 Corrosion Fatigue

Research indicates that the environment can have a strong effect on both crack initiation (fatigue life) and crack propagation rate. The fatigue life of stainless steel (number of cycles to development of a small crack) is affected by temperature, oxygen content and conductivity of the water, and the strain rate. The largest adverse effect is associated with slow strain rates in water environments. Fatigue life is lower in low oxygen content water than in high oxygen content water although, as discussed below, crack growth rates are affected in the opposite manner. Appropriate ways to account for environmental effects when evaluating fatigue usage factors for plants applying for life extension are the subject of ongoing discussions between the industry and the NRC.

Fatigue crack growth rates for stainless steel are affected by temperature, exposure to water environments, oxygen concentration or ECP, conductivity, and the degree of sensitization. Part of the crack growth appears to be due to SCC. This results in crack growth rates increasing due to the same factors that cause SCC to increase: increasing as stress intensity increases, oxygen levels and ECP increase, conductivity increases, and degree of sensitization increases. Appropriate crack growth rates for use in ASME Code Section XI flaw evaluations are presently under review by the ASME.

4.5.5 Sensitization

The degree of sensitization has a strong effect on the propensity of stainless steel to initiate cracks in BWR environments and other oxidizing environments, and also on the crack growth rate in these environments. The electrochemical potentiokinetic reactivation (EPR) method described in ASTM G108 is often used to measure the degree of sensitization.

Research regarding the occurrence of sensitization indicates that sensitization is promoted by increasing carbon content, increased time in the sensitization temperature zone of about 800-1600°F (427-871°C), and by prior cold work that induces martensite. Grinding a surface can generate a layer of martensite that can sensitize at 550°F (288°C) in two to ten years. Nitrogen and molybdenum in the steel decrease sensitization, helping to explain the improved resistance of Type 316 vs. Type 304 stainless steel, and providing part of the basis for development of nitrogen-containing nuclear grade stainless steels.

Most research results on effects of long times at operating temperatures indicate that sensitization can occur or worsen during plant lifetimes as a result of low-temperature sensitization.

4.5.6 Development of Long Length Seamless Pipe Segments

A 1997 paper reports the successful development of long lengths of forged austenitic stainless steel piping for use in reactor coolant system pipe service.[xiii] The material composition was similar to that of 304NG, i.e., had low carbon to avoid sensitization, and enough nitrogen to restore strength to normal carbon grade levels. Lengths up to about 24 feet were fabricated. The incentives for developing such forgings were reported as being a desire to (1) minimize welds and the in-service inspection burden associated with welds, (2) avoid use of castings, which raise embrittlement concerns as the result of the ferrite content required in castings to minimize hot cracking, and (3) improve inspectability by avoiding use of castings, which are difficult to inspect by UT.

4.6 References

i NRC Licensee Event Report (LER) 99-004-01, "Control Rod Drive Seal Housing Leaks and Crack Indications," Consumers Energy Company - Palisades Nuclear Plant, Docket No. 05000255, report date Nov. 8, 2000, filed in NRC's Public Electronic Reading Room.

ii. P. Berge, et al., "Corrosion and Cracking of Stainless Steels and Cobalt Alloys in Primary Circuit Piping of Light Water Reactors," Proceedings of the Fourth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, p11-74 to 11-87, NACE 1990.

iii. J. Economou, et al., "Stress Corrosion of Piping Sections in the Dead Legs of the Primary Circuit," Proceedings of the International Symposium Fontevraud III, Contribution of Materials Investigation to the Resolution of Problems Encountered in Pressurized Water Reactors, p547-557, SFEN 12-16 Sept. 1994.

iv. NRC Information Notice No. 84-89, "Stress Corrosion Cracking in Nonsensitized 316 Stainless Steel," Dec. 7, 1984.

v. BWR Water Chemistry Guidelines - 2000 Revision, EPRI TR-103515-R2, Feb. 2000.

vi. J. R. Davis, Ed., ASM Specialty Handbook, Stainless Steels, p22&23, ASM, 1994.

vii. Alternative Materials: BWR Recirculation Piping System Replacement, Volume 1, Table 1-1, EPRI Report NP-6723-D, June 1990.

viii. PWR Primary Water Chemistry Guidelines, Revision 4, Volumes 1 and 2, EPRI TR-105714-R4, March 1999.

ix. BWR Water Chemistry Guidelines - 2000 Revision, Section 2.10.5.1, EPRI TR-103515-R2, Feb. 2000.

x. Evaluation of Crack Growth in BWR Stainless Steel RPV Internals (BWRVIP-14), EPRI TR-105873, Mar. 1996.

xi. P. L. Andresen, "Effects of Flow Rate on SCC Growth Rate Behavior in BWRs," Proceedings of the Eighth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, p603-614, ANS, 1997.

xii. A. Sudo and M. Itow, "SCC Growth and Intergranular Corrosion Behavior of Type 316L Stainless Steel in High Temperature Water," Proceedings of the Fifth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, p251-257, ANS, 1992..

xiii. F. Morin, et al., "Forging Technology Adapted to the Manufacture of Nuclear PWR Austenitic Primary Piping," ASTM STP Steel Forgings, Volume 2, p65-78, ASTM, 1997.

5. Stainless Steel Welds and Clad

Austenitic stainless steel welds are widely used to join austenitic wrought and cast stainless steel base materials to each other. They are also sometimes used in dissimilar metal welds to join carbon or low-alloys steel nozzles to stainless steel piping. In addition, austenitic stainless steel weld-deposited cladding is widely used in carbon steel or low-alloy steel vessels and piping to separate the steel from the reactor coolant so as to reduce corrosion rates, and thereby, reduce pickup of corrosion products by the coolant.

5.1 Service Experience

Service experience with stainless steel welds and cladding has generally been good, with some exceptions:

- IGSCC at weld joints of stainless steel piping in applications involving exposure to BWR reactor coolant was first detected in 1965 and became a large-scale problem starting in the mid 1970s. At first, only smaller-diameter pipes appeared to be affected. However, in the late 1970s and early 1980s, cracks in larger-diameter lines were detected showing that pipes of all diameters were susceptible. Large-scale research and development efforts were performed during the 1970s and 1980s to identify the causes of the IGSCC and to develop remedies. Based on this research, remedial measures were developed and applied at all operating domestic BWRs during the 1980s. By the mid 1980s, application of these remedial measures had reduced the impact of IGSCC in BWR piping to low levels, where it remains today.

- IGSCC started affecting BWR core internals in the 1990s, with much of this IGSCC being at weld joints. Since then, BWRVIP has been working on the development of methods to deal with this problem.

- Significant IGSCC has not been observed in stainless steel weld joints in PWR reactor coolant system service. This is attributed to the low oxygen levels and ECP in PWR reactor coolant systems which make the material not susceptible to IGSCC, even if sensitized. While at least one case of minor propagation of PWSCC cracks in Alloy 182 weld metal into the adjoining stainless steel has been noted, this experience showed that the cracks quickly terminated once they penetrated into the stainless steel.[i]

- Significant numbers of alloy steel parts that had been clad with stainless steel or nickel-chromium-iron weld deposits experienced underclad cracking during original construction. The underclad cracking developed as the result of two different mechanisms. The first mechanism involved reheat cracking, and the second involved cold (hydrogen) cracking. These experiences, and ways to prevent them, are covered in Section 9 of Chapter (I)1 of the Materials Handbook (Report 1002792, Dec. 2002).

- Service-induced cracks have occasionally been detected in austenitic stainless steel cladding in BWRs. These cracks are[ii] the result of sensitization of the cladding caused by vessel post-weld heat treatment, combined with low ferrite numbers in the cladding and the oxidizing conditions at the outlet of the BWR core. In a few cases, it has been concluded that the cladding cracks may have penetrated into the base material as the result of service, but it appears more likely that such penetrations occurred during fabrication, i.e., are underclad reheat or hydrogen-induced cracks and not extensions of SCC in the cladding. The clad cracking, especially where penetration into the base material has been noted, has required significant inspection and analysis to demonstrate the continued safe condition of the affected parts. This experience, and ways to prevent such cracking, are covered in Section 9 of Chapter (I)1 of the Materials Handbook (Report 1002792, Dec. 2002).

5.2 Material Compositions and Properties

The main welding alloys that are used for welds joining austenitic stainless steels to each other are Types 308, 308L, 316 and 316L. The most common welding processes used are gas tungsten arc welding (GTAW), shielded metal arc welding (SMAW), and submerged arc welding (SAW). Types 309 and 310 are sometimes used in dissimilar metal welds between carbon or low-alloy steel parts and austenitic stainless steel parts. Weld deposited cladding is most often made using the SAW process, and is often deposited using Type 308 or 308L, although sometimes Type 309 is used to provide for dilution by the base material. The compositions of bare electrodes of common grades are shown in Table 5-1.

Table 5-1
Typical Specified Compositions of Stainless Steel Bare Electrodes (wt %)[iii] (maximum values except where otherwise indicated)

Type, UNS Number	C	Mn	Si	Cr	Ni	P	S	Mo	N	Cu
ER308, S30880	0.08	1.0-2.5	0.30-0.65	19.5-22.0	9.0-11.0	0.03	0.03	0.75	--	0.75
ER308L, S30883	0.03	1.0-2.5	0.30-0.65	19.5-22.0	9.0-11.0	0.03	0.03	0.75	--	0.75
ER309, S30980	0.12	1.0-2.5	0.30-0.65	23.0-25.0	12.0-14.0	0.03	0.03	0.75	0.060-0.10	0.75
ER310, S31080	0.08-0.15	1.0-2.5	0.30-0.65	25.0-28.0	200.0-22.5	0.03	0.03	0.75	0.060-0.10	0.75
ER316, S31680	0.08	1.0-2.5	0.30-0.65	18.0-20.0	11.0-14.0	0.03	0.03	2.0-3.0	--	0.75
ER316, S31683	0.03	1.0-2.5	0.30-0.65	18.0-20.0	11.0-14.0	0.03	0.03	2.0-3.0	--	0.75

Section NB-2431.1 of Section III, Division I of the ASME Code requires that weld materials have tensile strength and ductility and impact properties that match those of either of the base materials being welded, as demonstrated by tests using the selected weld material and the same or similar base materials. Section NB-2432.2 of Section III, Division I of the ASME Code requires that the chemical composition of the welding material be in accordance with an appropriate ASME Code welding specification (in Section II.C of the Code), but leaves the choice of the specific material up to the manufacturer.

5.3 Main Limitations

The main limitation regarding austenitic stainless steel welds and cladding are the following:

Austenitic stainless steels, including weld metal and cast material if ferrite content is low, are susceptible to IGSCC in BWR normal water chemistry environments. Accordingly, precautions need to be taken to prevent sensitization such as use of low carbon grade materials and low heat input welding. For safety related applications, depending on plant specific licensing commitments, specifications for materials and repair work may need to require compliance with NRC Regulatory Guide 1.31, Control of Ferrite Content in Stainless Steel Weld Metal, Revision 3, April 1978, and Regulatory Guide 1.44, Control of the Use of Sensitized Stainless Steel, May 1973. Further guidance related to welds that will be exposed to BWR environments is provided by the NRC in NUREG 0313, Revision 2, and NRC Generic Letter 88-01, which indicate that the material should have less than 0.035% carbon and a minimum of 7.5% ferrite.

Austenitic stainless steel cladding is subject to sensitization during vessel heat treatment and to IGSCC in service in BWRs. If IGSCC causes cracks to develop in the cladding, they can penetrate into the base material, i.e., into the carbon or alloy steel plate or forging. Prevention of this problem involves use of cladding materials that are resistant to sensitization and IGSCC, e.g., have carbon less than 0.035% and have a ferrite content over 7.5%.

Significant numbers of low-alloy steel parts that were clad with stainless steel weld deposits experienced underclad cracking during original construction. The underclad cracking developed as the result of two different mechanisms. The first mechanism involved reheat cracking, and the second involved cold (hydrogen) cracking. Preventive measures include use of a low heat input cladding process, and use of relatively high preheat and post heat temperatures and soak times for each weld pass (not just the first pass).

5.4 Welding and Heat Treatment

For safety-related applications, depending on plant-specific licensing commitments, specifications for materials and repair work may need to require compliance with NRC Regulatory Guide 1.31, Control of Ferrite Content in Stainless Steel Weld Metal, Revision 3, April 1978, and Regulatory Guide 1.44, Control of the Use of Sensitized Stainless Steel, May 1973.

Post-weld heat treatment can lead to sensitization of normal carbon grade stainless steel materials, and thus, should be avoided. However, low carbon grades can often be post weld heat treated without significant sensitization.

5.5 Research and Development Results

5.5.1 Methods to Ameliorate and Prevent IGSCC at Weld Joints

A large amount of research has been conducted to determine the causes of the IGSCC affecting welds in stainless steel piping in BWRs, and then to develop methods to ameliorate this IGSCC. This subject is covered in the Section 4.5.2 and, in more detail, in the BWR Water Chemistry Guidelines,[iv] and in numerous other EPRI reports.

5.5.2 Methods to Inspect Welds

A large amount of research has been conducted to develop improved methods to inspect for flaws in welds in austenitic stainless steel piping. The results of this work are summarized in various reports, including reference[v].

5.5.3 Irradiation-Assisted Stress Corrosion Cracking (IASCC)

Research is ongoing to develop a better understating of the mechanisms involved in the IASCC that can occur at welds in reactor internals and how to predict and deal with it in both BWRs and PWRs.[vi]

5.6 References

i. G. V. Rao, "Metallurgical Investigation of Cracking in the Reactor Vessel Alpha Loop Hot Leg Nozzle to Pipe Weld at the V.C. Summer Station," WCAP-15616 Rev. 0, Jan 2001.

ii

iii. ASME Specification SFA-5.9, "Specification for Bare Stainless Steel Welding Electrodes and Rods," ASME Code, Section II-C.

iv. BWR Water Chemistry Guidelines – 2000 Revision, EPRI, Palo Alto, CA: 2000. Report TR-103515-R2.

v. Development and Qualification of Procedures for Rapid Inspection of Piping Welds, EPRI, Charlotte, NC: 2003. 1007891.

vi. BWRVIP-119: BWR Vessel and Internals Project – Proceedings of the 2003 Workshop on Fracture Toughness and Crack Growth, EPRI, Palo Alto, CA: 2003. 1007822.

6. Cast Stainless Steel

This section covers cast austenitic stainless steel (CASS) that is used in reactor coolant system pressure boundary service and in reactor internals service in nuclear power plants. CASS has been widely used for reactor coolant piping applications in Westinghouse design PWRs, and is used for reactor coolant pump and valve pressure boundary applications in both BWRs and PWRs. CASS is also used in BWR reactor internals in applications such as jet pump assemblies.

While CASS is called an austenitic material, it actually is a duplex austenitic-ferritic material, with ferrite in the 10 to 25% range. Its corrosion and embrittlement properties depend strongly on the ferrite content. Ferrite contents of 7.5% or more, when combined with carbon content less than 0.035%, provide high resistance to IGSCC in BWR normal water chemistry environments.

6.1 Service Experience

Experience with CASS in BWR and PWR reactor coolant service has been as follows:

- In normal BWR reactor coolant system applications service, a few cases of IGSCC of CASS and austenitic stainless steel weld metals have been experienced.[i] Evaluation of service experience and tests have shown that normal carbon grade CASS (e.g., CF8 and CF8M) is somewhat susceptible to IGSCC in locations where it has been sensitized or cold worked, although significantly less susceptible than the corresponding wrought materials (Types 304 and 316). The time required for IGSCC to develop to detectable levels is a function of the degree of sensitization, the amount of ferrite, the level of stress, the amount of cold work, and the specific environment. Section 4, covering wrought stainless steels, should be consulted regarding the process of sensitization and how it affects susceptibility to IGSCC in oxygenated BWR environments. The same process applies to CASS, but less severely as the ferrite content increases. Over the years, essentially all CASS used in reactor coolant applications that did not meet the 7.5% minimum ferrite and 0.035% carbon maximum limits in NUREG 0313, Rev. 2, has had to be remediated to address this susceptibility. At the present time, this program of remediation has resulted in very infrequent new cases of IGSCC being detected in CASS materials in BWR reactor coolant service.

- Service experience with CASS in PWR reactor coolant system service has been excellent, with no reported service induced problems such as stress corrosion cracking or other forms of corrosion.

- CASS is subject to embrittlement as it ages. This is a result of metallurgical changes that occur in the ferrite phase. This can become an issue for reactor coolant applications as plant service life increases. The factors that influence the degree of embrittlement include service time and temperature, ferrite content, molybdenum concentration, and fabrication method. While this issue has not resulted in any reported physical problems, it has been identified as an aging management issue that needs to be addressed in license renewal. It appears that most, if not all, potential embrittlement problems can be adequately addressed by susceptibility assessments, increased inspections, and fracture mechanics analyses. The current situation is that it is considered unlikely that embrittlement will require replacement of CASS in PWRs during 40- or 60-year plant lifetimes.

6.2 Material Compositions and Properties

Typical nuclear power plant applications and material grades of CASS include:

- Reactor coolant and auxiliary system piping: CF8A, CF8M, CPF3M
- Reactor coolant pump casings: Types CF8, CF8A, CF8M
- Reactor coolant valve bodies and fittings: Types CF8A, CF8M

The specifications involved generally are the following:

- ASME/ASTM SA 351/A 351 Castings, Austenitic, Austenitic-Ferritic (Duplex), for Pressure-Containing Parts
- ASME/ASTM SA 451/A 451 Centrifugally Cast Austenitic Steel Pipe for High-Temperature Service

Typical composition limits and room temperature mechanical properties of CASS are shown in Tables 6-1 and 6-2.

Table 6-1
Specified Compositions of Typical Cast Austenitic Stainless Steels (wt %)[ii,iii] (maximum values except where otherwise indicated)

Name, UNS No.	C	Mn	Si	Cr	Ni	P	S	Mo
Type CF8, Type CF8A	0.08	1.50	2.00	18.0-21.0	8.0-11.0	0.040	0.040	0.50
Type CF8M	0.08	1.50	1.50	18.0-21.0	9.0-12.0	0.040	0.040	2.0-3.0
Type CF3, Type CF3A	0.03	1.50	2.00	17.0-21.0	8.0-12.0	0.040	0.040	0.50
Type CF3M, Type CF3MA	0.03	1.50	1.50	17.0-21.0	9.0-13.0	0.040	0.040	2.0-3.0
Type CPF3A	0.03	1.50	2.00	17.00-21.00	8.00-12.00	0.040	0.040	--
Type CPF8A	0.08	1.50	2.00	18.00-21.00	8.00-11.00	0.040	0.040	
Type CPF3M	0.03	1.50	1.50	17.00-21.00	9.00-13.00	0.040	0.040	2.00-3.00
Type CPF8M	0.08	1.50	1.50	18.00-21.00	9.00-12.00	0.040	0.040	2.00-3.00

Table 6-2
Specified Room Temperature Mechanical Properties for Typical Cast Austenitic Stainless Steels[ii,iii]

Grade	Thermo-Mechanical Treatment	Tensile Strength min (ksi)	Yield Strength min (ksi)	Elong. min (%)
Type CF8	annealed	70	30	35
Type CF8A	annealed	77	35	35
Type CF8M	annealed	70	30	30
Type CF3	annealed	70	30	35
Type CF3A	annealed	77	35	35
Type CF3M	annealed	70	30	30
Type CF3MA	annealed	80	37	30
Type CPF8A	annealed	77	35	35
Type CPF8M	annealed	70	30	30
Type CPF3A	annealed	77	35	35
Type CPF3M	annealed	70	30	30
Type AL-6XN	welded, solution treated, and annealed	104	46	30

6.3 Main Limitations

The main limitations with regard to use of CASS in reactor coolant system and reactor internals applications are as follows:

- The material composition and properties should be controlled to provide high resistance to IGSCC, especially if intended for service in a BWR reactor coolant environment. Guidance in this regard is provided by the NRC in NUREG 0313, Revision 2, and NRC Generic Letter 88-01, which indicate that the material should have less than 0.035% carbon and a minimum of 7.5% ferrite.

- Stainless steel in BWR reactor coolant applications, including CASS, is susceptible to IGSCC in areas where it is cold worked. For this reason, cold working of CASS used in BWR reactor coolant applications should be avoided.

- CASS typically has a duplex austenitic-ferritic structure that is susceptible to embrittlement as a result of long times at high temperatures. This is not known to have resulted in any fractures in service, but has resulted in the probable need for increased inspections and analyses for some cast components. For this reason, consideration should be given to limiting embrittlement factors in new applications of CASS, e.g., by limiting the ferrite content and molybdenum content. Guidance regarding factors that influence susceptibility to embrittlement is contained in Section 10.7, "Aging of Cast Stainless Steel," in Chapter I(3) of the Materials Handbook (Report 1002792, Dec. 2002).

6.4 Welding and Heat Treatment

CASS is readily weldable using conventional welding processes. The requirements of NRC Regulatory Guides 1.31 (RG 1.31), Control of Ferrite Content in Stainless Steel Weld Metal, Revision 3, April 1978, and 1.44, (RG 1.44), Control of the Use of Sensitized Stainless Steel, May 1973, should be considered when making welds in safety-related systems, even though they are mainly addressed at wrought materials. RG 1.31 requires that welds have a minimum level of ferrite in order to decrease their susceptibility to hot cracking. RG 1.43 places limitations on applications where weld sensitized material may be used, and includes requirements for material controls and tests to ensure that unacceptable sensitization is not present.

For welds in parts exposed to BWR reactor coolant, the requirements of the NRC in NUREG-0313, Revision 2, and Generic Letter 88-01, “NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping,” January 25, 1988, should be considered. This indicates that low carbon weld metal such as 308L, 316L and 309L should be used, and that the minimum level of ferrite should be 7.5%, or the minimum ferrite number should be 7.5.

For new CASS parts, the material should be solution annealed and quenched after any necessary repair welds are made. Post-weld stress relief is normally not performed following installation welding into the RCS since the stress relief could lead to sensitization and is generally not necessary. Nevertheless, stress relief of low carbon grade materials can be safely be performed assuming that the weld as well as the base materials are low carbon and that the cast and weld materials both have sufficient ferrite (7.5% or more) to provide IGSCC resistance.

6.5 Research and Development Results

CASS is subject to embrittlement as a result of microstructural changes that occur with aging at operating temperatures. The degree of embrittlement is a function of the chemical composition and casting method, with statically cast, high ferrite content, high molybdenum content material having the highest susceptibility. The main effect of the embrittlement is to reduce the fracture toughness at operating temperature. No cases of embrittlement requiring corrective action, such as part replacement, have been reported. However, the NRC is requiring that plants applying for life extension include evaluation of this type of embrittlement in their aging management plans. Two recent EPRI reports provide guidance with regard to evaluation of CASS in connection with life extension.[iv,v] These reports should be consulted for detailed information on this topic. In addition, NRC guidance on this topic is provided in a recent staff evaluation sent to the NEI.[vi] In this evaluation, the NRC provided Table 6-3 showing CASS thermal aging embrittlement screening criteria that are acceptable to them. Materials falling into the potentially susceptible category require evaluation for possible embrittlement. With regard to estimating the delta ferrite content, the NRC recommends that Hull's method be used; this method is covered in a NUREG/CR report.[vii]

Table 6-3
CASS Thermal Aging Susceptibility Screening Criteria

Mo Content (Wt. %)	Casting Method	δ–Ferrite Level	Susceptibility Determination
High (2.0 to 3.0)	Static	≤ 14%	Not susceptible
		> 14%	Potentially susceptible
	Centrifugal	≤ 20%	Not susceptible
		> 20%	Potentially susceptible
Low (0.5 max.)	Static	≤ 20%	Not susceptible
		> 20%	Potentially susceptible
	Centrifugal	ALL	Not susceptible

6.6 References

[i] Justification for Extended Weld-Overlay Design Life, EPRI NP-7103-D, January 1991.
[ii] Specification for Castings, Austenitic, Austenitic-Ferritic (Duplex), for Pressure-Containing Parts, ASME/ASTM SA/A 351.
[iii] Specification for Centrifugally Cast Austenitic Steel for High-Temperature Service, ASME/ASTM SA/A 451.
[iv] Evaluation of Thermal Aging Embrittlement for Cast Austenitic Stainless Steel Components, EPRI 10000976, Jan. 2001.
[v] Evaluation of Thermal Aging Embrittlement for Cast Austenitic Stainless Steel Components in LWR Reactor Coolant Systems, EPRI TR-106092, Sep. 1997.
[vi] Letter from C. I. Grimes (NRC) to D. J. Walters (NEI), “License Renewal Issue No. 98-0030, ‘Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components,’” with attached staff evaluation of same title, May 19, 2000.
[vii] O. K. Chopra, Estimation of the Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems, NUREG/CR-4513, Rev. 1, August 1994.

7. Wrought Nickel Alloys (Including Alloy X-750)

This section covers wrought nickel-base alloys that are used for pressure boundary and reactor internals structural applications in BWRs and PWRs, exclusive of steam generator tubes. The base material alloys covered are Alloys 600 and 690. These alloys are widely used in BWR applications such as reactor vessel nozzle safe ends, core support structures, and shroud bolts. These alloys are also widely used in PWRs for applications such as penetrations and nozzles in PWR reactor coolant system components, control rod drive mechanism (CRDM) and control element drive mechanism (CEDM) nozzles in reactor vessel heads, and instrument nozzles in pressurizers and RCS piping. Alloy X-750, a high strength precipitation hardening alloy is also covered in this section since it has a composition similar to that of Alloy 600, and is susceptible to the same main degradation mechanisms as Alloy 600, i.e., IGSCC in BWRs and PWSCC in PWRs.

Steam generator tubes are not covered here because they are adequately covered in other specialized books, reports and guidelines, e.g., the original and revised Steam Generator Reference Book, various guidelines for tube materials, inspections, and water chemistry, and SGMP reports of numerous workshops covering primary side corrosion, secondary side corrosion, and remedial measures. NEI 97-06, Steam Generator Program Guidelines, references and requires implementation of the following guidelines and assessment documents that address degradation of steam generator tubes:

- PWR Steam Generator Examination Guidelines
- PWR Primary-to-Secondary Leak Guidelines
- PWR Secondary Water Chemistry Guidelines
- PWR Primary Water Chemistry Guidelines
- Steam Generator Integrity Assessment Guidelines
- In Situ Pressure Test Guidelines
- PWR Steam Generator Tube Plug Assessment Document
- PWR Sleeving Assessment Document

7.1 Service Experience

Alloys 600 and X-750 have exhibited significant degradation in both BWRs and PWRs, whereas no corrosion related problems have been detected with Alloy 690. In summary, experience with Alloys 600 and X-750 has been as follows:

- Alloy 600 base material in BWR service has mostly been free of IGSCC except in areas with crevices or welds, although there have been a few exceptions, such as in non-creviced areas in core shroud supports. Alloy 600 in BWR applications has experienced extensive IGSCC at areas such as reactor vessel nozzle safe ends and core support structures with crevices and welds. It is believed that this IGSCC has been aggravated by the sensitization and residual stresses associated with welds.

- Alloy 600 in PWR penetration and nozzle applications has exhibited an increasing amount of PWSCC as PWRs have aged.[i,ii] This type of cracking was first experienced in pressurizer heater sleeves and nozzles, with the early cracking being attributed to the high temperatures in pressurizers (about 650°F (343°C)). As the plants aged, PWSCC then occurred in other lower temperature penetrations and nozzles such as CRDM and CEDM penetrations, and reactor coolant loop instrument nozzles.

- Alloy X-750 has been widely used in internals applications such as fuel assembly holddown springs, control rod guide tube support pins, jet pump beams, and reactor internals bolting. Some components have performed satisfactorily. However, there have been sufficient SCC failures to lead to replacements of many parts, and to extensive research into the causes of the failures and into methods of reducing SCC susceptibility. Several contributory factors to the failure mechanisms have been identified: (1) use of material in a less than optimum heat treatment condition, and thus, with a relatively highly susceptible microstructure, (2) the presence of high peak surface stresses due to a combination of design and residual stresses, and (3) the presence of surface damage from the fabrication process. Extensive laboratory testing indicates that new Alloy X-750 parts will perform satisfactorily if they meet current specification requirements with regard to optimized heat treatment, fabrication sequence, and stress and strain limits. Despite these improvements, several nuclear steam supply system (NSSS) vendors have chosen to replace Alloy X-750 with alternative materials.

7.2 Material Compositions, and Properties

All of the Alloy 600 and 690 base material used in power plant pressure boundary applications is in the wrought or forged form. The base materials, and the matching weld materials, are covered by applicable ASME/ASTM and AWS specifications. The specified compositions of the nickel-base alloys most commonly used for U. S. nuclear plant pressure boundary applications are shown in Table 7-1. Specified mechanical properties are shown in Table 7-2.

Material Information Summaries

Table 7-1
Specified Compositions of Typical Nickel-Base Alloys Used in Pressure Boundary Applications, and Alloy X-750 (wt %)

ASME/ASTM Spec. Type Grade, UNS No.	Ni	Cr	Fe	Mn	C	Cu	Si	S
SB/B 166, Ni-Cr-Fe Alloy Rods, Bars and Wire N06600	72.0 min	14.0 – 17.0	6.0 - 10.0	1.0 max	0.15 max	0.5 max	0.5 max	0.01 5 max
SB/B 166, Ni-Cr-Fe Alloy Rods, Bars and Wire N06690	58.0 min	27.0 - 31.0	7.0 - 11.0	0.5 max	0.05 max	0.5 max	0.5 max	0.01 5 max
SB/B 167, Ni-Cr-Fe Alloy Seamless Pipe & Tube N06600	72.0 min	14.0 – 17.0	6.0 - 10.0	1.0 max	0.15 max	0.5 max	0.5 max	0.01 5 max
SB/B 167, Ni-Cr-Fe Alloy Seamless Pipe & Tube N06690	58.0 min	27.0 - 31.0	7.0 - 11.0	0.5 max	0.05 max	0.5 max	0.5 max	0.01 5 max
SB/B 168, Ni-Cr-Fe Alloy Plate, Sheet & Strip N06600	72.0 min	14.0 – 17.0	6.0 - 10.0	1.0 max	0.15 max	0.5 max	0.5 max	0.01 5 max
SB/B 168, Ni-Cr-Fe Alloy Plate, Sheet & Strip N06690	58.0 min	27.0 - 31.0	7.0 - 11.0	0.5 max	0.05 max	0.5 max	0.5 max	0.01 5 max
SB/B 564, Ni-Cr-Fe Alloy Forgings N06600	72.0 min	14.0 – 17.0	6.0 - 10.0	1.0 max	0.15 max	0.5 max	0.5 max	0.01 5 max
SB/B 564, Ni-Cr-Fe Alloy Forgings N06690	58.0 min	27.0 - 31.0	7.0 - 11.0	0.5 max	0.05 max	0.5 max	0.5 max	0.01 5 max
Alloy X-750 Huntington Alloys Handbook 10M 2-79 T-38 (1)	70.0 0 min	14.00 – 17.00	5.00 – 9.00	1.00 max	0.08 max	0.50 max	0.50 max	0.01 0 max

(1) Ti 2.25-2.75, Al 0.40-1.00, Cb+Ta 0.70-1.20, Co 1.00 max

Table 7-2
Specified Room Temperature Mechanical Properties of Nickel-Base Alloys Used in Pressure Boundary Applications

ASME/ASTM Spec. Type, Grade, UNS No.	Tensile Strength min, (ksi)	Yield Strength min, (ksi)	El. in 2" min (%)
SB/B 166 Ni-Cr-Fe Alloy Rods, Bars and Wire Cold worked annealed or hot worked annealed, N06600	80	35	30
SB/B 166 Ni-Cr-Fe Alloy Rods, Bars and Wire Cold worked annealed or hot worked annealed, N06690	85	35	30
SB/B 167 Ni-Cr-Fe Alloy Seamless Pipe & Tube, hot worked or hot worked annealed, ≤ 5” diam., N06600	80	30	35
SB/B 167 Ni-Cr-Fe Alloy Seamless Pipe & Tube, hot worked or hot worked annealed, > 5” diam., N06600	75	25	35
SB/B 167 Ni-Cr-Fe Alloy Seamless Pipe & Tube, cold worked annealed ≤ 5” diam., N06600	80	35	30
SB/B 167 Ni-Cr-Fe Alloy Seamless Pipe & Tube, cold worked annealed > 5” diam., N06600	80	30	35
SB/B 167 Ni-Cr-Fe Alloy Seamless Pipe & Tube, hot worked or hot worked annealed, ≤ 5” diam., N06690	85	30	35
SB/B 167 Ni-Cr-Fe Alloy Seamless Pipe & Tube, hot worked or hot worked annealed, > 5” diam., N06690	75	25	35
SB/B 167 Ni-Cr-Fe Alloy Seamless Pipe & Tube, cold worked annealed ≤ 5” diam., N06690	85	35	30
SB/B 167 Ni-Cr-Fe Alloy Seamless Pipe & Tube, hot worked or hot worked annealed, > 5” diam., N06690	85	30	35
SB/B 168 Ni-Cr-Fe Alloy Plate, Sheet & Strip Hot rolled plate, annealed, N06600	80	35	30
SB/B 168 Ni-Cr-Fe Alloy Plate, Sheet & Strip Hot rolled plate, annealed, N06690	85	35	30
SB/B 564 Ni-Cr-Fe Alloy Forgings, annealed, N06600	80	35	30
SB/B 564, Ni-Cr-Fe Alloy Forgings, annealed, N06690	85	35	30

Typical age hardening treatments and mechanical properties for various grades of Alloy X-750 are shown in Table 7-3.

Table 7-3
Typical Age Hardening Heat Treatments and Room Temperature Mechanical Properties for Alloy X-750[iii,iv]

Condition	Thermo-mechanical Treatment	Tensile - ksi MPa	Yield - ksi (MPa)	Elong. - %	Red. In Area - %
AH	Hot finish at 1800°F (982°C); equalize at 1625°F (885°C) for 24 hours; age at 1300°F (704°C) for 20 hours; air cool	173 (1193)	119 (821)	26	44
BH	Hot roll; solution anneal at 1800°F(982°C) for 1 hour; age at 1300°F (704°C) for 20 hours	198 (1365)	145 (1000)	22	41
HTH*	35% min. reduction on last hot roll; solution anneal at 2025°F (1107°C) for 1 hour; rapid cool; age at 1300°F (704°C) for 20 hour; air cool	165 (1138)	105 (724)	27	31
HOA*	35% min. reduction on last hot roll; solution anneal at 2025°F (1107°C) for 1 hour; rapid cool; age at 1400°F (760°C) for 20 hour; air cool	154 (1062)	92 (634)	27	36
#1 Temper	Solution anneal at 2100°F (1149°C); 15% cold reduction in area; age at 1350°F (732°C) for 16 hour; air cool	167 (1151)	124 (855)	24	24
Spring Temper*	Solution anneal at 2100°F (1149°C); 30-65% cold reduction in area; age at 1350°F (732°) for 4 hour; air cool	204 (1407)	176 (1214)	16	31

* The EPRI material specification (NP-7032) has three heat treatment conditions: core internals basic (CIB) that is similar to HTH, core internals overaged (CIOA) that is similar to HOA, and core internals spring temper (CIST) that is similar to spring temper. Two of these conditions, CIB and CIOA, are also included in the ASME Code via Code Case N-60-5. These new conditions are recommended for use in future applications of Alloy X-750.

7.3 Main Limitations

The main limitations with regard to use of nickel-base alloys in pressure boundary applications are as follows:

- Alloy 600 in BWR reactor coolant system applications has been found to be susceptible to IGSCC in crevice areas, especially at welds where both weld-induced sensitization and residual stresses are present. For this reason, Alloys 600 is no longer selected for such applications. Alloy 690 and its weld materials have been found to be much more resistant to IGSCC in BWR environments.

- Alloy 600 base material has been found to be susceptible to PWSCC in PWR reactor coolant system applications in locations where the wetted surface has stresses over about 30 ksi (207 MPa). For this reason, Alloy 600 is no longer selected for such applications. Alloy 690 and its weld materials have been found to be highly resistant to PWSCC in such applications.

The main limitations and concerns when using Alloy X-750 in applications involving exposure to reactor coolant are:

- Experience has shown that Alloy X-750 has relatively high susceptibility to intergranular stress corrosion cracking (IGSCC) in reactor coolant environments if the material was produced in accordance with thermo-mechanical processing sequences that were standard in the 1960s and 1970s. This is true both for PWR and for BWR reactor coolant environments. Conditions AH and BH in Table 7-3 are typical of the heat treatments with relatively high susceptibility to IGSCC.

- Even improved versions of Alloy X-750 can experience IGSCC if total tensile stresses are close to or exceed yield, especially if surface damage is present, such as intergranular penetrations from electrodischarge machining (EDM) or damage (e.g., superficial grain boundary oxidation) caused by the precipitation heat treatment.[v] For this reason, it is important that final machining be performed after precipitation heat treatment to ensure that damaged surface layers are removed.

- Surveillance tests of Alloy X-750 performed in core locations where substantial neutron fluences were accumulated show that irradiation greatly limits the strain to failure in corrosive environments, i.e., IASCC is experienced.[vi] Some service experience indicates that microstructural and mechanical property changes occur as a result of time at temperature and irradiation that increase susceptibility to cracking. It is important that allowances be made for such changes (e.g., by reducing allowable stress levels) if Alloy X-750 is selected for use in near-core applications.

7.4 Welding and Heat Treatment

- Alloy 600 and Alloy 690 are routinely welded, generally using Alloys 82 and 182 for Alloy 600, and Alloys 52 and 152 for Alloy 690. There has also been some consideration of using Alloy 72 for welding Alloy 690 based on tests that indicate it provides increased resistance to corrosion.

- Welded Alloy 600 parts have been found to be susceptible to IGSCC in BWR reactor coolant environments, especially when the weldment includes crevice areas. Thus, designs for new or replacement applications should avoid use of welded Alloy 600 located in crevice areas. Wherever possible, use of Alloy 600 should be avoided altogether, and Alloy 690 and Alloy 690 type weld materials used instead.

Welded Alloy 600 parts and Alloy 600 type weld materials in PWRs have been found to be susceptible to PWSCC at locations where areas with high weld residual stresses are exposed to high temperature reactor coolant. Weldments that were stress relieved after welding have rarely experienced PWSCC, even though the material has been sensitized by the stress relief heat treatment. Extensive testing indicates that Alloy 690 and Alloy 690 type weld metals have very high resistance to PWSCC. Thus, Alloy 690 and Alloy 690 type weld materials should be used in lieu of Alloy 600 and Alloy 600 type weld materials in PWR reactor coolant system applications.

Thermal treatment has been found to increase the resistance of Alloys 600 and 690 to PWSCC. Based on tubing practice, thermal treatment of Alloy 600 normally involves 15 hours at 1300°F (704°C), and thermal treatment of Alloy 690 normally involves 10 hours at 1320°F (716°C). Because of its high chromium content, Alloy 690 weldments can be subjected to ASME Code type stress relief heat treatments (e.g., 10 hours at 1150°F (621°C)) with little concern regarding sensitization.

7.5 Research and Development Results

7.5.1 IGSCC Initiation in BWR Reactor Coolant Environments

There have been concerns since the 1970s regarding the possible occurrence of IGSCC in nickel-base alloys in BWR reactor coolant system service. These concerns initially focused on pressure boundary applications such as safe ends, but more recently concerns have arisen in connection with reactor internals applications. In response to these concerns, an extensive amount of testing has been performed to explore the conditions under which IGSCC initiates in nickel-base alloys. The main results of the investigations of the susceptibility to initiation of IGSCC in nickel-base alloys in BWR environments include the following:

- The mechanism involved in IGSCC of Alloy 600 and its weld metals in BWR environments appears to be oxidant driven IGSCC that requires some level of sensitization for it to occur.
- Alloy 600 and its weld metals are susceptible while Alloy 690 and its weld alloys are much more resistant.
- Material in the solution annealed and quenched condition is highly resistant to IGSCC in BWR environments. Material in the furnace sensitized or low temperature sensitized condition is sometimes more susceptible than mill annealed material, but the increase in susceptibility is not large. This is attributed to the mill annealed material already being somewhat sensitized.
- Susceptibility increases as the maximum tensile stress at the wetted surface increases.
- Susceptibility of Alloy 600 in BWR environments is strongly increased by the presence of crevices. Initiation of IGSCC in Alloy 600 without crevices present is difficult, and seems not to have happened in plants, except possibly recently in core support structures.
- Susceptibility to SCC strongly increases as water purity decreases.
- ECPs associated with NWC (e.g., 200 ppb oxygen) are in a range where Alloy 600 is highly susceptible. Reducing the ECP to values provided by hydrogen water chemistry (HWC) or noble metal chemical addition (NMCA) greatly reduces susceptibility.

7.5.2 Crack Growth Rates (CGR) in Nickel-Base Alloys in BWR Reactor Coolant Environments

The major conclusions developed based on tests of CGR for IGSCC in nickel-base alloys in BWR environments are as follows:

- CGR disposition curves have been developed by BWRVIP and are documented in the BWR Water Chemistry Guidelines and a BWRVIP report.[vii]

- CGRs are highest in Alloy 182, next highest in Alloy 600, and very low and about the same in Alloys 82 and 690.

- CGRs decrease strongly as oxygen levels and corresponding ECP values decrease. Crack growth rates in HWC are significantly lower (~10 times) than in NWC.

- CGRs increase as the concentration of sulfates increases.

- Adding zinc in the 5 to 10 ppb range reduces CGRs.

- Sensitizing heat treatments generally increase CGRs in Alloy 600 in BWR environments, but decrease them in Alloy 182.

7.5.3 PWSCC Initiation in PWR Reactor Coolant Environments

Based on extensive testing, the main factors affecting the time to initiation of PWSCC in Alloy 600 are as follows:

- The time to PWSCC crack initiation decreases as the peak total tensile stress at the wetted surface increases, approximately as stress to the minus 4 to 7 power. There appears to be a threshold stress below which no PWSCC occurs; this seems to be at about the elastic limit, which often is about 80% of the conventional yield strength.

- Time to cracking tends to increase as the chromium content increases. In this regard, Alloy 690TT, with about 30% chromium, appears to be nearly immune to PWSCC.

- There appears to be some correlation between processing history and susceptibility to PWSCC of Alloy 600. Low-temperature mill anneals that result in the absence of grain boundary carbides appear to increase susceptibility to PWSCC. Hot worked and annealed material is less susceptible than cold worked and annealed material, when both have decorated grain boundaries. Increased resistance to PWSCC is provided by a stress relief type heat treatment, independent of grain boundary carbide decoration. Material that has been uniformly cold worked to increase its yield strength has lower susceptibility than non-cold worked material when stressed to the same absolute stress. However, it has higher susceptibility when stressed to the same percent of yield strength. Surface cold work that leads to high residual surface tensile stresses after the part is installed strongly increases susceptibility to PWSCC. On the other hand, compressive residual stresses (e.g., from peening) on the surface tend to inhibit PWSCC.

- Susceptibility to PWSCC tends to correlate with microstructure, with the most susceptible material having few intragranular carbides, many random intragranular carbides, and a small grain size, and the least susceptible material having larger grains with many carbides at the grain boundaries. Despite this tendency, there are exceptions, with occasional “good” microstructures being highly susceptible, and “bad” microstructures being resistant.

- Susceptibility to PWSCC tends to increase as strength increases. An exception to this correlation with strength occurs when material has been uniformly cold worked to a higher strength, and then is loaded to a total surface stress (including any residual stresses from deformation) to less than the yield stress. In this case, the material is generally highly resistant to PWSCC.

- PWSCC initiation times decrease as temperature increases, with typical activation energies in the range of 40 to 60 kcal/mol.

- The effects of water chemistry on PWSCC are not reviewed here since they are thoroughly reviewed in the PWR Primary Water Chemistry Guidelines.

7.5.4 PWSCC Crack Growth Rates (CGRs) in PWR Reactor Coolant Environments

Tests indicate that the main factors affecting the CGR of PWSCC in Alloy 600 and its weld alloys are as follows:

- CGR in materials of different types, thicknesses, microstructures, and orientations indicate that correlations between microstructure and growth rate are not reliable, that stress relief does not reliably reduce growth rates, and that there is no effect of test specimen thickness

- CGRs for a given heat of material tend to increase with increasing amounts of cold work and as the yield strength produced by the cold work increases.

- CGR increases as stress intensity increases, typically exhibiting a power law type behavior, with a threshold stress intensity of about 8 ksi√in. (9 MPa√m). The MRP CGR disposition curve is given in MRP-55.[viii]

- Periodic unloading significantly increases CGRs. Measured CGRs tend to decrease as the hold time between periodic unloading increases.

- CGR increases as temperature increases, with an apparent activation energy of 31.0 kcal/mol.

- The water chemistry environment generally has been found to have a relatively minor effect on the CGR, for pH_T of 7.4 or less. The influence of chemistry on CGR is evaluated in the PWR Primary Water Chemistry Guidelines, and is not reviewed in detail here.

7.5.5 Fatigue Crack Initiation in Nickel-Base Alloys

Fatigue crack initiation in nickel-base alloys has not been a service induced problem in LWRs. However, fatigue analyses were required by the ASME Code and the NRC for original plant design to demonstrate high resistance to fatigue cracking. These analyses need to be updated when modifications are made to the original design, and when plant life is extended. For this reason, significant amounts of fatigue crack initiation testing have been performed. Fatigue crack initiation tests in LWR environments indicate that crack initiation in Alloys 600 and 690 and their weld metals is not sensitive to normal reactor coolant water and temperature environments, and that the ASME Code design curve can be safely used.

7.5.6 Fatigue Crack Growth Rates (CGRs) of Nickel-Base Alloys

Evaluation of fatigue crack growth rates in nickel-base has not been considered a significant issue in LWRs since fatigue-driven cracks have not been an issue in these materials. This is because crack growth has been considered as being mainly caused by SCC. Nevertheless, some data have been developed, with results as summarized below.

- Fatigue CGRs of nickel-base alloys in reactor coolant environments are increased relative to those in air.

- Fatigue CGRs, in terms of da/dN for constant ΔK_I, increase as the R ratio (ratio of minimum to maximum load) increases.

- Fatigue CGRs are increased in oxygenated water, especially under conditions where low crack growth rates per cycle are expected based on air crack growth rates (this generally involves conditions with low frequencies).

- Fatigue CGRs in weld materials tend to be significantly higher than in base materials.

7.5.7 Research Regarding Alloy X-750

Research regarding Alloy X-750 has been mainly directed at developing an understanding of its susceptibility to SCC in BWR and PWR environments, and at identifying methods to reduce its susceptibility. Some main results of this research are as follows.

- The heat treatment condition strongly affects the susceptibility of Alloy X-750 to cracking in PWR and BWR environments. Heat treatments that are suitable for applications in non-LWR applications such as in gas turbines result in high susceptibility to SCC in high-temperature water environments, e.g., the standard equalized plus aged heat treatment, material condition AH, makes the alloy highly susceptible to SCC. The mechanistic reasons for this high susceptibility are not certain, but appear to be related to the distribution and morphology of hardening phases and carbides.

- A high-temperature anneal followed by single step aging results in increased resistance to SCC. The EPRI material specification (NP-7032) provides guidance on appropriate heat treatments.

- Increasing stress level strongly decrease times to SCC, with the time decreasing approximately to the inverse fourth power of the stress. The EPRI guidelines for Alloy X-750 (NP-7338-L) indicate the levels of total stress that are considered to provide protection against SCC for plant lifetimes. These levels vary depending on whether the material will be exposed to PWR or BWR environments and on the heat treatment condition.

- Increasing temperature increases CGR, with an apparent activation energy of about 50 kcal/mol. Activation energies for crack initiation are not available.

- Susceptibility to SCC initiation increases, and CGR increases, as oxygen levels, ECP, and impurity concentrations increase. Material in the HTH condition is less susceptible and has a lower CGR than material in the AH and BH conditions.

- Material composition has some effect on susceptibility to SCC. The EPRI material specification report (NP-7032) should be consulted for details.

- Material processing steps have a strong influence on susceptibility to SCC. For example, electrodischarge machining (EDM) results in a shallow damaged surface layer and shallow intergranular penetrations that increase susceptibility to SCC. EPRI guidelines (NP-7338-L) should be consulted for guidance in this area.

- Evaluations of the resistance of Alloy X-750 to IASCC indicate that it is more susceptible in BWR environments than in PWR environment, that machining after aging improves performance, and that irradiation reduces the threshold stress for SCC and increases CGRs.

- Tests indicate that Alloy X-750 is susceptible to low temperature crack propagation (LTCP) in low temperature hydrogenated water, with susceptibility being highest for the older heat treatments such as AH and BH.

7.6 References[ix]

[i]. E. S. Hunt and D. J. Gross, PWSCC of Alloy 600 Materials in PWR Primary System Penetrations, TR-103696, July 1994.

[ii]. E. S. Hunt, et al., PWSCC of Alloy 600 Type Materials in Non-Steam Generator Tubing Applications — Survey Report Through Mid-2002; Part I. PWSCC in Components Other than CRDM/CEDM Penetrations; Part II. PWSCC in CRDM/CEDM Penetrations, in course of publication as of November 2002, EPRI.

[iii]. Improved Stress Corrosion Resistance of NiCrFe Alloys, EPRI, Palo Alto, CA: 1990. NP-6908-SD.

[iv]. W. J. Mills, et al., "Effect of Irradiation on the Stress Corrosion Cracking Behavior of Alloy X-750 and Alloy 625," Proceedings of the Sixth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, p. 633-643, TMS, 1993.

[v]. C. Benhamou, J. Daret, and J. C. Van Duysen, "Contribution of Laboratory Studies to the Understanding and Solution of Tube Guide Pin Cracking Occurrence," Proceedings of the International Symposium Fontevraud II, Contribution of Materials Investigation to the Resolution of Problems Encountered in PWR Plants, p. 30-42, SFEN, 10-14 September 1990.

[vi]. P. Dewes, et al., "Measurement of the Deformability of Austenitic Stainless Steels and Nickel-Base Alloys in Light Water Reactor Cores," Slow Strain Rate Testing for the Evaluation of Environmentally Induced Cracking: Research and Engineering Applications, p. 83-101, ASTM STP 1210, ASTM, 1993.

[vii]. Evaluation of Crack Growth in BWR Nickel Base Austenitic Alloys in RPV Internals (BWRVIP-59), EPRI TR-108710, December 1998.

[viii]. Materials Reliability Program (MRP) Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Material (MRP-55), EPRI, May 30, 2002.

8. Nickel-Base Welds and Clad

The performance of Alloy 600 and 690 type weld metals is covered in this section. Alloys 82 and 182 are the Alloy 600 type weld metals for gas tungsten arc welding (GTAW) (also known as tungsten inert gas or TIG welding) and shielded metal arc welding (SMAW), respectively, and Alloys 52 and 152 are the corresponding Alloy 690 type weld metals. These materials are used to weld Alloys 600 and 690 to themselves, to austenitic stainless steels, and to carbon and low-alloy steel parts. In addition, these alloys are used as buttering and welds on carbon and low-alloy parts, especially where these materials are joined to stainless steel piping, such as at nozzles.

8.1 Service Experience

Service experience with nickel-base alloy welds and cladding has been as follows:

- No service-related corrosion problems have been reported for Alloys 52 and 152. The initial applications of Alloy 52 and 152 weld materials started in the 1990s.

- Alloy 600 in some BWR applications has experienced intergranular stress corrosion cracking (IGSCC), such as at crevice locations in reactor vessel nozzle safe ends and core support structures. As discussed in more detail in Section 7, this IGSCC has generally occurred at welds and has been aggravated by the sensitization and residual stresses associated with welds. Alloy 182 and 82 weld metals in BWRs have experienced cracking in locations such as nozzles and core support structures. This cracking has sometimes, but not always, been associated with crevices.

- Starting in 1982, Alloy 600 in PWR penetration and nozzle applications has exhibited an increasing amount of PWSCC as PWRs have aged. While most of this cracking has been in the Alloy 600 base material (see Section 7), the stresses driving the cracking have often been the result of weld shrinkage that occurred during initial installation.

- Starting about 2000, a number of cases of PWSCC have been observed in Alloy 182 weld metal in PWRs, e.g., in reactor vessel outlet nozzles and at CRDM nozzle to reactor vessel head welds.

8.2 Material Compositions and Properties

The chemical compositions of the nickel-base weld alloys that have typically been used for LWR applications are shown in Table 8-1.

Section NB-2431.1 of Section III, Division I of the ASME Code requires that weld materials have tensile strength and ductility and impact properties that match those of either of the base materials being welded, as demonstrated by tests using the selected weld material and the same or similar base materials. Section NB-2432.2 of Section III, Division I of the ASME Code requires that the chemical composition of the welding material be in accordance with an appropriate ASME Code welding specification (in Section II.C of the Code), but leaves the choice of the specific material up to the

Table 8-1
Specified Compositions of Nickel-Base Weld Alloys Used in LWRs (wt %)

ASME/ASTM Spec. Type Grade, UNS No.	Ni	Cr	Fe	Mn	C	Cu	Si	S	P	Ti	Cb + Ta	Note
ENiCrFe-3[i], Alloy 182 Welding Electrodes for SMAW (600 type), W86182	59.0 min	13.0 – 17.0	10.0 max	5.0 - 9.5	0.10 max	0.50 max	1.0 max	0.015 max	0.03 max	1.0 max	1.0 - 2.5 (1)	(2) (3)
ENiCrFe-7[i], Alloy 152 Welding Electrodes for SMAW (690 type), W86152	Rem.	28.0 – 31.5	7.0 – 12.0	5.0 max	0.05 max	0.50 max	0.75 max	0.015 max	0.03 max	0.50 max	1.0 - 2.5	(2) (4)
ERNiCr-3[ii], Alloy 82 Bare Welding Electrodes (600 type), N06082	67.0 min	18.0 - 22.0	3.0 max	2.5 - 3.5	0.10 max	0.50 max	0.50 max	0.015 max	0.03 max	0.75 max	2.0 - 3.0 (1)	(2) (3)
ERNiCr-7[ii], Alloy 52 Bare Welding Electrodes (690 type), N06052	Rem.	28.0 – 31.5	7.0 – 11.0	1.0 max	1.0 max	0.30 max	0.50 max	0.015 max	0.02 max	1.0 max	0.10 max	(2) (5)
ERNiCr-4[ii], Alloy 72 Bare Welding Electrodes (690+ type), N06072	Rem.	42.0 – 46.0	0.5 max	0.20 max	0.01-0.1	0.50 max	0.20 max	0.015 max	0.02 max	0.3 - 1.0	--	(2)

(1) Tantalum 0.30% maximum when specified
(2) Sum of all other elements 0.50% maximum
(3) Cobalt 0.12% maximum, when specified
(4) Al 0.50% maximum, Mo 0.50% maximum
(5) Al 1.10% maximum, Mo 0.50% maximum

8.3 Main Limitations

The main limitations with regard to use of nickel-base weld alloys in reactor coolant and reactor internals applications are as follows:

- Alloy 182 in BWR reactor coolant system applications has been found to be susceptible to IGSCC in crevice areas, especially where both weld-induced sensitization and residual stresses are present. Alloy 182 has been found to be susceptible even in non-creviced areas. For these reasons, Alloy 182 is no longer selected for such applications. Alloy 690 and its weld materials, and Alloy 82, have been found to be much more resistant to IGSCC in BWR environments.

- Alloy 182 and (to a lesser extent) Alloy 82 have been found to be susceptible to PWSCC in PWR reactor coolant system applications in locations where the wetted surface has stresses over about 30 ksi (207 MPa). For this reason, these alloys are no longer selected for such applications. Alloy 690 and its weld materials have been found to be highly resistant to PWSCC in such applications.

- Tests indicate that Alloys 52, 82, 182 and 690 are susceptible to low temperature crack propagation (LTCP) as a result of hydrogen induced cracking when exposed to hydrogenated water at low temperature. Alloy 600 was not susceptible to this type of cracking. LTCP requires the presence of sharp, pre-existing cracks produced by another mechanism, e.g., PWSCC. The net result of concerns regarding LTCP in susceptible materials is that pre-existing cracks could propagate at low temperature (<300°F (150°C)) as the result of the combined presence of cracks, residual and applied stresses, and high hydrogen in the water.

8.4 Welding and Heat Treatment

Alloy 600 and Alloy 690 are routinely welded, generally using Alloys 82 and 182 for Alloy 600, and Alloys 52 and 152 for Alloy 690. There has also been some consideration of using Alloy 72 for welding Alloy 690 based on tests that indicate it provides increased resistance to corrosion.

In a 1991 paper, GE researchers reviewed the occurrence of microfissuring in nickel alloy welds, and the contribution of such microfissuring to the occurrence of IGSCC.[iii] They noted that welds in Alloy 600 are quite prone to microfissuring due to alloy and/or impurity segregation that occurs during cooling of the weld metal. Liquidation microfissuring occurs when previously solidified weld metal undergoes partial melting and separation at dendrite boundaries during reheating while depositing additional weld passes. The microfissuring ranges from very fine, short and discontinuous to large fissures that can be seen by the naked eye. The GE paper notes that selected weld compositions and welding parameters can be used to produce essentially microfissure free welds, but that it is very dependent on the skill of the welder. The paper further notes that the presence of microfissures appears to act as an initiator for the occurrence of IGSCC in welds. As illustrated by this microfissuring issue, welding of Alloys 600 and 690 can be difficult because of hot cracking and microfissuring problems, especially in situations with high restraint. This needs to be considered when designing welds and when selecting and qualifying procedures and personnel. Special Metal, a supplier of Alloy 690 type welding material, has recently announced that an improved filler metal, 52M, has been developed. It is reported to have improved resistance to cracking during fabrication.

Welded Alloy 600 parts have been found to be susceptible to IGSCC in BWR reactor coolant environments, especially when the weldment includes crevice areas. Thus, designs for new or replacement applications should avoid use of welded Alloy 600 located in crevice areas; in fact, use of Alloy 600 should be avoided and Alloy 690 and Alloy 690 type weld materials used instead.

As discussed in Section 7, welded Alloy 600 parts and Alloy 600 type weld materials in PWRs have been found to be susceptible to PWSCC at locations where areas with high weld residual stresses are exposed to high temperature reactor coolant. Weldments that were stress relieved after welding have rarely experienced PWSCC, even though the material has been sensitized by the stress relief heat treatment. Extensive testing indicates that Alloy 690 and Alloy 690 type weld metals have very high resistance to PWSCC. Thus, Alloy 690 and Alloy 690 type weld materials should be used in lieu of Alloy 600 and Alloy 600 type weld materials in PWR reactor coolant system applications.

Because of high chromium content, Alloy 690 weldments can be subjected to ASME Code type stress relief heat treatments (e.g., 10 hours at 1150°F (621°C)) with little concern regarding sensitization.

Use of Alloy 690 type welding materials, Alloys 52 and 152, is covered by ASME Code Cases 2142-1 and 2143-1. Use of these code cases apparently still requires that NRC approval be obtained since they have not as yet been accepted by the NRC in Regulatory Guide 1.85. The NRC has routinely granted approval for use of these weld materials.

8.5 Research and Development Results

The general trends regarding susceptibility to IGSCC in BWR environments and to PWSCC in PWR environments are covered in Section 7, "Wrought Nickel Alloy." Some highlights related specifically to welds are covered here.

8.5.1 IGSCC Initiation and Growth in BWR Environments

- The susceptibility ranking of the materials used in original BWRs is Alloy 182, 600 and 82, in decreasing order of susceptibility. Alloy 690 and its weld alloys, 52, 72 and 152, are much more resistant to SCC than Alloys 600, 182 and 82. The resistance of welds to SCC in BWR environments improves as their chromium concentration increases, such that materials with 30% or more chromium have low susceptibility.

- Susceptibility of Alloys 600, 182 and 82 in BWR environments is strongly increased by the presence of crevices. However, IGSCC can initiate in welds without designed-in crevices, possibly as the result of small crevices being present at weld flaws.

- In BWR environments, crack growth rates (CGRs) are highest in Alloy 182, next highest in Alloy 600, and very low and about the same in Alloys 82 and 690. CGR disposition curves have been developed by BWRVIP and are documented in the BWR Water Chemistry Guidelines and a BWRVIP report.[iv,v]

- CGRs are much higher in Alloy 182 in a direction parallel to the dendrites than in a direction perpendicular to the dendrites. This may also be true for Alloy 82, but does not appear to have been systematically investigated.

8.5.2 PWSCC Initiation and Growth in PWR Environments

- Tests of Alloys 82 and 600 indicate that Alloy 82 has about the same susceptibility as Alloy 600.[vi,vii]

- Tests of weld alloys with chromium compositions ranging from 14% to 30% showed that susceptibility to PWSCC decreased as chromium concentration increased.[viii] Alloy 182 (Cr ~ 14.5-15%) was more susceptible than Alloy 82 (Cr~18.0 -19.8%). Alloys with 30% chromium did not crack, while an alloy with chromium concentration of 21% cracked after a long time. Thus, the minimum chromium needed to ensure resistance to PWSCC of weld metals seems to be between 22 and 30%.

- Both wrought materials and weld materials have increased resistance to PWSCC initiation if given a stress relief type heat treatment, e.g., at 1100°F (593°C), independent of grain boundary carbide decoration.

- An EPRI study of several heats of Alloy 182 showed that CGR is significantly higher in Alloy 182 weld metal than in Alloy 600. The MRP is in the process of developing an appropriate CGR disposition curve for Alloy 182 in PWRs.

- Tests indicate that cracks grow 2 to 10 times faster parallel to dendrites than perpendicular to them.

8.5.3 Low-Temperature Crack Propagation (LTCP)

Tests have shown that Alloys EN82H and EN52H can have very low fracture toughness at 54°C (119°F) when tested in hydrogen containing water environments.[ix] The EN82H and EN52H alloys discussed here are essentially identical to Alloys 82 and 52 used in commercial PWRs. Very low fracture toughnesses can occur in these alloys with 15 cc/kg hydrogen in the water, with tearing modulus <10, compared to a fully ductile tearing modulus of about 300. This raises a concern that pre-existing cracks in these weld metals might propagate as a result of residual or transient stresses if there is hydrogen present in the water during some low temperature phase of a shutdown or startup operation. LTCP is characterized by rapid crack growth, in the many mils per second (mm/s) range, and can cause large crack propagations, across full test specimens. However, LTCP does not initiate cracks, and is only a concern if there is a pre-existing crack. Recent tests of Alloy 182 indicate that it behaves similarly to Alloy 82, i.e., is susceptible to LTCP.[x]

8.6 References

[i]. Specification for Nickel and Nickel-Alloy Welding Electrodes for Shielded Metal Arc Welding, ASME SFA-5.11 and AWS A5.11/5.11M-97.

[ii]. Specification for Nickel and Nickel Alloy Bare Welding Electrodes and Rods, ASME SFA-5.14 and AWS A5.14/A5.14M-97.

[iii]. K. S. Ramp and G. M. Gordon, "Fabrication and Operating History Considerations in Assessing Relative Susceptibility of BWR Components," Proceedings of the Fifth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, p365-371, ANS, 1992.

[iv]. BWR Water Chemistry Guidelines – 2000 Revision, EPRITR-103515-R2, February 2000.

[v]. Evaluation of Crack Growth in BWR Nickel Base Austenitic Alloys in RPV Internals (BWRVIP-59), EPRI TR-108710, December 1998.

[vi]. D. J. Seman, et al., "Primary Water Stress Corrosion Cracking of Alloy 600 – Effects of Processing Parameters," Proceedings: 1991 EPRI Workshop on PWSCC of Alloy 600 in PWRs, Appendix E3, EPRI TR-100852, July 1992.

[vii]. J. V. Mullen and R. J. Parrington, "Stress Corrosion of Alloy 600 Weld Metal in Primary Water," Proceedings: 1992 EPRI Workshop on PWSCC of Alloy 600 in PWRs, Appendix F3, EPRI TR-103345, December 1993.

[viii]. D. Buisine, et al., "PWSCC of Nickel Based Weld Metals with Various Chromium Contents," Proceedings: 1994 EPRI Workshop on PWSCC of Alloy 600 in PWRs, Appendix D5, EPRI TR-105406, August 1995.

[ix]. W. J. Mills, et al., "Fracture Behavior of Alloy 600, Alloy 690, EN82H Welds and EN52 Welds in Water," Proceedings: 2000 EPRI Workshop on PWSCC of Alloy 600 in PWRs (PWRMRP-27), Appendix D-11, EPRI report 1000873, November 2000.

[x]. Low-Temperature Cracking of Ni-Based Alloys and Weld Metals (MRP-108), EPRI, Palo Alto, CA: 2003. Report 1009400.