Inspection,Assessment,and Repair of Heavy Wall Reactor Vessels in High-temperature High-pressure Hydrogen Service

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Inspection,Assessment,and Repair of Heavy Wall Reactor Vessels in High-temperature High-pressure Hydrogen Service

1 Scope

This TR documents guidance for the inspection,assessment,and repair of heavy wall reactor vessels(nominally considered a wall thickness of 50 mm (2 in.)and greater)in high-pressure hydrogen service operating at temperatures below 455℃(850°F).It provides industry practices dealing with reactor vessels after construction and exposure to operating conditions.It focuses on reactor vessels fabricated from 2%Cr-1Mo,3Cr-1Mo,2/$_4$Cr-1Mo-/AV,and 3Cr-1Mo-$_4$V steels.It also offers some guidance for heavy wall reactor vessels fabricated from 1Cr-/$_2$Mo and 1/4Cr-Y$_2$Mo steels,but specifically does not pertain to C-/$_2$Mo steel vessels.However,guidance included in this document can be used for C-V$_2$Mo steel reactor vessels at the owner's discretion and with modifications as appropriate.

This document does not address damage found in hot wall catalytic reforming reactors,fluid catalytic cracking unit(FCCU)reactors,and delayed coking drums as they generally operate at higher temperatures and have thinner walls.

Since this is a technical report,it does not provide recommendations,but instead presents industry experience with case histories of repairs and recognized practices,much of which was documented as part of a Joint Industry Program (JIP)on Aging Reactor Vessels,conducted in two phases between 1995 and 2004.

2 Normative References

There are no normative references in thisdocument.

3 Terms,Definitions,Abbreviations,and Acronyms

3.1 Terms and Definitions

For this report,the following terms and definitionsapply in addition to those given inAPI Recommended Practices 934-A and 934-C.

3.1.1
cladding
Internal integrally bonded corrosion-resistant lining applied as a wrought product to the internal surface of a vessel,i.e.the vessel is constructed from clad plate.Clad plate is fabricated using a hot roll bonding process or an explosive bonding process,and is mostly used for thinner wall reactor vessels.See also lining,overlay,and loose lining.

3.1.2
disbonding
Areas of cladding or overlay that do not have a metallurgical bond to the base metal.Disbonding can occur during fabrication,and testing is typically done to detect and repair any locations.It can also occur due to high-temperature high-pressure hydrogen service depending on the characteristics of the bond areas.Testing of cladding and overlay procedures are done before fabrication to minimize the susceptibility to disbonding.

3.1.3
J-factor
An empirical relationship used to predict temperembritlement susceptibility of some low alloy steel grade base materials

$$\text{J-factor}=(Si+Mn)\times(P+Sn)\times 10{,}000$$

where compositional concentrations are expressed in weight percent.

3.1.4
lining
In this document,an internal layer of higher alloy metallic material applied toa less expensive base metal.Forms of lining include cladding,weld overlay,and loose linings.

3.1.5
loose lining
An internal layer of higher alloy metallic material applied to a less expensive base metal bywelding thin sheets of the high alloy onto the base metal.This leaves a void underthe sheets.Other names for these linings include strip lining and wallpapering.Use of a loose lining typically is not considered acceptable for high-temperature high-pressure hydrogen services.

3.1.6
weld overlay
Aform of metallic lining that is applied using a welding process and metallurgically bonded to the base metal.See also cladding,lining,and loose lining.

3.1.7
X-bar
An empirical relationship used to predict temper embritlement susceptibility of Cr-Mo alloy steel grade weld materials and 1Y$_4$Cr-/$_2$Mo base metals

$$\text{X-bar factor} = (10\times P+5\times Sb+4\times Sn+As)/100$$

where compositional concentrations are expressed in weight parts per million(wppm).

3.2 Abbreviations and Acronyms

AUT	automated ultrasonic testing
CUI	corrosion under insulation
DHT	dehydrogenation heat treatment
FCCU	fluid catalytic cracking unit
FFS	fitness-for-service
FMC	full matrix capture
FMR	field metallographic replication
GTAW	gas tungsten arc welding
HAZ	heat-affected zone
HBW	Brinell hardness with tungsten carbide indenter
HTHA	high-temperature hydrogen attack
ISR	intermediate stress relief
ITP	inspection and test plan
LED	light-emitting diode
LMP	Larson-Miller parameter
MDMT	minimum design metal temperature
MT	magnetic particle testing
NDE	nondestructive examination

PAUT	phased array ultrasonic testing
PQR	procedure qualification record
PT	penetrant testing
PWHT	postweld heat treatment
RCA	root cause analysis
RHC	reheat cracking
RT	radiographic testing
RTJ	ring-type joint
SAW	submerged arc welding
SCC	stress corrosion cracking
SMAW	shielded metal arc welding
SWUT	shear-wave ultrasonic testing
TFM	total focusing method
TOFD	time-of-flight diffraction(a UT technique)
UT	ultrasonic testing
WPS	welding procedure specification

4 Design,Materials,and Fabrication History

4.1 Range of Operating Conditions

The hydroprocessing reactors addressed in this report are fabricated from 2/$_4$Cr-1Mo,3Cr-1Mo,2/4Cr-1Mo-4V,or 3Cr-1Mo-/$_4$V steel alloys.They generally operate at temperatures up to 455℃(850°F)with operating pressure up to 20.7 MPag(3000 psig)and with a hydrogen partial pressure up to 90%of the total pressure.See Section 1 for further discussion of the scope of this document,especially on applying it to heavy wall reactors fabricated from C-Y$_2$Mo,1Cr-/$_2$Mo,and 1/$_4$Cr-/$_2$Mo.

4.2 Material Development History

In the early 1960s,most hydroprocessing reactors were made from annealed 2$_4$Cr-1Mo steel with a minimum specified tensile strength of 415 MPa(60 ksi).Hydroprocessing advancements then led to the need for greater catalyst volume and larger diameter vessels,which required increased reactor wall thickness.To minimize wall thickness,higher strength quenched and tempered 2/$_4$Cr-1Mo steel was developed for reactor vessels.

The first hydrocracking reactors fabricated from quenched and tempered 2/$_4$Cr-1Mo steel,which had a minimum specified tensile strength of 825 MPa(120 ksi),went on stream in 1966 in the USA.After several months of operation,hydrogen embrittlement cracking was detected in several of these reactors at the same site.Failure investigation of this incident concluded that limiting the tensile strength to~755 MPa(110 ksi)maximum would prevent hydrogen embritlement cracking.After this incident,and considering a safety margin,690 MPa(100 ksi) maximum specified tensile strength became the industry practice for quenched and tempered 2Y4CF-1Mo steel used in heavy wall reactor fabrication.1]

In the early 1970s,Japanese fabricators manufactured heavy wall reactors from forged rings that were heated to an austenitizing temperature,quenched in a bath of agitated water,and tempered to the desired strength level.During the same period,formulas limiting the steel impurity levels,namely the J-factorfor base metal and the X-bar for weld metal,were proposed as a means of controlling the susceptibility of Cr-Mo steels to temper embrittlement.At that time,steel-making practices would typically produce Cr-Mo steels with a J-factor of up to 300 for plate and forgings and an X-bar of up to 25 for weld deposits.Due to improvements in the steel-making process during the 1980-1990 timeframe,the J-factor and X-bar are now consistently reduced to less than 100 and 15,respectively.

In the early 1980s,a program was initiated to develop a new grade of 2$_4$Cr-1Mo steel for reactor vessels, resulting in an enhanced grade with a specified tensile strength range of 585 to 755 MPa(85 to 110 ksi)with a maximum design metal temperature of 455℃(850°F).Properties were enhanced by controlling tempering and postweld heat treatment(PWHT)to a narrower temperature range.Thisenhanced grade of 2/$_4$Cr-1Mo steel was first accepted by the American Society of Mechanical Engineers(ASME)in 1984 in Code Case 1960 and later in 1987 by British Standard BS 5500 in BS Enquiry Case 5500/73.To maintain the enhanced tensile strength of this grade of 2/$_4$Cr-1Mo steel,the minimum PWHT temperature was decreased from 675℃(1250°F)to 650℃(1200°F).There are some reactor vessels made of thermally enhanced 2/$_4$Cr-1Mo installed in Europe and Asia.

Cr-Mo steel strength levels were further enhanced by adding nominally 0.25%vanadium.In 1984,ASME Code Case 1961 was approved.This allowed 3Cr-1Mo-/$_4$V-Ti-B steel to be used at metal design temperatures up to 480℃(900 °F).The first reactors made of 3Cr-1Mo-/$_4$V-Ti-B were installed in Canada in 1990.A similar material,3Cr-1Mo-/4V-Cb-Ca,was approved as ASME Code Case 2151in 1993.The first reactors made of 3Cr-1Mo-%4V-Cb-Ca were installed in South America in 1995.

In 1991,ASME Code Case2098 was approved,permitting 2/$_4$Cr-1Mo-/$_4$V to beused at metal design temperatures up to 480℃(900°F)for Division 1 and 454℃(850°F)for Division 2.However,up to 1997,the published API Recommended Practice 941 temperature to avoid high-temperature hydrogen attack(HTHA)in the new 2$_4$Cr-1Mo-/4V alloy was represented by the 2/$_4$Cr-1Mo curve,limiting service temperatures for 2$_4$Cr-1Mo-/4V to 454℃(850°F)maximum.Then,in 1997,a separate curve was added for 2/4Cr-1Mo-/$_4$V,matching the 3Cr-1Mo curve that allowed exposure up to 510℃(950°F)at high hydrogen partial pressures.Most refiners apply a 28℃(50 °F)margin to the Nelson Curve,which results in a 482℃(900°F)limit for 2/4Cr-1Mo-/$_4$V.The first reactors made of 2V$_4$Cr-1Mo-/$_4$V were installed in the USAin 1998.Since then,2$_4$Cr-1Mo-/$_4$V steelhas become the most prevalent material choice for heavy wall hydroprocessing reactors.

4.3 History of Design Stresses from Various Design Codes

Before 1968,ASME BPVC,Section VII(currently ASME BPVC,Section VIII,Division 1),ASME BPVC,Section Ⅲ,other national pressure vessel design codes,and design criteria customized by individual companies were used to design hydroprocessing reactors.When using the design rules of ASME BPVC,Section VII,Division 1,the design stress was set at the lesser of 1/4 of the minimum specified tensile strength or 5/8 ofthe minimum specified yield strength,resulting in a very thick reactor wall.In 1968,ASME published BPVC,Section VII, Division 2,which provides a higher design allowable stress based on the lesser of 1/3 of the minimum specified tensile strength or 2/3 of the minimum specified yield strength.This resulted in a lower design wall thickness for reactor vessels.

In 2007,ASME BPVC,Section VII,Division 2 was rewritten,changing the design stress basis to permit higher allowable stresses based on the lesser of 1/2.4(42%)of the minimum specified tensile strength or 2/3 of the minimum specified yield strength.However,the maximum temperature for determining a Division 2 fatigue evaluation was limited to 371℃(700 °F).In October 2008,Code Case 2605 raised the fatigue evaluation temperature to 454℃(850°F)for vanadium modified 2.25Cr materials.In July 2019,Code Case 2605-4 further raised the fatigue evaluation temperature to 482℃(900 °F).Table 1 shows the increase in the design allowable stress at 454℃(850°F)provided by the revision in2007.(Note that allowable stresses based on time-dependent creep properties added to the allowable stress tables are shown in italics in ASME BPVC,Section I Ⅱ, Part D).

Table 1—ASME BPVC,Section VIII,Division 2 Change in Design Stress at 454℃(850°F)

Material	2004 Edition,ksi (MPa)	2007 Edition,ksi (MPa)	Increment
Conventional 2/4Cr-1Mo	21.9(151)	21.9(151)	0%
2/Cr-1Mo-/4V	24.5(169)	28.9(199)	18%
3Cr-1Mo-%4V	23.8(164)	25.8(178)	8%

4.4 API Publications on Reactor Materials and Fabrications

Priorto 2000,heavy wall Cr-Mo reactorvessels had unique requirements concerning material,welding,fabrication, inspection,and examination that were prepared by owners,process licensors,and/or engineering contractors.In 2000,API publishedAPI934(later superseded byAPI934-A),which consolidated basic consensus requirements for fabrication of heavy wall 2V$_4$Cr-1Mo reactors.In the second edition,issued in 2008,the various V-modified versions of Cr-Mo steels were added.

In early 2008,reheat cracking occurred in 2/4Cr-1Mo-4V submerged arc weld (SAW)deposits during reactor fabrication at several fabricators in Europe.2 After several investigative studies,Annex A,entitled Guidance for *Inspection for Transverse Reheat Cracking,was included to provide nondestructive evaluation(NDE)procedures* for detecting transverse reheat cracking.These methods include pulse echo ultrasonic testing(UT)and time-of-flight difraction(TOFD)UT on shell seams,and phased array UT(PAUT)on nozzle welds.Additionally, *Annex-B,entitled Weld Metal/Flux Screening Test for Reheat Cracking Susceptibility,provides testing methods* for evaluating the sensitivity of SAW welding consumables to reheat cracking.

In 2011,API published API Technical Report 934-B,Fabrication Considerations for Vanadium-Modified Cr-*Mo Steel Heavy Wall Pressure Vessels,which is a technical report providing explanations,experiences,and* knowledge gained from actual problems that have occurred during the fabrication of V-modified Cr-Mo steel heavy wall reactors.

API also addresses the fabrication of 1V$_4$Cr-Y$_2$Mo and 1Cr-Y$_2$Mo heavy wall reactor vessels operating at temperatures at or below 441℃(825°F)in API Recommended Practice 934-C.This document was originally published in 2008 and contains recommended practices for the fabrication of 1/$_4$Cr-/$_2$Mo and 1Cr-/$_2$Mo vessels in the quenched and tempered(Q&T)condition in nominal wall thicknesses between 2 in.and 4 in.In 2010,API published API Technical Report 934-D,which is a companion document to API 934-C with explanations and examples involving the fabrication of 1$_4$Cr-Y$_2$Mo and 1Cr-Y$_2$Mo vessels.

API has also published reports on temper embrittlement of Cr-Mo steels.This includes API Advisory Letter on In-*Service Embrittlement of Low-Aloy CF-Mo Steels,dated July 30,1974,and API Publication 959,Characterization Study of Temper Embrittlement of Chromium-Molybdenum Steels,issued in 1982.*

4.5 Summary of Historical Changes in Reactor Fabrication

Table 2 summarizes historical changes in the design code,materials of construction,applicable API publications, temper embritlement parameters,low-temperature toughness requirements,and weight and wall thicknesses records for conventional and V-modified Cr-Mo steel hydroprocessing reactors.

Table 2—Historical Changes in Reactor Materials and Fabrication[3

Type of Reactor Material and Fabrication	Year 64	66	68	70	72	74	76		78	80	82	84	86	88	90	92	94	96	98	00	02	04	06	08	10	12	14	16
Reactor Generationsa		1st-1965~1972					20d-1973~1980				3rd-1981~1987				4-1988~1997					5-1998~current								
Codes and Standards																												
ASME BPVC, Section VIII Code	Div.			Div.2(first issued in 1968)																					Div.2 Rewrite			
Design Stress b	Min.S.1/4 or S 5/8			Min.S.1/3 or S2/3																					Min.S.1/2.4 or S2/3			
API 934 milestones																				A				B	C,D	E		F
Materials for New Fabrication:d																												
Q&T 2/4Cr-1Mo and 3Cr-1Mo		Standard 214Cr-1Mo and 3Cr-1Mo																										
Thermally Enhanced 2/4Cr-1Moe														Enhanced 2/4Cr-1Mo														
3Cr-1Mo-/4V-Ti-B'															3Cr-1Mo-4V-Ti-B													
3Cr-1Mo-4V-Cb-Ca9																												
2/4Cr-1Mo-4Vh																		2/4Cr-1Mo -4V										
Material Properties:																												
J-factor(achievable max.limit)	Not Specified					300	250			180		150		130	100													
Low Temp.Toughness(℃)'	10						0		-7	-15		-20	-30															
Max.Weight (ton)		500	815			850					1,150					1,450							1,900					
Max.Wall Thickness(mm)d		211		297									344					351										

amon oTAr1ss4A(2UorC=rsteamon orAr1s54A mexn,Guoanceiornspecnon ior ransverse kenear uracring(cUTUrU=rsteamon orAr1ss45,rapncanon consieranons forVanadium-Modifed CF-Mo Stel heavy walPresure Vessels(2011);E=FirstedionofAP1934AAmex B,weld MetaVFux Screning TestforReheat Cracking Susceptoly 2012) F=Third edition of API 934-A,which encompasses AnnexAand Annex B(2019).

Code Case 2151 initially approved in 1993.
Code Case 2098 initially approved in 1991
Lowest temperature at which toughness of 55J(40 f-lbs)average/45J(33 ft-bs)minimum can be consistently achieved.

5 Potential Damage Mechanisms in Heavy Wall Reactor Vessels

5.1 General

There are several in-service damage mechanisms that can affect the properties or condition of Cr-Mo steels used in the fabrication of heavy wall reactor vessels.Each of these mechanisms can lead to cracking and other forms of degradation that can adversely affect the reliability of heavy wall reactor vessels.Additional information on damage mechanisms found in refinery equipment is given in API 571.

5.2 Temper Embrittlement

Temper embritlement is primarily observed in older reactors.Newer Cr-Mo reactor steels(since the 1990s) have lower concentrations of impurity elements (e.g.S,P,As,and Sn),and,thus,are more resistant to temper embritlement.It is the result of impurity elements,such as P,Sn,Sb,and As,that diffuse to grain boundaries during elevated temperature service.

Long-term exposure to service temperatures ranging from 345℃(650°F)to 595℃(1100°F)promotesdifusion of impurity elements to grain boundaries,resulting in a shift of the ductile-to-britle transition temperature(DBTT) to higher temperatures.Temper embrittlement typically is determined by Charpy V-notch impact testing.Long-term isothermal aging of Cr-Mo steel samples at elevated temperatures indicate that temper embrittlement reaches a maximum level after 20,000 to 30,000 hours of aging at temperature.

Temper embrittlement can promote brittle fracturewhen the susceptible vessels are pressurized at temperatures below 150℃(300°F).Figure 1 shows atypicalCharpy impactenergy transition curve fora Cr-Mo steel susceptible to temper embrittlement.In the as-fabricated condition,this Cr-Mo steel exhibits a low DBTT temperature typical for steels with good resistance to brittle fracture;however,after long-term elevated temperature service,the steel displays a high transition temperature and increased susceptibility to brittle fracture.

Today's Cr-Mo reactor steels are highlyresistantto temper embritlement with improved compositional controls on both the weld metal and base metal,as well as special testing requirements for the weld metal that demonstrate greatly reduced effects due to temper embrittlement.The J-factor is used to indicate the effect of impurity levels on temper embrittlement in base metals of 2/$_4$Cr-1Mo,3Cr-1Mo,and the V-modified versions of these steels.The X-bar is used to indicate the effect in allthe Cr-Mo weld metals covered by this document and in 1/$_4$Cr-Y$_2$Mo and 1Cr-/$_2$Mo base metal.X-bar is used for 1/$_4$Cr-Y$_2$Mo and 1Cr-Y$_2$Mo base metal because they contain intentional additions of Si(nominally 0.50 to 1.00 wt.%),which results in the J-factor not being an accurate indicator of temper embrittlement.See Section 2 for definitions of J-factor and X-bar.

Theoretically,the effects of temper embrittlement can be reversed by a high-temperature heat treatment followed by rapid cooling.A de-embrittling heat treatment involves holding at 600-625℃(1100-1150°F)for two hours minimum,followed by water quenching or fast cooling as outined in API 571 and API Publication 959.However, there are no reported cases where a de-embrittling heat treatment has been performed prior to weld repairs on Cr-Mo equipment affected by temper embritlement.This is likely to be because it is impractical to achieve the required fast cooling/water quenching from the de-embritting heat treatment temperature on a reactor without generating large thermal stresses in the area being cooled.As a result,other steps to minimize the risks from embrittlement have been implemented during repairs,such as bake-outs,preheat,and more,as discussed in other sections of this API 934-H document.

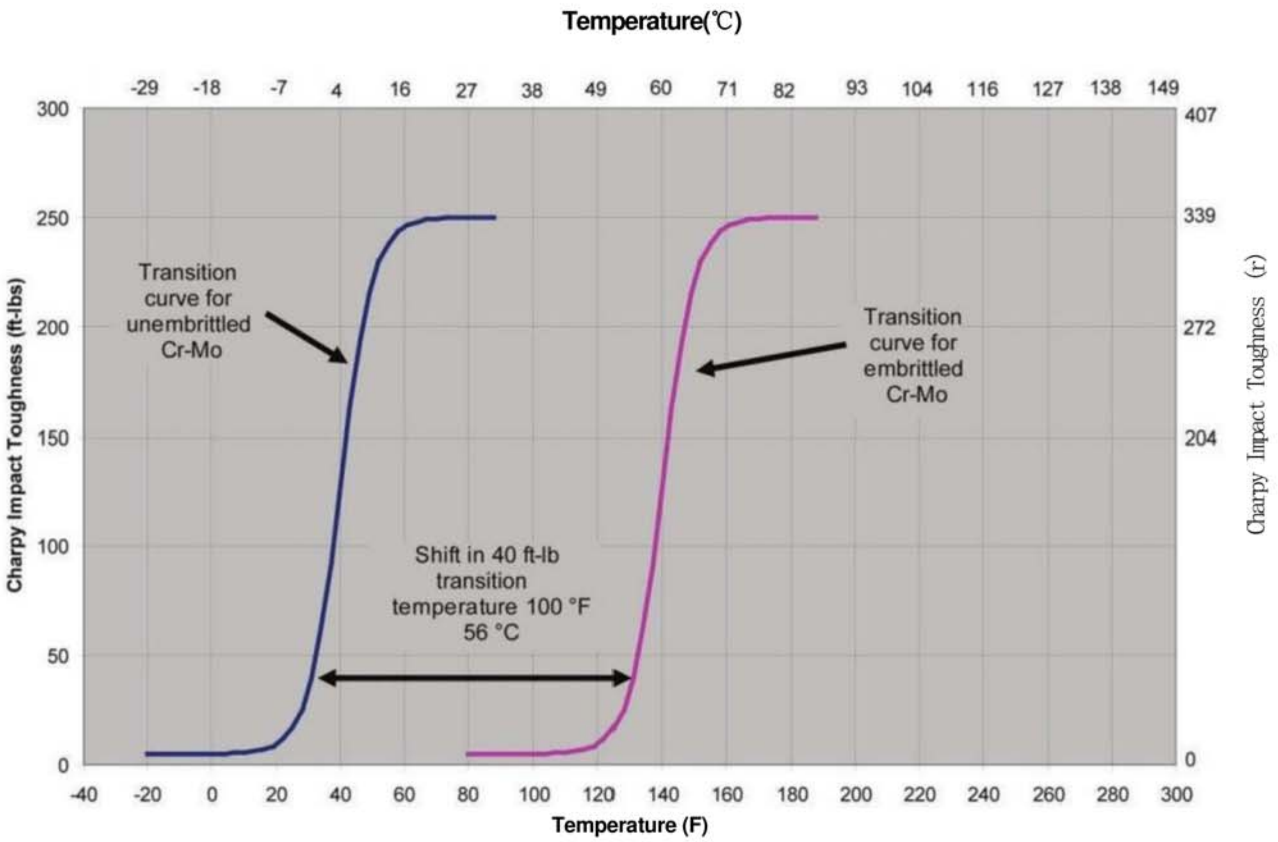


Figure 1-Temper Embrittlement Results in an Upward Shift of the Transition Temperature for Brittle Fracture

5.3 Hydrogen Embrittlement

Hydrogen embrittlement occurs when hydrogen atoms diffuse into susceptible steels.This difusion occurs while in service at high temperature,and once equilibrium is achieved,the hydrogen concentration in the steel is at the solubility limit.When the reactor is cooled,some of the atomic hydrogen gas diffuses out of the steel. However,the solubility limits varywith temperature,and while the reactor is being cooled for shutdowns,in many cases the hydrogen cannot diffuse out quickly enough and will xceed the solubility limits at lower temperatures. Hence,the hydrogen is trapped in the steel,which can cause temporary embritlement.The atomic hydrogen that remains in the steel promotes slow stable crack growth at low stress intensity levels-relatively small cracks propagating at low applied stress levels.This form of slow stable crack growth is commonly called hydrogen embrittlement cracking.

Hydrogen embrittlement can occur in Cr-Mo reactor steels at dissolved hydrogen levels normally above 1.5 ppm, at which level hydrogen embrittlement is typically evaluated.Research has shown that slow stable crack growth resulting from hydrogen embrittlement occurs below a threshold temperature that is determined by the hydrogen concentration and impurity levels in the steel.The higher the hydrogen concentration and impurity level in the steel,the higher the threshold temperature for slow stable crack growth.Testing has shown that Cr-Mo steels with a high impurity level and high dissolved hydrogen concentration can have a threshold temperature for crack growth as high as 130℃(270°F).These high threshold temperatures for hydrogen embrittlement are based on compact tension(CT)testing conducted at the University of Virginia on hydrogen charged plate samples with a J-factor of 302 and SAW deposit samples with an X-bar of 25.7 Recent testing on V-modified 2Y4Cr-1Mo steel has shown that it has an enhanced resistance to hydrogen embrittlement,largely due to its high solubility for hydrogen.This enhanced hydrogen solubility has been attributed to the hydrogen trapping sites provided by the presence of vanadium carbides in V-modified $2/_4$Cr-1Mo steel.4-8

The most common effect of hydrogen embrittlement in heavy wall vessels is cracking and disbondingin the Cr-Mo base metal immediately below SS weld overlay or cladding.Disbonding cracks occur most frequently in older vessels fabricated from Cr-Mo steel where a weld overlay was deposited using a high heat input welding process such as electroslag welding(ESW).The high heat input welding process generates large grains in the heat-affected zone(HAZ)of the Cr-Mo base metal immediately below the weld overlay.These large grains in the Cr-Mo base metal are particularly susceptible to hydrogen embrittlement and the resulting slow stable crack growth.Experience has shown that disbonding cracks propagate parallel to the surface and do not result in through-wall cracking.API 934-A outines possible testing(per ASTM G146)of Cr-Mo samples that have been weld overlayed to ensure that disbonding will not occur with the proposed weld overlay procedure.The testing is done at the hydrogen partial pressures and temperatures that simulate similar hydrogen charging as the service conditions.The V-grade steels have proven to be more resistant to disbonding,both by testing and experience, than conventional Cr-Mo grades of steel.Hence,disbonding tests are typically not required for V-grade steels. SeeAPI 934-A.

Limited disbonding tests have been conducted on explosively bonded cladding,showing favorable resistance when compared with weld overlay or roll bond applications.Both explosive bonded and hot roll bonded claddings have shown successful experience with no disbondment at typical operating conditions.For each application, testing of the clad materials for disbondment resistance is generally performed perAPI934-AorAPI934-C when justified based on the operating conditions.31,32

5.4 Sigma Embrittlement and Cracking of 300 Series Stainless Steel Overlays

Most heavy wall vessels contain a layer of 300 series stainless steel(SS)weld overlay or cladding on the inner diameter(ID)surface to provide resistance to sulfidation corrosion.In thinner wall reactor vessels,hot roll bonded or explosive bonded clad plate is often used to provide the internal lining.However,weld overlay is used on thicker vessels,and the SS weld deposits can experience sigma embritlement and cracking from exposures to high temperature.In the as-welded condition,the weld overlay will contain a level of delta ferrite (typically between 3 and 10 vol.%)that transforms to sigma phase during PWHT.In addition,this layer of 300 series SS weld overlay is left in residual tension following PWHT of the vessel during fabrication due to the difference in thermal coefficient of expansion between the lining and Cr-Mo base metal,and it then experiences additional cyclic thermal stresses each time the vessel goesthrough a startup and shutdown cycle.These tensile thermal stresses in the weld overlay are very high and exceed its yield strength level when the reactor is cooled to ambient temperature.9 This can result in low cycle fatigue cracking in the weld overlay after multiple startup and shutdown cycles,especially in areas where the thermal stresses can be concentrated,such as at the toes of internal attachment welds.This low cycle thermal fatigue effect is further discussed in 5.9.

In older vessels,SS weld overlays typically display a greater tendency to crack compared with a weld overlay applied today.This is because there were fewer compositional and impurity control requirements for the overlay welding consumables in older vessels,which resulted in high levels of sigma phase and/or other phases that promoted poor ductility in the overlay after the vessel was subjected to PWHT.I[9] .10 In the older versions of 300 series SS weld overlay,compositions favored higher delta ferite levels in the deposit to prevent hot cracking tendencies.This was particularly the case for older versions of Type 347 SS weld overlay,which has a stronger tendency for hot cracking in the weld deposit than other grades of 300 series stainless steel.In general,delta ferrite levels above 8%can result in significant reductions in fracture ductility due to sigma phase formation during PWHT.This low level of fracture ductility can result in cracking of the weld overlay after a relatively few startup and shutdown cycles.

In addition to an inherently poor resistance to cracking from thermal cycles,laboratory testing summarized in Welding Research Council(WRC)Bulletin 240 has shown that a 300 series SS weld overlay deposit with high levels of sigma phase can also display a reduced ductity due to hydrogen charged into the overlay during service

Today,the ferrite content and compositionof SS weld overlays are controlled to minimize these potential problems. Additionally,in cases where the weld overlay is applied in two layers,the second layer typically is applied after PWHT so that the second layer does not have the delta ferrite in the weld deposit converted to sigma phase.

5.5 Polythionic Acid Stress Corrosion Cracking of SS Items

In general,300 series SS exposed to temperatures above 370℃(700°F)forstraight grade Type 304 and Type 316 can become sensitized and susceptible to polythionic acid stress corrosion cracking(PASCC).These straight grades of SS also sensitize during welding.Low carbon grades of 300 series SS are resistant to sensitization from welding and typically do not sensitize in service at temperatures below ~400℃(750 °F).Chemically stabilized grades of SS such as Types 321 and 347 stainless steels are resistant to sensitization from welding and typically do not sensitize in service at temperatures below~454℃(850°F).The actual temperature above which sensitization will occur depends on the specific grade of 300 series SS and whether it is welded.Sensitization can also occur when some grades of SS are exposed to the PWHT temperatures of low alloy steels,but the low carbon and stabilized grades are generally resistant based on the relatively short times of exposure.

PASCC occurs during downtimes when an iron sulfide scale on the surface of sensitized SS is exposed to water and air.In reactor vessels,the tensile stresses that drive PASCC are usually associated with welding residual stresses or the residual stresses resulting from thermal cycling.

It has been shown that weld overlay linings are very resistant to PASCC.11 Reportedly,weld overlay displays a high resistance to sensitization due to the fine dendrite cell size produced in the overlay deposit.121 Hence,the primary concerns of PASCC are on SS cladding and internals.As a result,it is important to use grades of 300 series SS for wrought product cladding and internals that have an inherently enhanced resistance to sensitization both during fabrication and service,such as Types 304L,316L,321,and 347 stainless steels.

Severalowners report that they circulate or spray asoda ash solution in hydroprocessing reactors and sometimes additional components when cooled for shutdowns,as outlined in NACE SP0170,to prevent PASCC.This is reportedly to protect SS internals or cladding(when predicted to be sensitized),or otherSS piping,exchangers, etc.As noted in 5.6,remnant water from the circulation of a soda ash solution has caused chloride stress corrosion cracking (SCC)or alkaline (caustic)SCC of 300 series SS components in the unit,such as low point piping drains,shortly after startup.This has resulted in significant unplanned outages.As a result,many owners use other practices during a shutdown,as outlined in NACE SP0170,that do not involve the circulation or spray of a soda ash solution.However,if none of the SS components are predictedto be sensitized,then many owners do not require any of these practices.

5.6 Chloride Stress Corrosion Cracking of SS Items

Chloride SCC can occur in a 300 series SS if exposed to water and chloride ions above ~60℃(140°F).As discussed in 5.5,the source of the chloride ion in some reports of chloride SCC in hydroprocessing reactor circuits has been attributed to the soda ash solution used to neutralize the SS surface during downtimes to prevent PASCC.The SCC is believed to have occurred when the soda ash solution was not drained thoroughly, and pockets of solution became trapped in drains,low points,or dead legs.Once the unit is restarted,the soda ash solution heats up,and in some cases,boils at the high operating temperature and pressure.This results in conditions conducive to chloride SCC,even though the water and soda ash contained chloride ion levels that met the limits allowed in NACE SP0170.

Chloride SCC also has been identified as a contributing factor causing ring-type joint(RTJ)flange cracking as discussed in 6.8 and has been observed in SS vertical bed thermowells initiating from the inside surface.The latter has been attributed to rainwater getting into the thermowell during a shutdown.The cracking has typically occurred shortly after startup when the water heats up.Preventing water entry is the primary means of avoiding this cracking.Some older reactors have solid SS catalyst dump nozzles that have also suffered chloride SCC.In newer reactors,these nozzles are Cr-Mo overlayed with SS so that chloride SCC occurring in the overlay does not propagate through the underlying Cr-Mo.

5.7 High-temperature Hydrogen Attack

High-temperature hydrogen attack(HTHA)occurs when dissolved hydrogen reacts with carbides in the steel to form internal methane that promotes fissuring and cracking.The threshold conditions for HTHA of various materials and other information about the HTHA mechanism,detection,etc.are given in API 941.HTHA has not been observed in heavy wall vessels fabricated from Cr-Mo steels because these steels form carbides that contain adequate alloy to prevent the reaction with hydrogen that forms methane at the hydroprocessing operating conditions.There are a few incidents of HTHA in heavy wall C-/$_2$Mo reactors reported in API941.

It is recognized that a 300 series SS cladding or overlay integrally bonded to a heavy wall feritic steel reactor provides a reduction in the hydrogen content of the base metal during operation,and hence a reduction in the risk of HTHA.Due to the increased hydrogen solubility and reduced hydrogen diffusion rates in 300 series SS,the stainless steel acts as a barrier that essentially reduces the hydrogen partial pressure for the underlying feritic steel.API941 contains guidance on how to determine the lowering of hydrogen levels in base metals provided by an integrally bonded 300 series SS cladding or overlay in a heavy wall ferritic steel reactor.Some older vessels and exchangers used loose SS liners in nozzles that were welded at both ends of the cylinder,and these liners do not provide any benefit in reducing the risk of HTHA of the base metal.

5.8 High-temperature Sulfidation

Heavy wall reactor vessels containing high levels of H_2S are lined with a 300 series SS cladding or weld overlay to resist sulfidation corrosion.At maximum expected operating conditions of temperature and H_2S levels,the Couper-Gorman curves in API Recommended Practice 939-C indicate that a corrosion rate of well below 0.05mm/yr(2mpy)is expected on stainless steel.However,even at this very lowcorrosion rate,enough sulfide scale can accumulate in the catalyst bed and on the outlet screens to cause a high pressure drop across the reactor.Several owners have aluminized screens in the reactors to minimize scale formation and the resulting pressure drop.

Some hydrocracking processes have minimal sulfur in the feed and use Cr-Mo reactors with no internal linings. No significant sulfidation has been reported,but these reactors cannot be changed to a service in which the feed contains sulfur.Some reactor vessels were lined with 12Cr SS cladding (such as grades 405 or 410S).These grades display reduced sulfidation resistance compared with 300 series SS cladding,and the corrosion rates can be difficult to monitor.

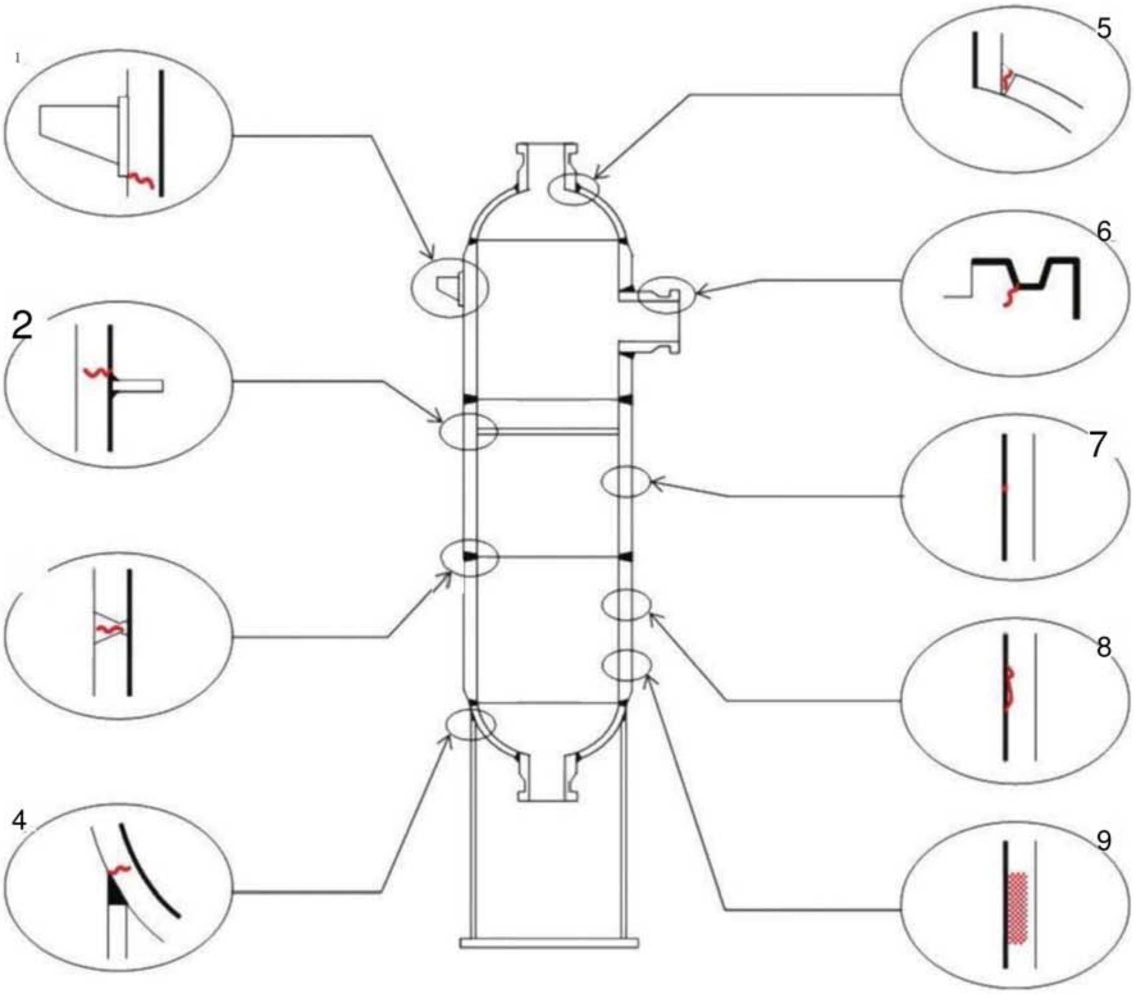
5.9 Thermal Fatigue Cracking of 300 Series Stainless Steel Weld Overlay

When a Cr-Mo reactor vessel with 300 series SS weld overlay cladding is heated up during operation and during a final PWHT,the 300 series SS expands more than the underlying Cr-Mo steel,resulting in thermal fatigue cracking.This results in stresses in the 300 series SS cladding that exceed the yield strength,resulting in residual tensile stress in the cladding when the reactor cools down.9 Thermal fatigue cracking typically occurs first at internal attachment welds that act as a stress concentration,as further discussed in 6.3.Experience also shows that this cracking is most pronounced in older Type 347 SS weld overlay deposits,which tend to have higher sigma phase levels.The effect of sigma phase level on cracking of SS overlay is further discussed in 5.4.9 10.121 In addition to sigma phase formation,it has been observed that Type 347 SS weld overlay with high levels of carbon and niobium can result in a very hard overlay deposit with limited fracture ductility and an increased susceptibility to cracking.

6 Flaws Found in Heavy Wall Reactor Vessels

6.1 General

There are several types of flaws that can occur in heavy wall reactor vessels,mostly occurring in older generation vessels.These are illustrated in Figure 2(published in WRC 566)and further described in 6.2 to 6.12.These flaws can result from the combined effects of the damage mechanisms described in Section 5.



1.Cracks at external attachmentwelds (See 6.2)
2.Cracks at internal attachment welds (See 6.3)
3.Cracks at longitudinal and circumferential weld seams(See 6.4)
4.Cracks atskirt attachment welds (See 6.5)
5.Cracks at nozzle attachment welds (See 6.6)
6.Cracks at ring joint grooves (See 6.8)
7.Cracks in weld overlay cladding (See 6.9)
8.Cracks at the interface between the cladding and base metal(See 6.10)
9.Damage in the Cr-Mo base metal(See 6.12)

Figure 2—Areas Where Damage May Occur in Heavy Wall ReactorVessels

6.2 Cracks in External Attachment Welds

Some reactors may have external attachments welded directly to the shell.Cracks may be caused by thermal stresses within or close to the external attachment welds.Thermal stresses at external attachments typically are caused by poor application of insulation in the area around the attachment.If the attachment is not fully insulated or insulation covered,it can act as a cooling fin,causing a high temperature gradient at the attachment, particularly during a heavy rain or snow.This can result in high thermal stresses.It is important that insulation and metal jacketing is properly reapplied after inspection of attachments is performed.As part of a thorough root cause analysis(RCA)in the event of crack-like flaw discovery,detailed fatigue evaluations based on heat transfer and thermal stress analyses have been employed to understand the impact of local cooling on the potential for fatigue-induced cracking.

6.3 Cracks at Internal Attachment Welds

Cracks are commonly found at 300 series SS internal attachment welds,such as to catalyst bed support rings, beam support pedestals,or outlet collectors.Figure 3 shows typical cracking in the weld overlay associated with an internal support ring.These surface-initiating cracks frequently propagate through the SS weld overlay,but seldom propagate deeply into the underlying base metal.A300 series SS weld overlay is susceptible to cracking due to the difference in the coefficient of thermal expansion between the 300 series SS and underlying Cr-Mo steel as discussed in 5.4 and 5.9.9.10.12 During each startup and shutdown cycle,300 series SS will experience very high stresses,typically well above the yield stress.These stresses will be tensile once the reactor has cooled from a high operating temperature.Cracking occurs at corners on internal support rings,pads,collectors, and other attachments because ofthe stress concentration that exists at these locations.

Type 347 SS weld overlay is commonly used for at least the outer layer of weld overlay in heavy wall reactors. Experience has shown that the Type 347 overlay in many olderreactors is more susceptible to cracking because of a higher level of delta ferite in the microstructure.DuringPWHT,delta ferrite transforms to sigma phase,which reduces overlay ductility.The chemical composition of the Type 347 weld consumables commonly used in older reactor vessels promotes higher levels of delta ferite in the overlay.While this is beneficial for resistance to hot cracking during welding,the resulting weld overlay and attachment welds are more susceptible to cracking from the repeated thermal stresses associated with startup and shutdown cycles.

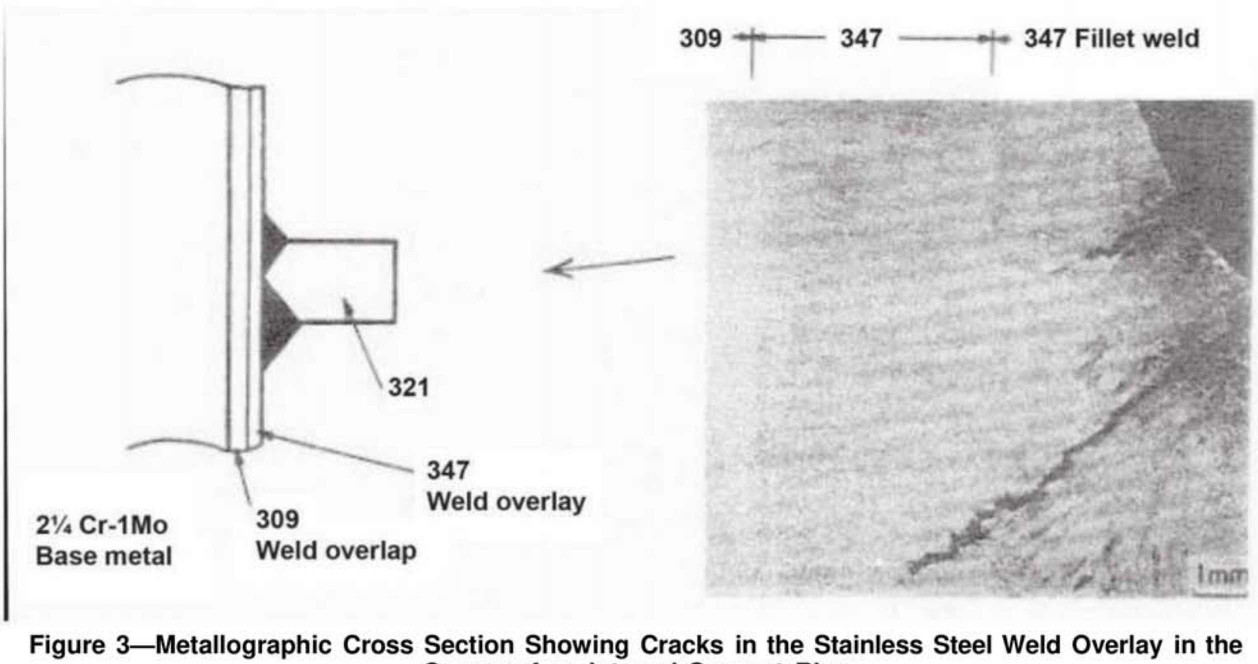


Figure 3—Metallographic Cross Section Showing Cracks in the Stainless Steel Weld Overlay in the Corner of an Internal Support Ring

6.4 Cracks in Longitudinal/Circumferential Weld Seams

Weld flaws,such as small hot cracks or lack of sidewallfusion,can form during fabrication.These types of weld flaws are normally embedded beneath external surfaces and are more common in older reactor vessels.These older reactors were inspected by radiographic testing(RT),which cannot routinely detect these types of planar flaws.Starting in the late 1980s,ultrasonic testing(UT)techniques have been used to more effectively inspect welds in reactor vessels.UT techniques have a greater abiity to detect embedded planar flaws when compared with inspection using RT.As a result,welds in newer reactors inspected using UT techniques during fabrication rarely display the embedded planar flaws that were commonly found in older reactors.Many older reactors,in which the welds were originally inspected using RT,were later reinspected using UT techniques to examine the welds.These reinspection efforts using UT technology found manyembedded flaws in these older reactors.

Figure 4 shows an example of an embedded flaw that was found at a weld in a reactor that was retired from service.In this case,the metallographic examination does display indications that this crack may have grown during each operating cycle.This was the conclusion reached by the investigators who performed the laboratory examination.13 Some experts speculate that the observed embedded crack occurred entirely during fabrication, but others believe that it may have grown during the startup/shutdown cycles.This remains a controversial issue in the assessment of heavy wall reactor vessels.

As part of the JIP onAging Reactors,16 two sets of laboratory tests were performed to determine if it is possible for embedded flaws in a reactor vessel to propagate during operating cycles,solely due to the buildup of molecular hydrogen pressure when the reactor cools to ambient temperature.The results of this testing showed that embedded cracks in a reactor did not grow during operating cycles,which is consistent with some UT examinations that have been performed on heavy wall reactors in the past.Repeated UT examinations were performed on several reactors after operating cycles to determine if any of the embedded flaws propagated in service.The results of these reinspections after operating periods indicated that the embedded flaws did not propagate during operation or the subsequent shutdown for these reactors.

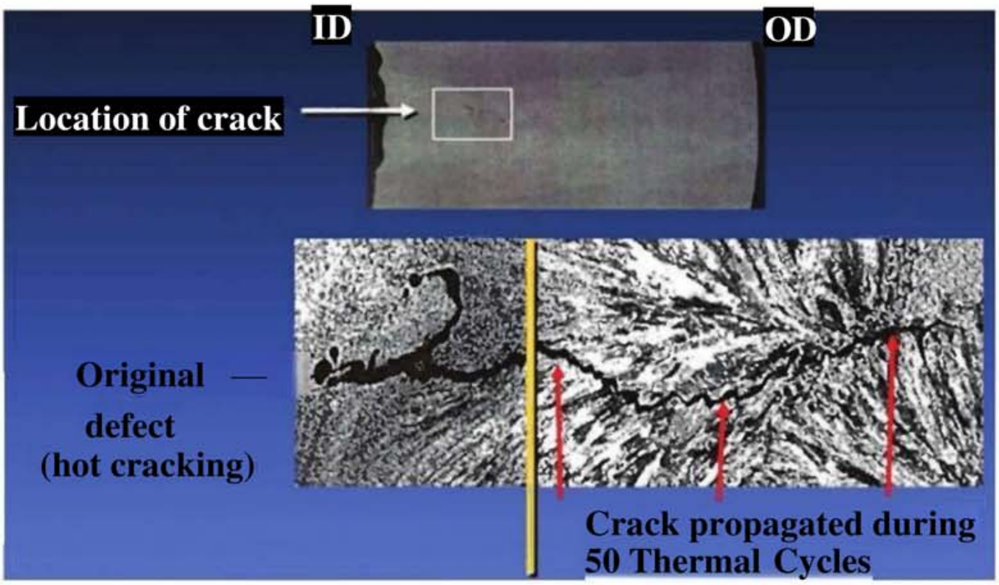


Figure 4—Embedded (Subsurface)Weld Flaw Depicted on a Metallurgical Cross Section113

Embedded crack-like flaws in a heavy wall reactor vessel in high-temperature high-hydrogen pressure service are difficult to assess using conventional fitness-for-service(FFS)assessment procedures as included in API 579.In additional to conventional applied loads,it is necessary to consider the internal molecular hydrogen pressure that is applied when the reactor vessel cools.If equilibrium molecular hydrogen pressure levels are reached within the embedded flaw,it is likely that the calculated stress intensity will exceed threshold values for incremental crack growth resulting from hydrogen embrittlement.33l Because of this uncertainty,and based on feedback from NDE operators suggesting in-service propagation of some flaws,some reactor owners have decided to perform periodic inspections on embedded flaws to make sure the observed embedded flaws have not grown.

6.5 Cracks in Support Skirt Attachment Welds

Reactor vessels,especially older vessels,may have a support skirt welded directly to the bottom head.This situation has caused some concern for the possibility of cracks initiating at the toe or the root side of the welds due to the combined effects of the stress concentration at these locations and thermal stresses generated by the temperature gradient commonly occurring in this dificult-to-insulate geometry.Thermal stresses in support skirt attachment welds can be mitigated by the installation of an insulated "hot box"that minimizes temperature gradients at the support skirt connection to the vessel.In some recently fabricated heavy wall vessels,the support skirt connection consists of a forged ring that moves the attachment weld away from an area with high stresses generated by the thermal gradient.

It has been reported by several owners that cracking at the skirt attachment has not been observed in heavy wall reactor vessels.As a result,they do not routinely inspect for this type of cracking;however,this may be contingent on the type of skirt attachment design.

6.6 Cracks in Nozzle Attachment Welds

Flaws associated with nozzle attachment welds,such as lack of sidewallfusion,can originate during fabrication. Experience shows that these embedded fabrication flaws have,on afew occasions,appeared to have propagated during service or startup/shutdown cycles.However,as discussed in 6.4 regarding embedded crack-like flaws, the observed flaws normally are evaluated with an FFS assessment.Due to the uncertainty associated with addressing the buildup of molecular hydrogen pressure within an embedded flaw,some reactor owners have decided to perform periodic inspections on embedded flaws to make sure the observed embedded flaws have notgrown.

6.7 Cracks in Quench Nozzles

Quench hydrogen is injected to maintain reactor bed temperatures and prevent excessive temperatures from the exothermic reaction that results from hydrotreating.Quench nozzle attachment areas can be prone to thermal fatigue cracking,especially if the nozzles are not equipped with a sleeve that runs through the nozzle(which is typically flanged to the internal distributor pipe)and distributes the cool quench hydrogen onto the catalyst bed. When the distributorpiping lines the quench nozzle,it acts as a thermal sleeve that minimizes thermal stresses acting on the nozzle.In cases where stainless steel liners are improperly installed in nozzles,they typically fail after a short operating period.

In a quench nozzle without an internal distributor pipe inside the nozzle,the nozzle wall is exposed directly to cool hydrogen gas.In this situation,cracking initiates at the corner where the end of the set-in nozzle meets the ID surface of the reactor.It then propagates axially along the inside surface of the nozzle.This type of cracking is more common in reactorvessels with internal SS cladding due to the high thermal coefficient of expansion for 300 series SS compared with the underlying Cr-Mo steel.Examination for this cracking is typically performed from the inside surface of the reactor during a turnaround using PT for clad or overlayed vessels and MT for vessels without cladding.An example ofthermal stress cracking observed on the inside surface of a vessel at a quench nozzle in a heavy wall reactor vessel is shown in Figure 5.

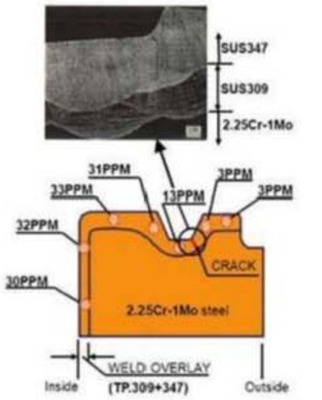
Figure 5—Cracking Observed at Quench Nozzle in a Heavy Wall Reactor Vessel

6.8 Cracks in the Ring-type Joint Gasket Grooves

Cracks have been found at the bottom corners of SS weld overlayed RTJ gasket grooves.Cracking typically has arrested in weld overlay,but in some cases,cracking has been found to propagate deeply into the base metal of the flange.Figures 6 and 7 show a typical crack in the corners of Type 347 SS weld overlayed RTJ grooves.14

Cracks have occurred on both the inside and outside corners of the groove,but tend to occur more frequently at the inside corner.Crack propagation is usually attributed to high stress concentration at the corners(especially if not properly radiused),temperature gradients through the flange,and high applied tensile stress in response to ring gasket compression.Crack initiation may also be caused by chloride SCC,and/or polythionic acid SCC during startup.

On heavy wall reactors thathave experienced ring groove cracking,many owners have modified the flanges or flange faces to change the original ring joint configuration to a raised face using a flat compressed gasket.The raised face flanges are not prone to RTJ cracking.Raised face flanges are now commonly used for new reactor fabrication rather than RTJ flanges.



DISTRIBUTION OF HYDROGEN IN
IP.347WELD OVERLAY RTJ GROOVE

Cause of crack

1)Sigma phase embrittlement by PWHT in thefabrication stage

2)Hydrogen embrittlement by high temperature and high hydrogen partialpressure in the long term operation

Figure 6—Cracks Associated with a RTJ Flange Typically Found at the Corner of the Groove

Figure 7—Cross Section of an RTJ Flange with Cracking from the InnerCornerthat Extended Deeply into the Cr-Mo BaseMetal

6.9 Cracks in SS Weld Overlay

Cracks have been found in 300 series SS weld overlay.These cracks can propagate through the weld overlay, but seldom propagate into the underlying Cr-Mo steel,with the exception of ring joint grooves as discussed in 6.8.This type of cracking is like cracking found at the corner of internal attachments as described in 6.3.The major difference is that the cracksfound at the corner of internal attachments occurmore readily because of the stress concentration that exists at the attachment.Figure 8 shows an example of cracking that occurs in the weld overlay.This figure also shows cracking originating at the corner of an internal attachment.In this case,cracks only propagated in the 300 series SS overlay and did not propagate into the underlying Cr-Mo steel.However, this cracking did expose the underlying Cr-Mo steel,resulting in sulfidation corrosion of the Cr-Mo steel.

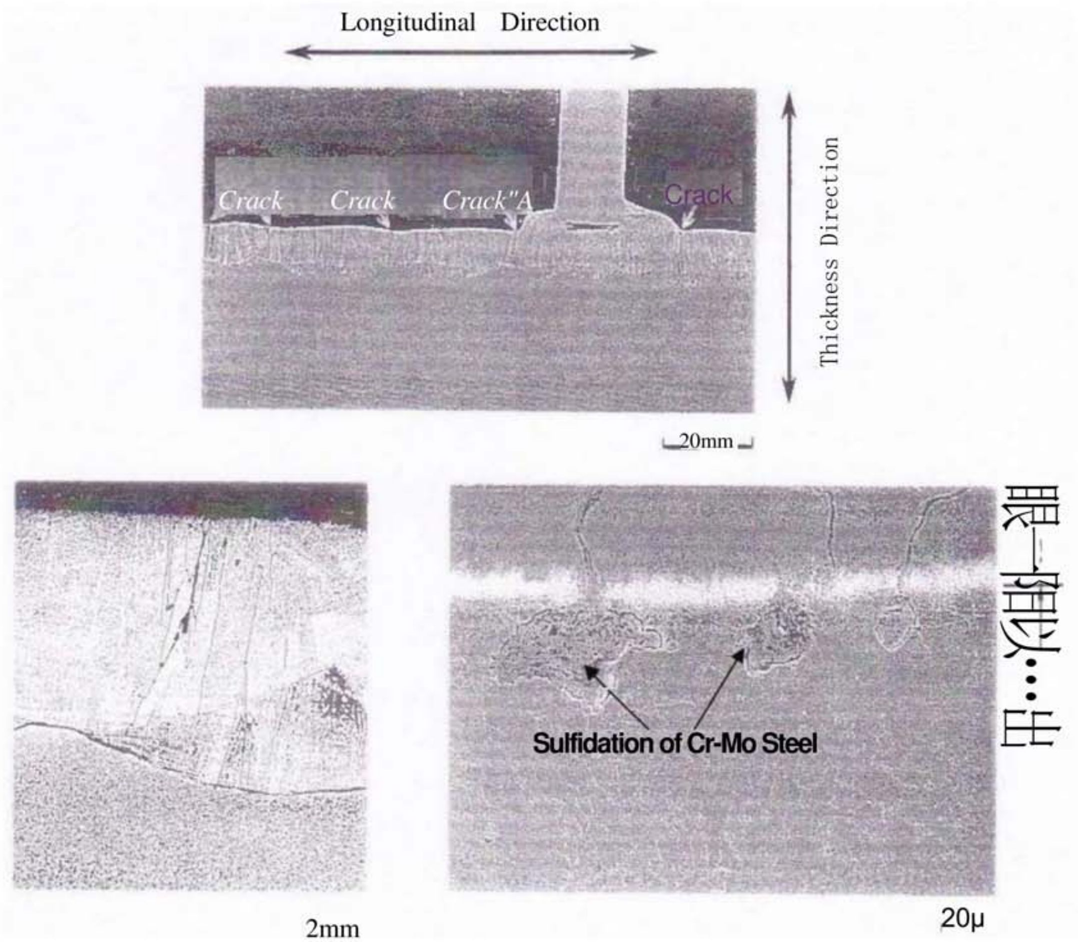


Figure 8—Cracks in Weld Overlay;With and Without Sulfidation of Underlying Base Metal

6.10 Cracks Along the Weld Overlay/Base Metal Interface (Disbonding)

Cracks have been found along the interface between 300 series SS weld overlay and base metal.This type of cracking is often called weld overlay disbonding.Cracking can occur either in the SS weld overlay or the Cr-Mo base metal near the interface.At elevated temperatures in service,hydrogen difuses through the overlay and base metal;however,during the shutdown process,dissolved hydrogen tends to build up along the overlay-to-base metal interface due to the difference in solubility and difusivity of hydrogen between the Cr-Mo base metal and the SS weld overlay.The high concentration of hydrogen at the interface during shutdown periods when the reactor vessel is cool promotes hydrogen-assisted cracking driven by stresses generated by the difference in coefficient of thermal expansion between the 300 series SS overlay and the Cr-Mo base metal.The vanadium-enhanced grades of Cr-Mo base metals have shown to be resistant to disbonding with SAW or ESW weld overlays applied using typical procedures.4-8]

Figure 9 shows an example of disbonding cracks at the interface between the SS weld overlay and the base metal.This cracking is most common in older reactor vessels that contain a weld overlay applied by a high heat input welding process,which is typical for early applications of ESW.A high heat input welding process can result in a coarse-grain structure in both the weld deposit and HAZ in the Cr-Mo base metal at the interface.15 These coarse-grain structures are more susceptible to hydrogen-assisted cracking when the reactor is cooled to ambient temperature.Cracking from weld overlay disbonding runs parallel to the surface of the reactor wall and experience has shown that this type of cracking does not propagate in the through-wall direction.

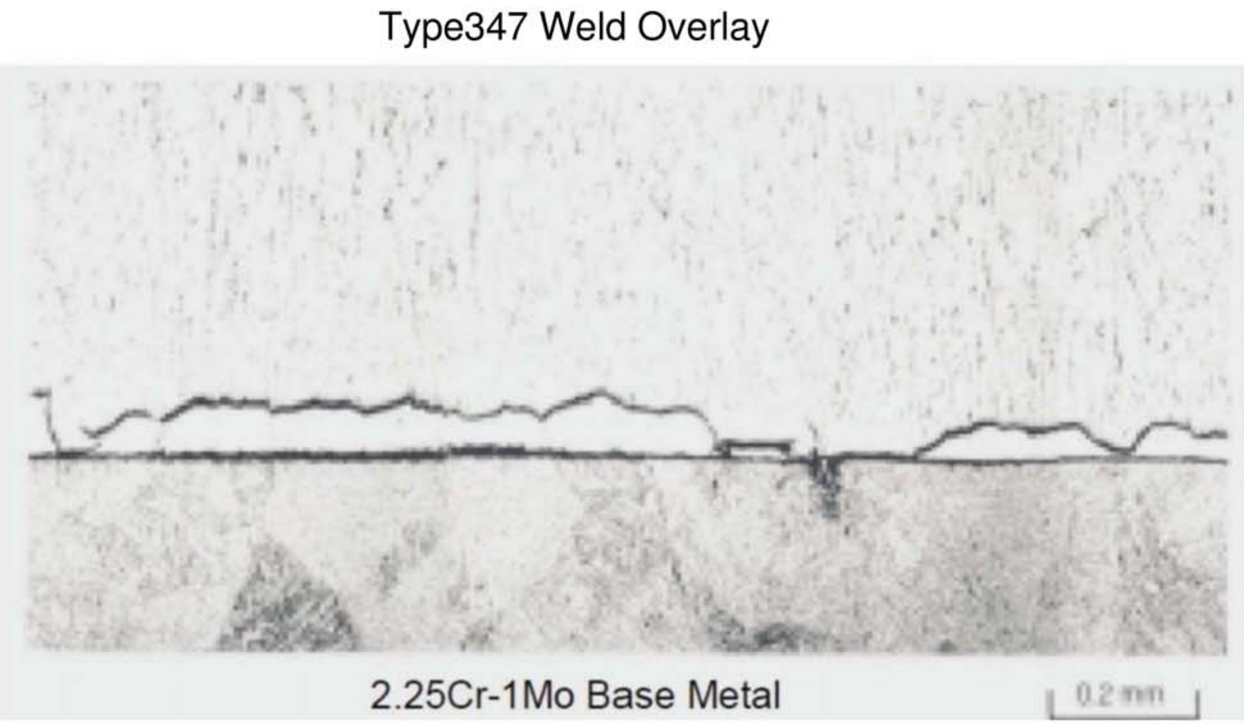


Figure 9-Weld Overlay Disbondment (Cracks Can Be in SS or Cr-Mo HAZ)[15]

In the last 30 years,the ESW process for applying a weld overlay in reactor vessels has been adjusted to avoid the formation of coarse-grain weld deposits and HAZs in the underlying base metal.This has greatly reduced the possibility for disbonding at the interface between the weld overlay and underlying base metal.Today,it is common that the procedure for applying a weld overlay is qualified with a test to demonstrate a high resistance to disbonding.These testing requirements are provided in API 934-A,which also reflects the better resistance of the vanadium-enhanced grades,as the testing requirements for these grades are much less stringent.

In addition to the disbonding cracking associated with weld overlay applications,there is also a risk of disbonding associated with hot roll or explosive bonded cladding.This has occurred at the interface between the SS cladding and the Cr-Mo base metal and,in most cases,has been related to inadequate quality control during the original cladding process.Roll bond cladding is considered to have a weaker bond between the cladding and the base metal compared with overlay,but in the lower temperature and lower pressure applications where it is normally used,the bond strength has proven to be adequate.As discussed in 5.3,explosive bonded cladding has been tested using the same high-temperature high-pressure autoclave used for weld overlay and has been found to provide an adequate resistance to disbonding cracking in conditions where explosive bonding is used.31,32

6.11 Cracks in Wrought 300 Series Stainless Steel Support Rings

In many cases where solid SS rings are used to support catalyst beds,it has been found that the ring cracks after multiple startup and shutdown cycles.Figure 10 shows a typical radial crack found in a solid SS support ring in a reactor vessel.This type of cracking,which is not llustrated in Figure 2,results from the difference in thermal coefficient of expansion between 300 series SS and the Cr-Mo reactor steel.This produces very high residual tensile stresses in the ring on shutdown when the vessel cools to ambient temperature.

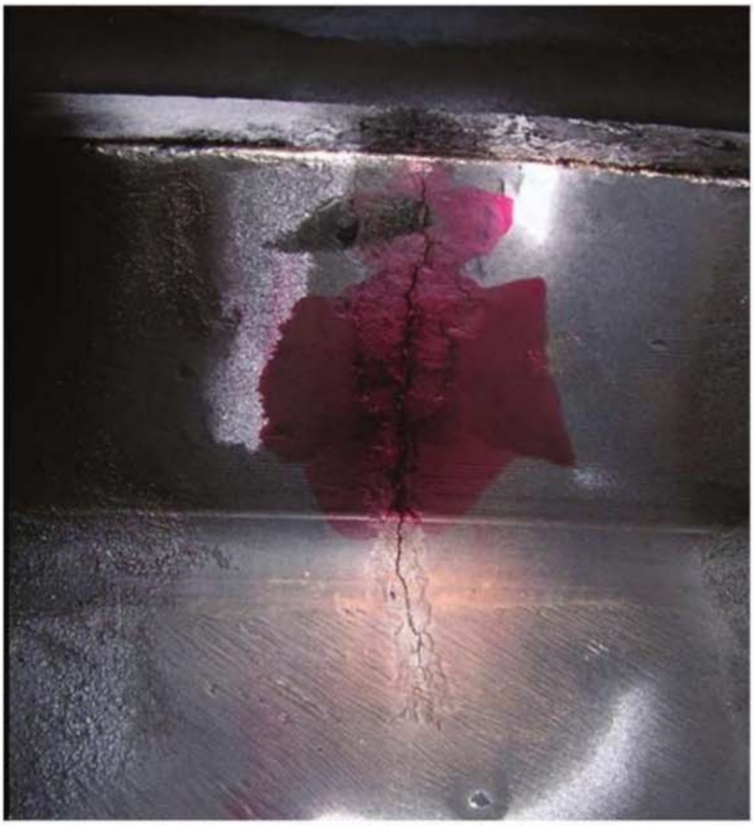


Figure 10—Radial Cracking on Solid 300 Series SS Support Rings

This type of cracking is avoided in more recent designs by constructing the ring from the same Cr-Mo steel as the shell,welding it to the shell base metal(not the SS overlay),and then overlaying the ring and exposed base metal with SS to provide the needed corrosion protection.In modern reactors constructed from forged sections, these Cr-Mo support rings are sometimes integrally included as part ofthe forging.In older reactors,the support rings were made with Cr-Mo weld buildups in some reactors.

6.12 HTHA in Base Metal

As described in API941,HTHA can cause fissures and cracks in carbon or low alloy steel vessels,as well as other equipment and piping operating at high temperatures and high hydrogen partial pressures.Fissures and cracks initiate at internal grain boundaries from the pressure of methane gas formed by the combining of the carbon with dissolved atomic hydrogen in the steel,which also results in the decomposition of steel's metal carbides.The formation of fissures and cracks in a vessel depends on the alloy content of the carbides in the steel,operating temperature,effective hydrogen partial pressure,imposed stresses including thermal and residual welding stresses,and time in service.

While there have been a few reported incidents of HTHAin C-½Mo reactor vessels,to date there have been no reported incidents of HTHA in heavy wall vessels fabricated from any of the grades of Cr-Mo steels.

6.13 Damage Due to Reactor Overheating

During aprocess temperature exotherm excursion or a fire,a Cr-Mo reactor vessel can be heated to above 760℃ (1400 °F),which is the temperature at which Cr-Mo steel can undergo a transformation to martensite during cooling.If untempered martensite is present in the Cr-Mo reactor steel,it can have a very high hardness-well above the hardness level where Cr-Mo steels are adversely affected by hydrogen embrittlement.In each case where a significant overheating situation is experienced,it is appropriate to collect the inspection and hardness data necessary to determine if transformation to a hard martensite phase has occurred,to the point where there is a significant risk for cracking due to hydrogen embrittlement.In one case where a reactor experienced fire damage,high hardness readings were observed only on the exterior surface of the reactor vessel.

Almost all heavy wall reactor vessels are insulated on the outside surface.This is a first line of defense in preventing fire damage.It also provides an initial screening for the likelihood for fire damage.After a fire,if the reactor external insulation and weather jacking is in good condition without any signs of distress,it is a strong indication that the reactor itself was not heated above the transformation temperature (760℃or 1400°F)at which the reactor can suffer metallurgical damage.API 579 Part 11 is typically used to determine suitability for reactors exposed to potential fire damage.

It is generally accepted that the hardness limit above which Cr-Mo steels can be affected by hydrogen embrittlement is~225 BHN for 2_4Cr-1Mo steels and 248 BHN for V-modified steels.

Excessive heating of a reactor vessel during a fire or process temperature excursion can also adversely affect the resistance ofan internal 300 series SS overlay to intergranular corrosion and cracking mechanisms,such as polythionic acid SCC.

There are very few incidents of fire damage and processtemperature excursions that have altered the properties of the Cr-Mo base metal or the internal 300 series SS overlay.This is not surprising,because it would take a very longtime for a refinery fire or a process temperature excursion to heat up a heavy wall reactor vessel to a temperature where damage would occur.Typically,a refinery fire will be extinguished quickly,and operating measures will be implemented to correct a process temperature excursion in a short period oftime,before the reactor metal can heat up to a temperature where damage could occur.

Areactor vessel heated toa high temperature inthe 595 to 650℃(1100 to 1200°F)range while still at operating pressure can experiencea short-term rupture.Short-term ruptures have occurred in reactor vessels and in outlet piping during process temperature excursions associated with exothermic reactions in reactor catalyst beds. Typically,process operators can control these exothermic reactions so that overheating is confined to a small section of the reactor catalyst bed and the reactor metal temperature does not approach a level much above normal operating levels.

7 Inspection and NDE of Heavy Wall Reactor Vessels

7.1 During Fabrication

There are several recommended inspections and examinations during the various stages of fabrication of heavy wall reactor vessels.They are defined in API 934-A for heavy wall reactors fabricated from conventional 2_4Cr-1Mo,conventional 3Cr-1Mo,2/$_4$Cr-1Mo-/4V,and 3Cr-1Mo-/AV steels,and in API 934-C for heavy wall reactors fabricated from conventional 1Cr-/$_2$Mo and 1/$_4$Cr-/$_2$Mo steels.It is important to understand the extent of NDE performed on a reactor vessel during fabrication.This establishes a baseline inspection level for a reactorvessel that is used to plan future inspections after service,which is the primary focus of this section.

7.2 Implication of Using Different Inspection Methods than Used in Original Fabrication

Several NDE techniques have been used on heavy wall reactor vessels after operating in service.In many cases,the postconstruction NDE techniques use technologies and approaches that did not exist during the original fabrication.In some cases,the resolution of modern NDE techniques,such as advanced UT,greatly surpasses the resolution of NDE techniques,such as RT,used originally.Owners may find flaws that are hard to differentiate as original fabrication flaws or flaws created in service.In either case,the owner will need to appropriately interpret and manage the NDE results.(See Section 8on FFS analyses.)

7.3 Qualified NDE Personnel

Given the nature and criticality of heavy wall reactors,the need for proper use of NDE technology and interpretation of NDE results warrants measures to ensure the use of qualified NDE technicians.Most owners require that NDE technicians be qualified to American Society for Nondestructive Testing(ASNT)or equivalent qualification systems as a minimum.Typically,a reactor inspection justifies the use of ASNT SNT-TC-1A Level I or Level qualified NDE technicians.Some owners go beyond industry certification and require the use of "performance qualified"NDE personnel who have passed company-specific qualification tests that helps insure a proficiency in the use of the NDE technique and interpretation of its results.Performance qualification is most often used for ultrasonic inspection technologies such as time-of-fight crack tip diffraction(TOFD),phased-array ultrasonics(PAUT)and full matrix capture(FMC),in which interpretation of the ultrasonic signals can vary from NDE technician to NDE technician.Frequently,a performance qualification is performed on mock-ups to test the technician performing the examination.

When relevant indications are found,the implications of this finding(perhaps leading to FFS evaluation for possible acceptance or possibly a difficult and expensive reactor repair,with an associated heattreatment)may justify obtaining a confirmation inspection(i.e.a second opinion)by another similarly qualified NDE technician. The follow-up evaluation may be done by another NDE technician and/or an independent NDE entity.

7.4 NDE Methods

7.4.1 General

This section discusses various NDE technologies and procedures used to detect the potential damage mechanisms in heavy wall reactor vessels,which were discussed in Section 5,at the typical locations that were described in Section 6.Details about NDE methods,techniques,and essential variables can be found in the 2019 Edition of ASME BPVC Section V,API 510,API 572,and References [24],[25],and [26].Guidance on the sensitivity of flaw detection and sizing for FFS evaluations is found in BS 7910,Annex T.Section 7.5 discusses how these NDEtechniques can be applied more specifically at the reactor component level.Once a flaw is found and vetted,Section 8 on fitness for service can be used to determine if a repair is needed,and if so,what kind of repair is needed.

7.4.2 Penetrant Testing

Penetrant testing(PT),commonly referred to as dye penetrant testing,frequently is used to examine the inside surface of internal 300 series SS weld overlay for crack-ike flaws.It also has been used to examine 300 series SS supports,internals,and ring grooves on flanges.This examination is performed when the reactor is cool, catalyst is removed from the reactor vessel,and internals blocking internal surfaces are removed.Examination typically focuses on filet welds associated with internal attachments where a stress concentration can exist, making cracking more likely.PT testing also is commonly used to detect cracking in ring grooves on flanges. Most 300 series SS surfaces on heavy wall reactor vessels are examined by PTusing a red penetrant with white developer/background.

For optimal detection of cracks by PT,it is necessary to remove surface scale by cleaning techniques such as water/steam cleaning,light grit blasting,flapper wheel cleaning,and,under rareconditions,lightgrinding.When grit blasting is considered for surface preparation,the blasting equipment is preferably adjusted to a low air pressure with a high volume of blast media.Soda blasting is a good alternative to cleaning surfaces compared with cleaning techniques using more aggressive blast media such as sand or garnet.Grit blasting can compress the surfaces to be inspected,resulting in closure ofcracks.Also,grit can become embedded in cracks open to the surface.

Selection ofa cleaning technique depends on the surface being examined and the amount of area to be cleaned. Light grit blasting is usually satisfactory for all surfaces but may incurproblems with the dust itcreates and how it affects other work in the area.Light grit blasting is particularly well suited for preparing ring groove surfaces and will ot greatly affect machined surfaces.Flapper wheel preparation is good for smallerareas and has minimal effect on workers but is not suitable for machined surfaces.Finally,light surface grinding may work for preparing the base of fllet welds or other suitable locations.Wire wheel surface cleaning is generally not used due to SS smearing and causing closure of tight cracks.

PT typically provides very good crack detection;however,it does not determine the depth of a crack into the weld overlay or whether the crack propagates into the underlying Cr-Mo reactor wall.Frequently,light grinding will be performed to determine the depth of a crack detected by PT.However,experience has shown grinding on a surface-breaking crack in 300 series SS weld overlay in a reactor usually results in a crack entirely through the weld overlay thickness.For this reason,owners frequently do not grind on cracks in 300 series weld overlay and instead determine if cracking has penetrated into the underlying base metal using either TOFD (see 7.4.7) or PAUT(see 7.4.8)from the outside surface.Figure 6 shows a photo of a surface-breaking crack in 300 series SS weld overlay on a ring joint groove found by PT examination.

7.4.3 Magnetic Particle Testing

Magnetic particle testing(MT)is performed on the external surfaces of Cr-Mo vessels(e.g.external attachment welds,nozzle attachment welds,and skirt attachment welds)and on the internal surfaces of Cr-Mo vessels without SS cladding/weld overlay.MT has two primary techniques:dry MTand wet florescent MT(WFMT).Dry MT provides less crack resolution but is easier to perform withfewer restrictions.WFMThas better resolution of fine cracking than dry-MT but must be performed at near-ambient temperatures and needs low-light conditions for proper viewing.It also is possible to use wet MT employing a black powder on a white background.This provides a good contrast with sensitive crack detection without the need for low-light conditions.When scale needs to be removed for MT examination,surface preparation as provided for PTin 7.4.2 is typically employed.

MT primarily detects surface-breaking cracks,but can detect subsurface cracks that almost break to the surface. It does not determine the depth of cracking.

7.4.4 Radiography

On older reactor vessels,RT typically was performed during vessel fabrication on welds to find volumetric flaws. Since the wall ofa reactor vessel is thick,a high-intensity radiation source is needed to achieve the necessary sensitivity.The need for a high-intensity radiation source makes the use of RT of a reactor vessel impractical once it has been placed in service in a refinery.Most postconstruction inspection conducted in the field occurs during a turnaround when other workers are present.As a result,RT is not typically used to inspect vessels to find damage that occurs in service.

In the 1990s,fabricators of heavy wall reactors started to use UT techniques to inspect new reactors in lieu of RTI[3]as permitted by Code Case 2235.This Code Case was approved byASME B&PV Commiee in 1996 for vessels constructed to the rules of Section VIII Division 2 ofthe ASME BPVC and has since been incorporated into the Code.In many cases,fabricators use mock-up assemblies with built-in volumetric and crack-like flaws to qualify their UT inspection procedures.Experience has shown that since fabricators have gone with qualified UT inspection procedures,there has been a noticeable reduction and almost elimination of embedded crack-like flaws in heavy wall reactor vessels.Additional details on the use of UTinspection technology on heavy wall reactor vessels is included in7.4.5,7.4.6,7.4.7,and 7.4.8.

7.4.5 Straight-beam Ultrasonic Testing

Straight-beam ultrasonic testing(UT)has been used primarily for two specific in-service concerns in heavy wall reactor vessels.First,straight-beam UT from the outside surface of the reactor is useful in determining if disbonding of the weld overlay or cladding has occurred.Second,straight-beam UT has been used from the internal bore of a nozzle to determine if cracking has initiated from the corner of a ring groove in an RTJ flange. In some unique situations,straight beam UT can be used in conjunction with other inspection techniques(e.g. shear wave UT,PAUT)to locate cracks,confirm orientation,and,in certain situations,size cracks.It can also be used to measure wall thicknesses.

7.4.6 Manual and Automated Angle Beam Ultrasonic Testing

In recent years,many new types of ultrasonic flaw detection technology have become readily available and have frequently been applied to heavy wall pressure vessels.In the past,the most commonly applied ultrasonic flaw detection for weld examination was shear wave UT(SWUT).The word"shear"most commonly referred to the orientation of sound propagation.In most cases,the ultrasonic wave propagation during SWUT was angled between 45,60,and 70 degrees to the entry surface.However,"shear wave"is a form of sound-energy propagation.There are multiple types of ultrasonic wave modes;the two most common are longitudinal and shear. Sound propagation entering material at an angle can be in the shear mode or longitudinal mode,especially with current technology.Because of the two means of sound propagation,the industry has adopted the term“angle beam"in lieu of shear waveto refer to ultrasonic testing using sound propagation at an angle to the entry surface.

The latest technology includes ultrasonic transducers and instruments that operate multiple piezoelectric elements.This technology is referred to as phased-array ultrasonic testing(PAUT).Both shear and longitudinal wave modes can be produced and used during PAUT.See 7.4.8 for a discussion of PAUT.

sWUT has been the UT technique with the most variability in interpretation of defects between technicians and has particularly benefited from performance testing to find the most skilled individuals,and from seeking second opinions as noted above.Historically,SWUT was used to examine reactor pressure welds,nozzle welds,and gasket groove cracks.SWUT around heavy wall nozzles to detect potentially embedded cracking has been particularly difficult.Tools and techniques to improve SWUT in difficult orientations were developed over the years but have mostly been surpassed by use of PAUT or other techniques.

Ultrasonic transducers,including angle and straight beam,attached to a mechanical device that uses an encoder to track position,is commonly referred to as automated ultrasonic testing(AUT).In most cases,the mechanical device is referred to as a scanner.Such mechanical devices fall into two categories:automated and semi-automated.Both automated and semi-automated have one or more encoders tracking the position of the ultrasonic transducer.Automated scanners use electric motors to propel the UT transducer(s)and mechanical device in one or more directions,while semi-automated scanners are moved manually.Tracking the position of the UT transducer via encoding greatly complements imaging of the ultrasonic sound energy reflected and returning from anomalies.Imaging improves characterization,sizing both length and depth alongwith creating a permanent record of the areas scanned.

Automated angle and straight beam ultrasonic testing(AUT)provides recording of UT data including UT transducer position and location of anomalies.Straight beam AUT is one of the best methods to examine reactor shells and heads for potential disbonding of the weld overlay or cladding and provides a record,while manual straight beam UT(MUT)does not.Angle beam AUT can also be used to look for cracking in shell and shell-to-head pressure butt welds.Once a flaw is located via AUT,other ultrasonic techniques are generally employed to supplement the AUT to confirm the size and orientation of the flaw.

7.4.7 Time-of-Flight Crack Tip Diffraction Ultrasonic Testing

Time-of-flight crack tip diffraction ultrasonic testing(TOFD)is an ultrasonic technique used to detect diffracted waves from crack tips and size the cracks from the arrival times of those waves.TOFDhas two major advantages compared with the conventional pulse echo technique.First,it is only weakly sensitive to flaw orientation, whereas pulse echo relies on specular reflection of the waves.Second,the determination of the flaw size relies onlyon being able to measure the arival time of the signals and not,as with pulse echo,on measuring the signal amplitude.

The basic arrangement of the twin probe technique for TOFD is illustrated in Figure 11.One probe(transducer Tx)transmits ultrasound and the other (transducer Rx)acts as the receiver.Probes with beams as diverging as possible are used to increase the beam coverage.The beam coverage is usually very close to the separation between probes,as llustrated in Figure 11,which istwice the probe-center separation.This separation typically can cover not only the weld cross section but alsothe heat-affected zone and part of the base metal.29

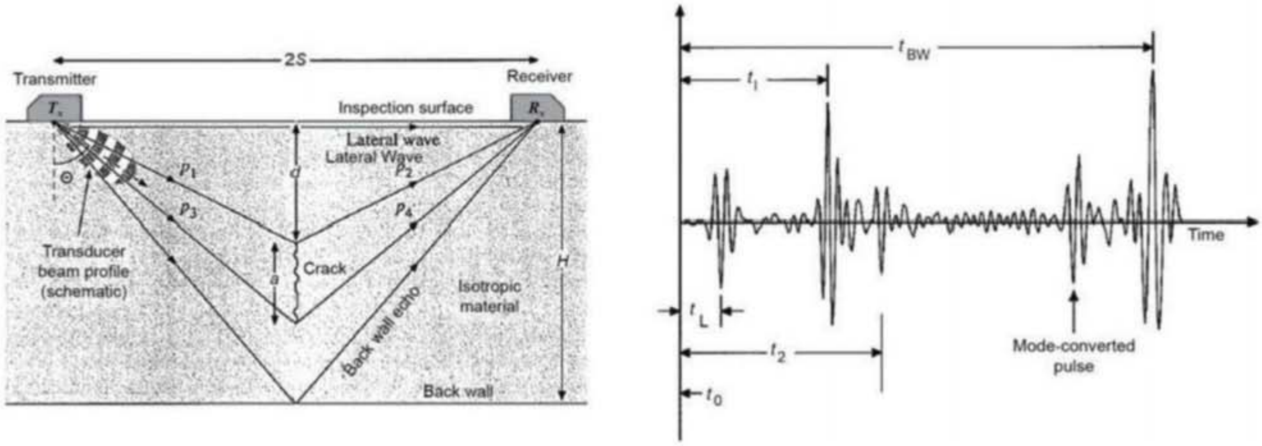


Figure 11—Typical Transducer Arrangement for TOFD Inspection of Welds[29]

TOFD has an advantage over conventional UT examination techniques because it can detect the difracted signals from discontinuities and/or flaws,process the diffracted signals,and compare them to the back-wall signal and structural noise to locate and size the discontinuity and/or flaw.It also allows the creation of an image that shows the discontinuity and/or flaw location in relation to the wall thickness.

The TOFD technique provides information not only on the diffracted waves but also on the different propagation modes,such as:

—The lateral wave:This is a propagation wave following the shortest path between the transducers that is the first wave arriving at the receiving transducer.These surface waves detect surface-breaking flaws and indicate the location of the flaws in relation to the transducers.

-The longitudinal andtraverse waves:These waves spread inside the material at different speeds,sweeping the entire volume,detecting internal discontinuities/flaws and producing a specular reflectionfrom the opposite surface.This second signal,commonly called a back-wall reflection,reflects off of the opposite surface from where the transducers are located.The region of the image defined by the lateral wave and the back-wall reflection represents a full thickness cut through the thickness.

—*The diffracted and reflected waves:These waves take place due to the interaction of the ultrasonic beam with* the discontinuities/flaws inside the material.These signals arrive in an intermediate time between the lateral wave and the back-wall reflection.

Detection and sizing of imperfections between TOFD sensors are in relation to the time fight of difracted and reflected signals as the ultrasound beam interacts with discontinuities/flaws.Orientation of the detected discontinuities/flaws does not have effect on TOFD processingof the signals.

Signals obtained from the propagation waves,shown in Figure 11 on a two-dimensional image,commonly known as a B-scan,are obtained in real time,as the two transducers move on the surface of the component.A typical image is shown in Figure 12,where the horizontal axis represents the displacement of the transducers on the inspection surface.The vertical axis represents the times of arrival of the signals coming from the lateral wave and the reflected and diffracted waves for the observed discontinuities/laws in the material,and the back-wall reflection.

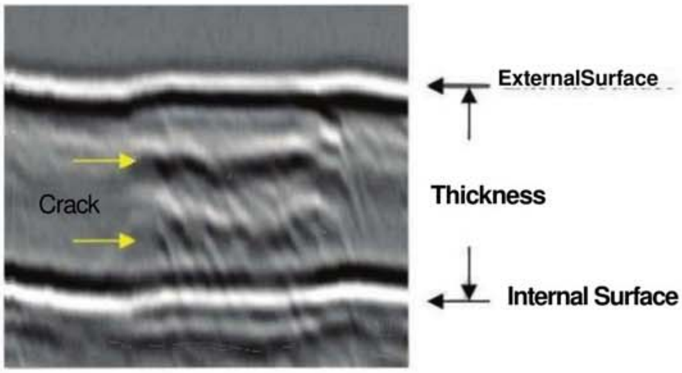


Figure 12—B-scan Image Obtained by the Application of the TOFD UT Technique Showing the Presence of an Embedded Flaw

TOFD has primarily been used from a reactor outerdiameter(OD)surface during downtimes to monitor cracks observed on the ID surface,typically found by PTexamination of the 300 series SS overlay on the interior surface of a reactor.TOFD has also been performed on welds to detect both surface-breaking and embedded crack-ike flaws.

TOFD is normally used for examination of reactor pressure boundary welds such as circumferential and long seam welds with examinations conducted by transducers both parallel to and normal to the weld.TOFD is not applicable for nozzle neck-to-head or nozzle neck-to-shell welds.

Shallow cracks connected to the inside or outside surfaces cannot be detected by TOFD due to dead zones on the lateral and back wall signals.Normally,other UT techniques,such as PAUT and SWUT,are used to detect shallow surface-breaking flaws.

7.4.8 Phased-array Ultrasonic Testing

Phased-array ultrasonic testing(PAUT)provides multiple ultrasonic angle and straight beams of sound propagation while performing UT examinations.Angle-beam PAUT can generate 30 refracted angles of sound propagation during sectorial scanning.Straight-beam PAUT can generate an effective coverage up to 125 mm (5in.)wide.PAUT examination is used interchangeably with TOFD.PAUT has the advantage of being able to produce an image of the flaws and other surfaces for interpretation and sharing with others.Manual monolithic probes and conventional ultrasonic flaw detectors used to generate SWUT do not provide such an image,but rather require calculation of angles and sketches of what the technician is seeing ultrasonically.Similar to TOFD, PAUT examination of pressure welds and nozzles requires removal of insulation.

The PAUT technique is a process wherein UT data are generated by constructive phasing formed by a single PAUT probe containing multiple elements (e.g.10,16,32,or 64 elements)controlled by accurate time-delayed pulses to each element.28APAUT probe can sweep the sound through an angular range (sectorial or S-scans) at a fixedangle(electronic or E-scans),focus the sound beamwith lateral or line scans,or perform raster scans, depending on the array and programming of the PAUT instrument.Each element consists of an individually wired UT probe with appropriate pulsers,multiplexers,and converters.Each of the PAUT elements are acoustically insulated from each other.Imaging using a PAUTinstrument includes A-scans,B-scans,C-scans and S-scans.As shown in Figure 13,the echo from the desired focal point hits the various transducer elements with a computable time shift.

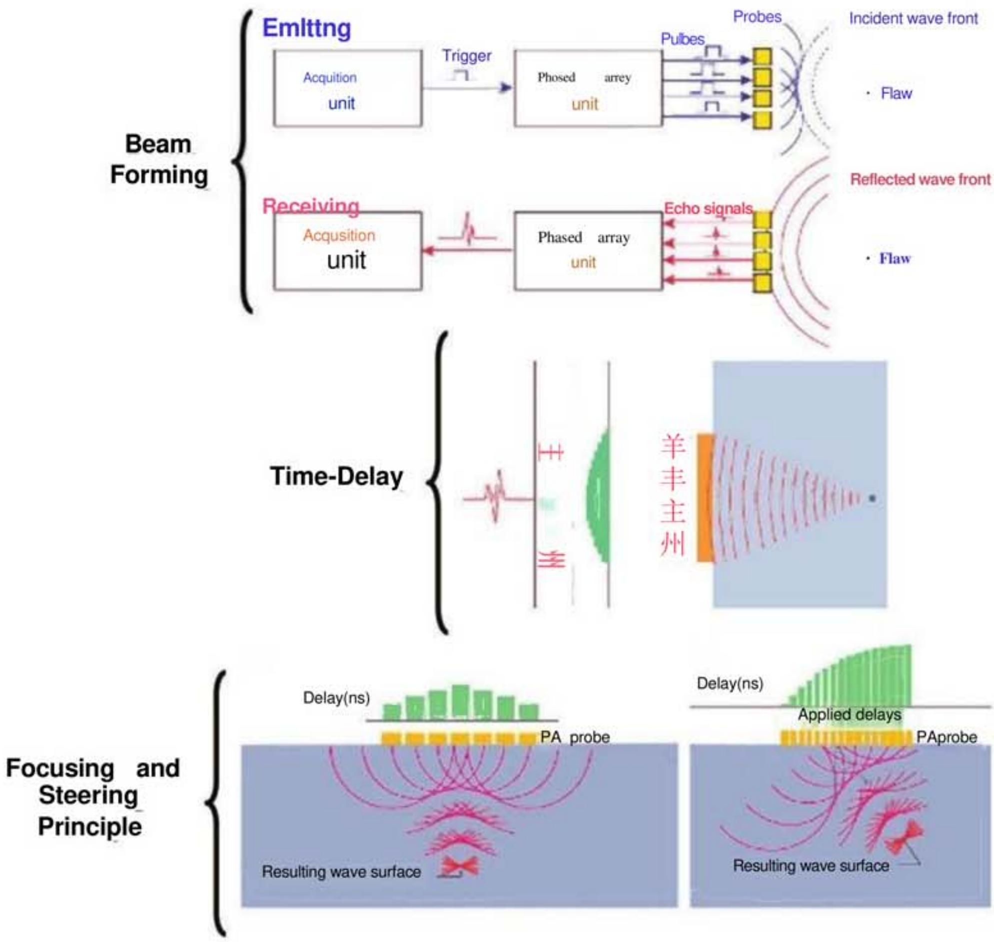


Figure 13—Beam Forming,Time Delay,and Focusing and Steering Principle Involved with a PAUT Probe

There are several characteristics associated with the application of PAUT,as follows:

1)PAUT technology has a significant advantage over conventional single-crystal UTtechnology in detecting and sizing flaws of variable orientation.Figure 14 shows the difference between a single crystal and a PAUT in detecting crack-like flaws that are randomly oriented in a steel plate.

2)PAUT hardware is more complex and expensive than conventional UT hardware because of the enhanced capabilities it provides.Typically,PAUT hardware integrates conventional UT,AUT,and TOFD functionalities.

3)PAUT technology basic and advanced training are readily available,but developing qualified technicians for large-scale inspection effort such as a refinery turnaround requires more time and planning than for conventional UT.

4)PAUT technique calibration requirements for the probe and instrument and periodic routine checking for the system functionality are comparable to the other UT techniques,but involve longer calibration time because of the multiple elements in the probe and the multiplexing architecture of the electronics.

5)PAUT multiview(A-B-C-D-S scans)analysis and interpretation provides more reliable and accurate data for the damage dimensions and FFS assessment,but is time consuming.

6)PAUT is more commonly used for vessel inspection than TOFD and is already integrated into key existing standards despite the complexity of the technology.

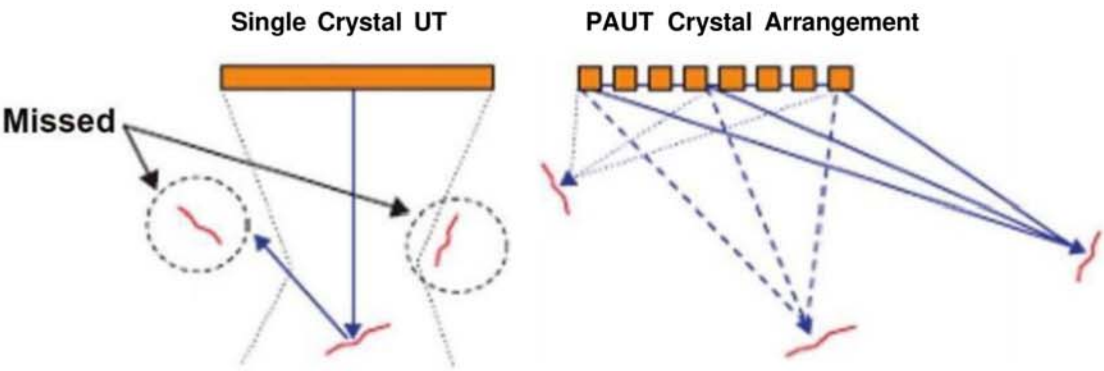
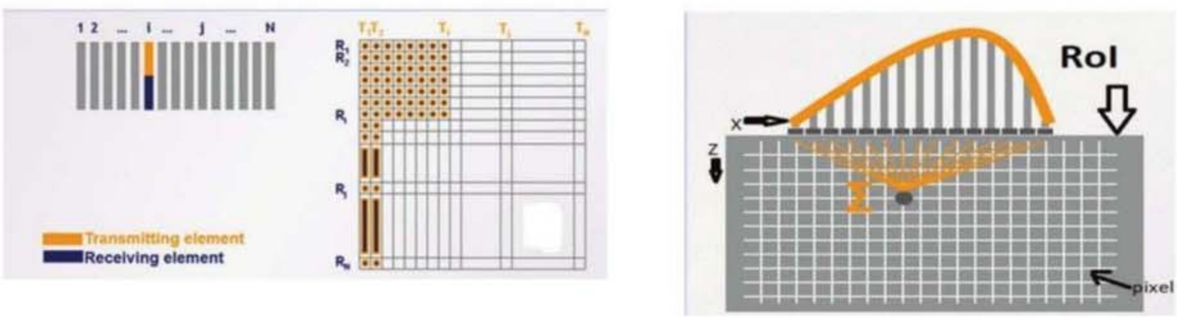


Figure 14-PAUT Crystal Arrangement Provides Improved Ability to Detect Crack-like Flaws Compared with Single-Crystal UT

7.4.9 Full Matrix Capture/Total Focusing Method To Characterize Cracks for FFS

The full matrix capture/total focusing method(FMC/TFM)is codified by ASME as a UT technique when examinations are performed for fracture-mechanics-based acceptance criteria.At the FMC acquisition step,the complete raw data set from every transmitter-receiver pair is collected and buffered into a matrix as shown in Figure 15,panel a.During one sequence,an ultrasonic wave is emitted by one single element of the array;the reception of the corresponding echoes is made by all the elements of the array.The process is repeated until all elements of the array have been used as transceivers.The most often used TFM postprocessing algorithm generates an image in which the beam has been focused at the reception on every point within the region of interest in a grid of colored pixel representations,as shown in Figure 15,panel b.The use of FMC/TFM is an enhancement over PAUT and TOFD.FMC/TFM provides improved detectability and characterization and better sizing of crack-like damage.



a)Classic FMC b)TFM

Figure 15—Classic FMC Basics and TFM Concept

7.4.10 In Situ Metallography or Field Metallographic Replication

In situ metallography,also known as field metallography or field metallographic replication(FMR),is a nondestructive technique to perform an evaluation of the microstructure or morphology of a crack in either a 300 series SS overlay or Cr-Mo base material.The technique involves polishing and etching the metal surface and then transfering the etched microstructure and associated crack features on the surface onto an acetate-based film that can be examined with a field microscope or materials laboratory metallograph.

Field metallography requires a high level of training and experience to do correctly.It is common that field metallography samples do not always properly transfer the microstructural features onto the acetate tape. Therefore,it is importantthat a skilled technician perform the work to ensure that samples properly capture the microstructure.Field microscopes help confirm whether good samples have been obtained.

7.4.11 Three-Dimensional Optical Measurement

Commercial imaging tools are available to assist in measuring the depth and area of corrosion loss or depth or the area of an excavated crack.The information these tools collect is useful in providing a three-dimensional (3D)map of the surface that can be fed into a formal FFS analysis.These commercial tools vary in their imaging methods,but most use some form of laser and LED structural light technology,as shown in Table 3.Resolution and accuracy of the optical technique can be evaluated prior to selection of the technology,and results are typically calibrated using standard gages.

Table 3—Comparison of Typical Optical Measurement Methods

Features	Laser Light-Section Method	Photogrammetry Method	3D Structured Light Method
Setting	Needs precise setting in most cases.Calibration on site may be required.	Easy to set only when the angle and the base line of two cameras are fixed.	Very easy to set.Objects just need to be in the range of the specified working distance.
Time to capture	One line scan is very fast	Slow,because many shots are neededto cover all angles.	0.3 seconds by one commercial method
Time to pointcloud	Multiple numbers of lines	Slow,due to matching of left and right images.	3 seconds by one commercial method
Resolution	Depends online sensor resolution.A fewmicrons most likely.	Depends on camera resolution.A few millimeters in most cases.	With 300K camera,XY resolution is 0.2 mm~0.4 mm.
Objects that are unable to be scanned	Objects with no reflection	Objects with no texture.	Objects with no specular reflection.
Errors	Multiple resolutions.	Depends on the objects. Sometimes very large.	Multiple resolutions.
Training required?	Yes,a few days.	Yes,but easy to understand how to use.	Yes,about half a day training is required for better scanning.

7.4.12 Ferrite Testing

Ferrite testing is typically a quality control check performed as part of a 300 series SS overlay application or repair.Ferrite testing is not included as part ofa reactor inspection during a downtime.The purpose of ferrite testing is to ensure a minimum level of ferrite to prevent hot cracking of the weld as it cools,while also verifying the maximum amount of ferrite that can potentially transform to britle sigma phase during a subsequent heat treatment.This testing is not performed as part of a routine compliance inspection when no weld repairs are completed.Company-specific and industry-related welding guidance,such as API Recommended Practice 582, sets acceptable levels of ferrite in 300 series SS welds.

Measurements are typically taken using a commercially available ferrite scope or ferrite meter.Ferrite can also be determined by in situ metallography using the ferrite point-count method,or estimated by chemical composition using a constitution diagram,such as contained in WRC 1992.Diagram available in WRC Buletin 519 and in ASME BPVC Section II,Part C,SFA5.4.

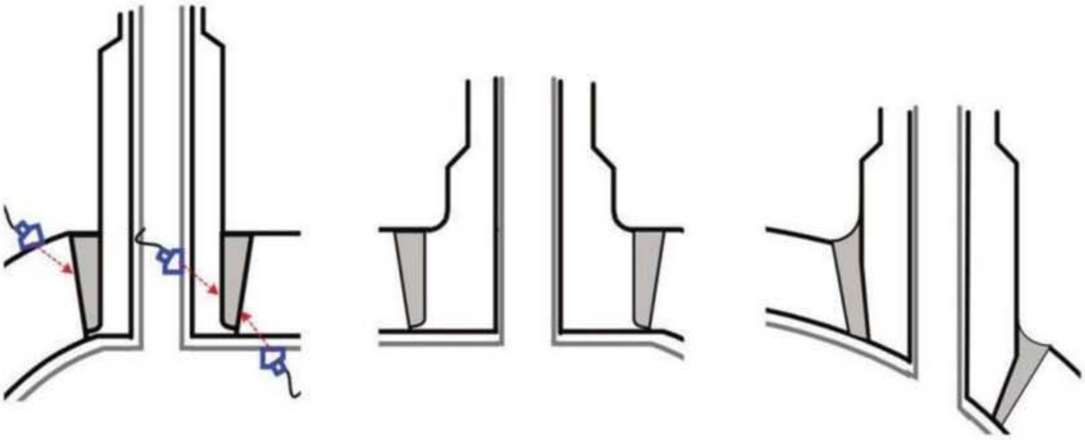
7.5 Reactor Component Inspection

7.5.1 General

The types and frequency of inspection of specific hydroprocessing reactor components depends on the expected susceptibility of a particular reactor to the damage mechanisms and locations discussed in Sections 5 and 6. This section discusses basic guidance on the application of specific NDE techniques to the relevant reactor components.

7.5.2 Reactor Nozzle Welds

Reactor nozzle weld examination is performed using manual SWUT or PAUT,although PAUT is currently preferred.Figure 16 provides additional details on UT examination of reactor vessel nozzles.The main issue with this type of examination is obtaining full coverage of the weld,as well as providing proper interpretation of the results,for the entire 360 degrees around the nozzle.PAUT may be the more valuable technique due to geometry issues associated with varying inner surfaces that may be dificult to discern using conventional SWUT. Hillside nozzleslocated on the side of the head (see panel c)of Figure 16)can be particularly difficultto examine and are prone to inaccurate calls due to geometry effects.If suspected flaws are located during an external examination,it is worthwhile to confirm the presence of the flaw from the internal surface,if possible.



A:Nozzle perpendicular to shell or head.This figure depicts a conventional set-in nozzle design where the shell is beveled for welding,but the nozzle neck is straight.Figure shows a typical approach for manual UT.

B:The nozzle is also perpendicular to shell or head in this figure,but the nozzle is a sweep-type with shoulders that permit radiography of the nozzle weld during fabrication. Approach for UT is similar to Figure 16,panel a).

C:Hillside or oblique nozzle is set into the head or shell at an angle.This configuration requires complex beveling and has difficult access for welding.This configuration is more likely to have weld defects and interpreting UT results ismore difficult.

Figure 16—UT Examination of Reactor Vessel Nozzles

The assessment results of multiple UT techniques capabilities and guidance on determination of appropriate techniques for inspections of reactor nozzle welds are presented in EWI,ASNT,and International Institute of Welding(IIW)publications.29.30Ultrasonicmodelingand simulation wereused with NDEengineering asaneffective way of determining that the internal flaws or damage in high risk geometries of heavy walled hydroprocessing reactors can be theoretically located and sized.After modeling and simulation work was completed,the best ultrasonic techniques were selected for experimental validation on specific damage in fabricated mockups with real flaws.Figure 17 shows an example PAUT of a typical nozzle perpendicular to the shell weld and imaging of a surface-breaking crack propagating beyond the cladding interface.

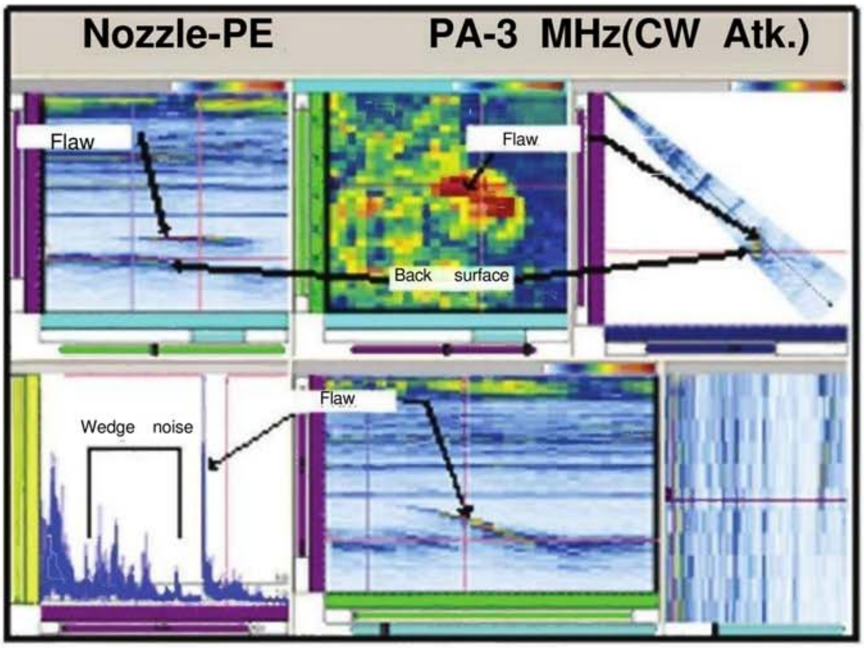


Figure 17—PAUT A-B-C-D-S-scan Imaging ofIndividual Crack-like Indication in a Nozzle Weld

7.5.3 Reactor Circumferential and Longitudinal Welds

If a volumetric inspection of reactor circumferential,longitudinal,or head meridian welds is requested during a turnaround,TOFD,PAUT,or a similar technique can be performed on the outside surface(after removing insulation in the area).Examination using UT methods will be most effective from the OD.ID inspection through the stainless overlay maycause interference and difficulty of interpretation.As noted in 7.5.2 above,a suspected flaw typically is confirmed from the opposite side,ifpossible.If inspection is only performed on the ID,on the 300 series SS overlay,the SWUT/PAUT inspection can be supplemented by a PT surface cracking inspection since surface-breaking defects can be missed with UT inspection techniques.

In some reactors,locating pressure welds may be difficult f they have been ground flush.Use of drawings and measurements may be needed to locate welds as well as understanding of weld bevel angles.Knowledge of joint details is important since defects will often follow the original joint preparation.Finally,familiarity with the reactor configuration and features,such as internal attachments,is critical to proper interpretation ofthe results.

Some companies have reported that they no longer routinely examine welds on reactor vessels for embedded flaws.As discussed in 6.4,industry experience has largely shown that embedded flaws in reactor main seam welds do not propagate during operating cycles.Efforts as part of the Aging Reactor JIP to grow embedded cracks in laboratory samples exposed to high temperature and high hydrogen partial pressure indicate that embedded cracks are not expected to grow even when exposed to conditions more severe than experienced by heavy wall reactor vessels in refinery service.There is one reported instance where embedded crack-like flaws in a reactor vessel appears to have propagated in service.However,there are owner/operators who continue to perform inspection ofwelds in heavy wall reactor vessels to detect any growth of embedded crack-like flaws.

7.5.4 Catalyst Bed Support Welds

One of the mostfrequently observed locations for cracking is on the ID surface at support ring and pad support attachment welds.This type of cracking typically is found by PT inspection.There is little concern for this type of cracking unless it propagates through the SS overlay and into the underlying base metal.If cracks are detected by PT,most owners perform either TOFD or PAUT from the outside to determine if the surface cracks propagate into the base metal.If repairs are not required,most owners do not grind on crack to determine crack depth because grinding can promote additional crack propagation and may also expose the base metal with resultant corrosion of the exposed area.

Optimized PAUT and TOFD procedures were validated and examiners were qualified for inspections of two typical catalyst bed support weld geometries.26-29 Figure 18 displays examples of PAUT and TOFD inspections imaging of typical crack-ike indication in catalyst bed supportweld geometry#1.The imaging validated a surface-breaking crack propagating beyond the cladding interface.Figure 19 displays examples of PAUT and TOFD inspections imaging of typical crack-like indication in catalyst bed support weld geometry #2.

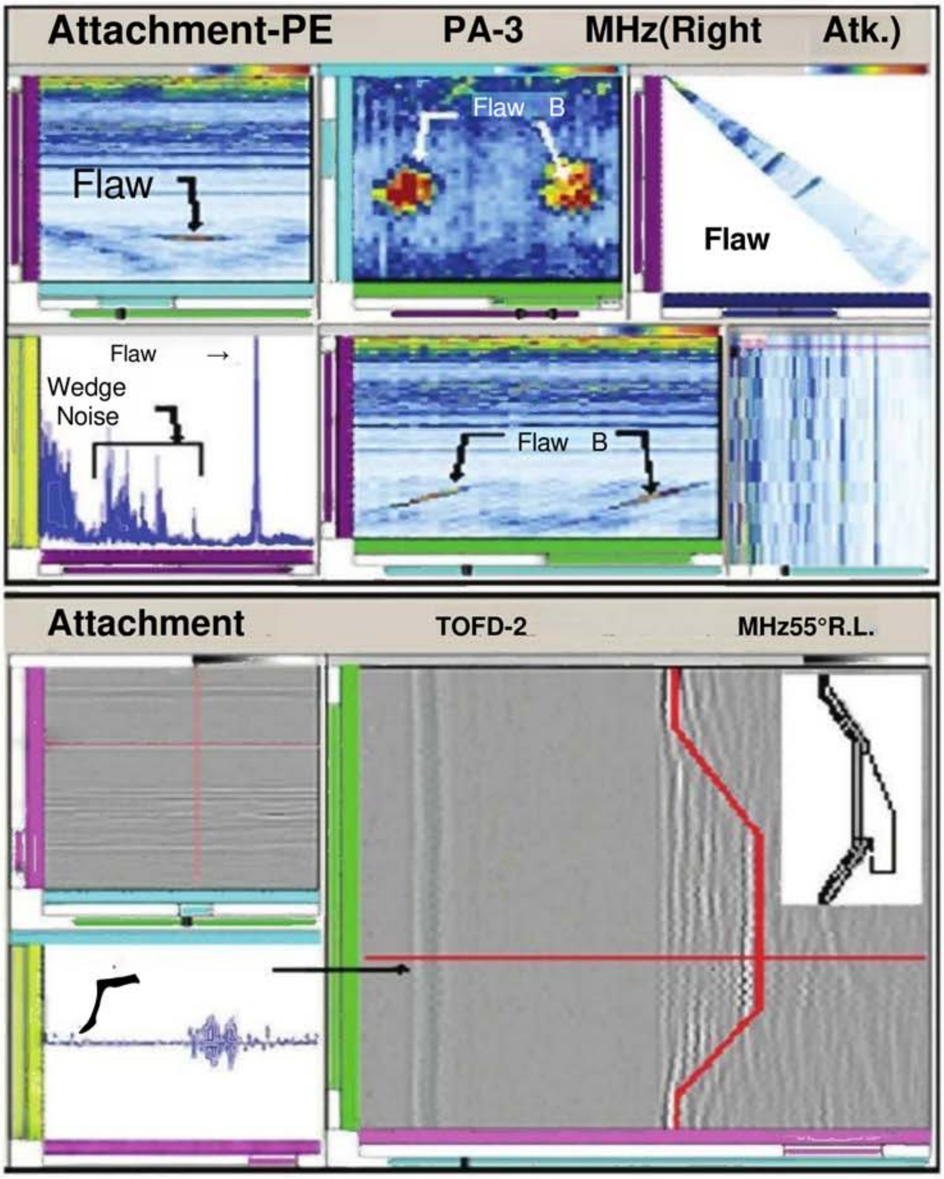
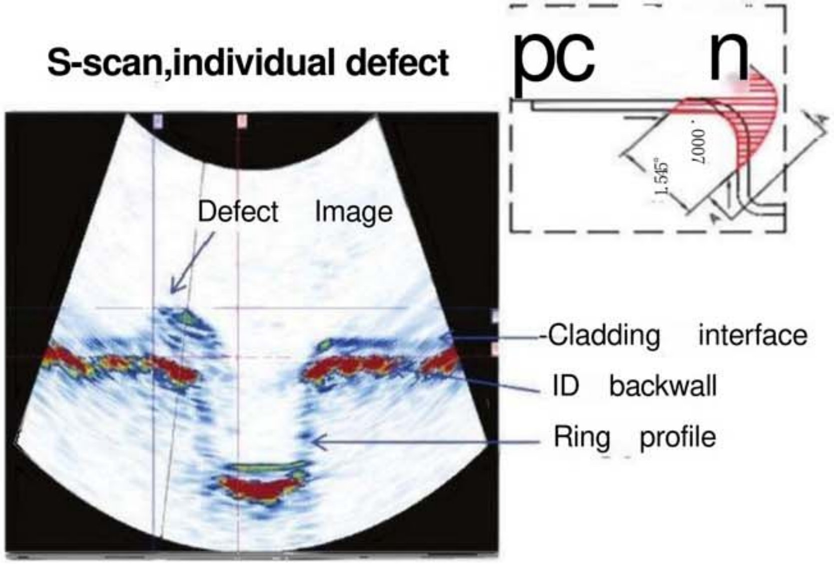
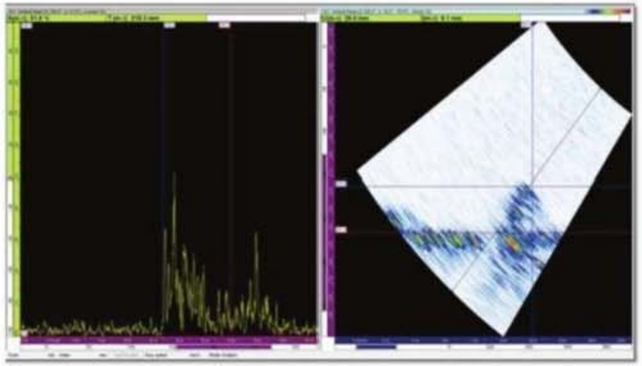


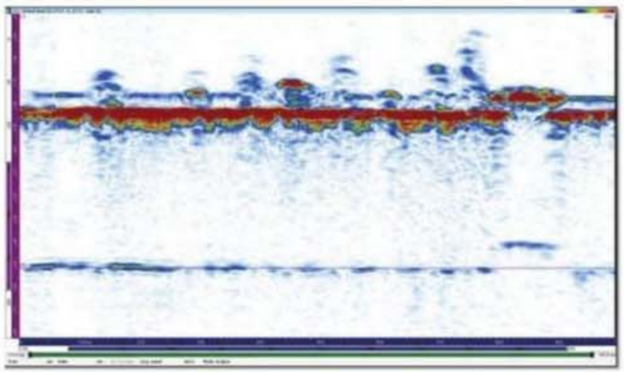
Figure 18—PAUT A-B-C-D-S-scan Imaging(Top)andTOFD A-D-scan Imaging(Bottom)of Typical Crack-like Indications in Catalyst Bed Support Weld Geometry #1



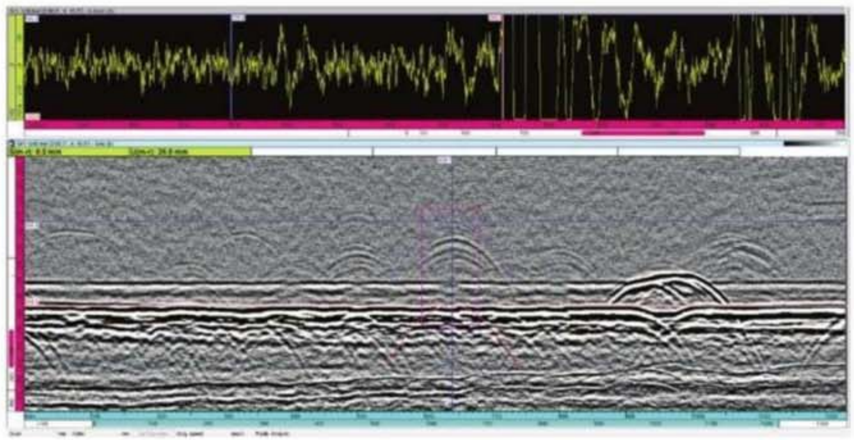
a)PAUT S-scan imaging of individual crack-like indication.



b)PAUTA-and S-scan imaging of individual crack-like indications.



c)PAUT D-scan imaging of multiple crack-like indications.



d)A-and D-scan TOFD imaging of multiple crack-like indications.

Figure 19-PAUT and TOFD Inspections of Typical Catalyst Bed Support Weld Geometry#2 and Imaging of Surface-Breaking Cracks Propagating Beyond the Cladding Interface

Typically,all internal attachment welds are full penetration welds as required by API 934-A and API 934-C.

7.5.5 Ring Joint Gasket Grooves

The grooves are inspected for cracks using PT if the reactor has a 300 series SS weld overlay over underlying base metal.If the reactor does not have a 300 series SS weld overlay,MT examination of the ring groove could be performed.There have been several reports of these cracks propagating deeply (such as 25mm to 50 mm [1 in.to 2 in.])into the underlying base metal.When PT examination uncovers cracks in the ring joint gasket groove,UT examination is used to determine the depth of cracking.This typically involves straight-beam UT examination from the inside surface of the nozzle or phased-array UT from the external surfaces.

7.5.6 Bottom Outlet Collector Ring Support

PT inspection is used to find cracks in the overlay at the bottom collector ring support.These rings are used to attach the internal outlet collector or catalyst support cone(also sometimes called the "elephant stool")to the bottom head.This type of cracking initiates atthe surface at the corner of welds between the collector ring support and the bottom head on the reactor vessel.There is little concern for this type of cracking unless it propagates into the underlying base metal.It is difficult to inspect from the outside surface in this area ofthe reactor so that most owners do not use TOFD or PAUT to determine if cracking has propagated into the underlying base metal. Typically,these cracks are ground out,and if they reach the Cr-Mo base metal,the area is repaired using one of the techniques discussed in Section 9(as long as the repairs meet the limits given).

7.5.7 Quench Nozzles

As discussed in 6.7,quench nozzles are prone to cracking,especially if they are not equipped with a distributor pipe that runs through the nozzle and minimizes the thermal stresses generated by the cool hydrogen quench gas.Cracking found in quench nozzles is typically detected by internal examination using PT if the vessel and nozzle are lined with 300 series SS,or MT if the vessel and nozzle are unlined.

7.5.8 SkirtAttachmentWelds

Owners have not reported a need to inspect the area around the skirt attachment to heavy wall reactor vessels. In most modern heavy wall reactor vessels,the skirt-to-shell connection is radiused either as a forged ring or as a weld buildup to reduce the geometric stress concentration where the skirt connects to the reactor vessel.Other designs use an insulated hot box at the skirt attachment welds to minimize temperature gradients and stresses. These design details have not displayed cracks and typically are not inspected on the outside surface.

If done,inspection typically would include MT and PAUT.PAUT of a weld at the skirt attachment can be a challenge due to the geometry.Some owners have used the mock-up developed during initial fabrication to qualify a PAUT inspection procedure.

7.5.9 Thermowells with Removable Thermocouple Bundles

In some older reactor designs,thermocouple bundles are included in a SS pipe or tube (typically called a thermowell)that runs from the top to the bottom of the reactor.The most frequent problem reported with these thermowells is chloride SCC from the inside surface.This has been observed shortly after start-up from water that collects in the thermowell during the shutdown period.During a shutdown,it is important to cover the thermowells,so that water (from rain or condensation)does not collect in them,in order to avoid chloride SCC during startup.Some operators have reported removing top-entry thermowells from reactors during downtimes to prevent water from getting inside.

Newer reactor designs use side-entry thermowell bundles,and they have proven to be less susceptible to chloride SCC because water is less likely to collect in them during shutdown periods.However,these thermowells are not immune to damage,such as binding or fatigue,and are typically periodically inspected for cracking or other signs of degradation.

7.5.10 Weld Overlay

Weld overlay cracking typically is inspected using PT.It may be possible to use eddy current testing,but this is not common.Inspection is typically focused on high-stress areas such as attachment welds of tray support rings,beam support pads,and outlet collector ring supports,as discussed in the previous subsections.Some owners report that they inspect high-stress areas on the weld overlay during each catalyst change-out if these areas have shown a tendency for cracking in the past.If PT inspection at critical areas finds cracking,inspection typically would be expanded to other potentially suspect areas.

Conventional cracking mechanisms affecting 300 series SS,such as chloride SCC,polythionic acid SCC,or sigma embritlement,do not affect the Cr-Mo base material.Hence,cracks in 300 series SS overlay often stop at the Cr-Mo base material.However,even with this understanding,it may be necessary to confirm the crack depth using SWUT or PAUT in situations where there is any doubt or concern.Ultrasonic inspection of overlay cracks is sometimes done from the reactor interior surface due to difficulties in accessing the exterior,i.e.the need for scaffolding and insulation removal;however,this requires a good procedure and skilled examiner.If it is determined thatacrack propagates beyond the overlay and into the base metal,ultrasonic sizing from outside the reactor wil likely be needed.Typically,cracking in SS weld overlay or cladding that has propagated to the base metal is monitored from the OD surface using TOFD or PAUT.

Smoothing of the overlay weld ripples bygrinding may be necessary to prepare the surface for SWUT or PAUT inspection from the inside surface.Care must be exercised not to remove too much of the weld overlay when preparing a surface for UT examination.

Generally,cracks in 300 series SS weld overlay are not ground out to determine the depth of the crack.After cooling to ambient temperature from high-temperature service,the 300 series SS overlay is under a high residual tensile stress that typically exceeds its yield strength.Grinding on a crackmay promote further propagation to the interface with the underlying Cr-Mo steel.

To properly assess an overlay crack,the thickness of the overlay must be known.There are straight-beam ultrasonic techniques that can locate the overlay interface and measure the overlay thickness.Overlay thickness measurement from the vessel interior surface is dificult to perform,so typically overlay thickness is measured from the exterior surface,requiring the removal of insulation.One caution is thatusing originalfabrication drawings with a specified minimum overlay thickness is frequently inaccurate.Hence,accurate thickness measurement of overlay is important when evaluating whether the measured crack depth indicates the crack has grown beyond the SS overlay and into the underlying Cr-Mo steel.

Overlay disbonding inspection may be necessary for specific reactors with a history of disbonding or if"sister" reactors have experienced disbonding.Disbonding inspections are typically conducted using manual UT thickness measuring devices orAUT.No reactors have suffered a loss of containment failure due to disbonding; however,disbonding inspections are conducted to monitor disbonding progression because more attention may be needed if the disbonding approaches a critical structure such as a support or nozzle.

7.6 Frequency and Extent of Inspection Performed on Reactor Vessels

The frequency and extent of inspection performed on a reactor vessel depends on several factors.Typically, an inspection plan for a reactorvessel is developed before each planned shutdown and considers the following factors:

—inspection results from previous shutdowns;

—fitness-for-service assessments performed in the past;

—repairs performed in the past;

—operating records during the run length and reports of any excursions/upsets.

The following is a list of possible preparation steps for planning nondestructive testing(NDT)of heavy wall reactors when volumetric inspection of major welds is being considered:

—Review of the original fabrication drawings to determine key locations for inspection.Special points of interest are pressure boundary weld preparation details for circumferential,axial,nozzle,and possibly head sectional welds.Consider that the head-to-shell weld geometry may be different than the shell-to-shell can section welds.Head-to-shell welds may require special UT setups.If heads are hemispherical,planning for the segmental weld seams could be different from the rest of the welds.

—Review of the inspection history.Identify where flaws have been detected in the past.Identify any repairs performed in the past.Prioritize locations for inspection based on prior history.

—Perform an industry-widesearch for papers written onthetype of reactors in similar service fordocumentation of noted problem points.Determine the design and license holder of the process and pressure equipment designer for possible bulletins regarding problem points.

-Use fracture mechanics guidance included in API Standard 579,Part 9 to determine what size flaws are targeted for detection during the intended inspection.Calculate flaw sizes for circumferential,axial,and nozzle welds.

—Plan the inspection protocol for all the areas to be inspected.

-Create a roll-out style inspection drawing of the reactor.Color code the welds based on type of inspection(s).

—Establish a data point for all measurements to be recorded.Use accurate location measuring equipment such as flat metal tape measures long enough to span the entire length or circumference of the pressure vessel.Careful physical measurement and location markings typically are documented.UTscans are plotted according to physical dimensions recorded on the pressure vessel and inspection map.

—Determine the type of surface preparation needed for the inspection to be applied.Heavy wall pressure vessels operating at elevated temperature could require extensive surface preparation on the OD surface due to heavy mill scale and hardened exterior surface.Abrasive blasting,needle gun,or sanding can be considered for surface preparation on the OD surface.The examination surface normally is similar to the surface on the calibration standard.A bright,smooth surface is optimal for quality UT examination.Width of surface preparation required adjacent to the welds can be determined by using UT beam tool plotting software.

-Some heavy wall pressure vessels welds are ground flush on the OD.Determining the location of the welds can be performed using contact eddy current testing.Smoothing the exterior surface and etching with 5% Nital acid is also effective for determining where the weld seams are located.

-Surface preparation on the ID includes removal of process buildup.Forreactors with an austenitic SS layer on the ID,steam/water blasting is the preferred cleaning method prior to penetranttesting.Abrasive blasting also has been used,and soda blasting is a good choice.If abrasive blasting is applied,a high-pressure steam/ water cleaning typically follows to remove blast media imbedded in cracks.

—Determine naming conventions for welds,locations,data points,direction of scans,etc.Also,determine what anomalies are to be recorded and their naming conventions.

—For UT inspection including SWUT,PAUT,and TOFD,a mock-up made of material with the same velocity and attenuation as the pressure vessel to be inspected frequently is used for calibration and performance demonstration.Place reflectors in the mock-up that are coordinated with critical flaw size calculations.The mock-up(s)have the same geometry as the location to be inspected.The inspection service provider typically qualifies the inspection procedure,equipment,and personnel performing the examination of the pressure vessel prior to,during,and at closure of the examination.

—Determine the inspection results to be reported and the reporting formats.

—A beam tool plotting software normally is used to determine coverage of the area to be inspected based on weld and heat-affected zone configuration.Determine the type of UT,transducers,positioning of the transducers,and how many scans will be needed to inspect the selected areas,especially to achieve full coverage through the entire wall or weld thickness.

—Types of UT equipment including systems,scanners,probes,and scan speed normally are all predetermined in advance of the scheduled inspection.Performance testing of the selected assemblies for UT inspection is essential for an effective inspection.

—Include examination for both longitudinal and transverse cracks in all welds being inspected.

—Include examination ofinsulation support ring fllet welds.

—Use positive materials identification(PMI)on all new welds and if there are any questions on the material verification of previously completed welds.

—Identify if the reactor has a closure weld that was heat treated separately from the furnace heat treatments given to the other welds.If inspection of the major welds is being done,inspect this weld especially thoroughly.

—For internal attachment welds inspected from the OD surface,such as catalyst bed supports,the attachment weld locations need to be identified from the OD of the vessel,so that UTexamination from the OD surface can be performed around the entire circumference.

In older reactor vessels,SS weld overlay typically has poor ductility,resulting in a higher likelihood of cladding cracks at support rings and pads.Additionally,high heat input weld procedures used to deposit weld overlay in older reactors has resulted in a greater tendency for disbonding cracks.As a result,these forms of cracking typically need to be monitored in older heavy wall reactor vessels.The newer vanadium-enhanced grades of Cr-Mo steels have shown superior resistance to disbonding.

Once cracking in the overlay or cladding is observed,a plan often is established to include inspection from the outside surface to ensure any cracks in the overlay or cladding do not propagate into the underlying base metal and through the reactor vessel wall.Inspection from the outside surface typically is performed using TOFD and/or PAUT procedures as discussed earlier.If the first few inspections confirm that the cracks in SS overlay or cladding do not propagate into the underlying base metal(as is typical),the frequency of this inspection fromthe outside surface is usually reduced.

7.7 Inspection for High-temperature Hydrogen Attack

Heavy wall Cr-Mo reactor vessels typically operate wel below operating conditions of temperature and hydrogen partial pressure expected to cause HTHA as indicated by the API 941 Nelson curves.As a result,inspection for HTHA typically is not performed on Cr-Mo reactor vessels.However,heavy wall reactor vessels fabricated from C-0.5Mo can operate at conditions of temperature and hydrogen partial pressure where HTHA damage can occur.As a result,C-0.5Mo heavy wall reactor vessels are frequently inspected for HTHA.As discussed in API941,there are several prescribed NDE technologies for detecting HTHA.This includes advanced ultrasonic backscatter testing(AUBT),PAUT,TOFD,and field metallography replication (FMR).

8 Fitness-for-Service on Heavy Wall Reactor Vessels

8.1 General

Once damage as described in Sections 5,6,and 7 is observed,an FFS assessment may be needed to determine if the reactor requires a repair.API 579-1/ASME FFS-1 is the recommended standard for performing an FFS assessment.API 579-1/ASME FFS-1 is organized in modules (i.e.Parts)based on the damage mechanism and type of flaw.Characterization of the damage or flaw is performed in accordance with the respective Part of API 579-1/ASME FFS-1.Additional data requirements for an FFS assessment are summarized in WRC Bulletin 584.

In addition to performing an FFS assessment on damage found during an inspection,some owners of heavy wall reactor vessels perform hypothetical FFS calculations prior to performing an inspection so that once damage is found it can be immediatelyevaluated to determine if a repair is required.An upfront FFS assessment generally is found to reduce the time necessary to make a judgement on whether a repair is needed and the extent of the repair.

8.2 FFS Performed on Crack-like Flaws

As indicated in Section 5,most of the observed damage in heavy wall reactor vessels is in the form of crack-ike flaws,and most of these occur in the 300 series SS overlay or cladding,butthey may propagate into the Cr-Mo reactor vessel wall.In addition,some embedded crack-like indications have been found in Cr-Mo welds.Some of these cracks are believed to be lack of sidewall fusion,which can occur during fabrication.Embedded crack-like flaws are more commonly observed in older reactor vessels that originally were inspected during fabrication using only RT examination,which has a limited ability to detect crack-like flaws.In general,embedded flaws are not believed to grow during service atthe operating conditions.However,embedded flaws may grow due to hydrogen embrittlement during shutdown or startup due to molecular hydrogen pressure buildup and/or thermal stress effects.

Once found,crack-like flaws,both embedded and surface breaking,can be evaluated through an FFSassessment and those deemed acceptable are generally notinspected online.However,many owners reinspect known flaws during subsequent shutdowns to look for any incremental growth that may have occurred since the last outage.

FFS of observed crack-like flaws can be performed using the guidance in Part 9 of API 579-1/ASME FFS-1, *Assessment of Crack-like Flaws.*

When performing an assessment for crack-like flaws,it is necessary to assign a fracture toughness,which provides a limit for fast fracture.In the case of heavy wall reactors,it is necessary to ensure that the fracture toughness chosen for the flaw assessment is consistent with that used to develop a minimum pressurization temperature(MPT).As for the establishment of the equipmentMPT,the fracture toughness usedin the assessment of crack-ike flaws typically considers the possible effects of temper embrittlement and hydrogen on fast fracture. Additionally,an MPT assessment needs to consider the effect of hydrogen on slow stable crack growth.

8.2.1 FFS on Surface-breaking Crack-like Flaws

Experience shows that surface-breaking cracks initiating in the 300 series SS overlay or cladding seldom propagate into the underlying Cr-Mo vessel shell.However,regardless of whether cracks exist only in the stainless steel orpropagate into the base metal,the FFS rules ofAPI 579-1/ASME FFS-1,Part9 can be applied.

Residual stress in the SS overlay or cladding will affect the crack stress intensity.In general,overlay/cladding residual stress results from the thermal expansion mismatch between the austenitic SS and the Cr-Mo base metal. At the PWHT temperature,the thermal expansion of the overlay/cladding is much more than that of the base metal.The greater thickness of the base metal (generally 10-20 times the overlay/cladding thickness or more) restrains the overlay/cladding and causes it to yield in compression.During cooling from the PWHT temperature, elastic recovery will occur until some neutral,stress-free temperature is reached.Various references state this neutral temperature as between 260℃ and 316℃(500°F and 600 °F).Below this temperature,the thermal contraction of the cladding will place it in tension.17 Operation of hydroprocessing equipment is usually above this neutral temperature,such that the overlay/cladding is in compression,with the base metal in tension during operation.Upon cooling after operation,the SS cladding will experience a residual tensile stress that exceeds the yield stress.Given that the thickness of the base metal is many times greater than that of the overlay/cladding, any tensile residual stress in the base metal from the overlay/cladding willact on a thin layer of the base metal at the cladding interface.

Additional information on the stresses generated in the cladding from the thermal cycling that occurs during startup and shutdown appears in Reference [9]and guidance for estimating the cladding residual stress for use in an FFS assessment appears in WRC 56217 and PVP2020-21583301.If the calculated stress intensity for a small surface-breaking crack in a heavy wall reactor vessel is low,an API579-1/ASME FFS-1,Part 9 assessment may often demonstrate that the likelihood for continued crack growth is also low,even when considering the effects of both temper embrittlement and hydrogen embritlement.This is consistent with the industry's typical experience that once cracks propagate through the stainless steel cladding/overlay,they often arrest and do not propagate deeper into the vessel wall.18,19 This generalization is contingent on controlling heating and cooling rates to avoid excessive stresses from thermal gradients and adhering to the vessel's MPT envelope to ensure adequate toughness.19]

A few historical exceptions to this experience have been reported.These generally involved a few exacerbating factors which may not affect more modern heavy wall reactor vessels.One case involved a highly sigmatized weld overlaythat readily cracked,generating multiple cracks which slowly propagated into the Cr-Mo base metal. 18.19 The Cr-Mo base metal,which had a measured tensile strength of~650 MPa(95 ksi),was severely temper embritled.The combined effects of temper embrittlement and hydrogen embrittlement of this Cr-Mo steel made it susceptible to continued crack growth through the underlying Cr-Mo steel beneath the 300 series SS weld overlay.Cracking into the Cr-Mo base metal from the ID surface penetrated the reactor shell to depths of up to ~22 mm(7/8 in.)into the 144 mm (511/16 in.)Cr-Mo reactor shell It was concluded,based on examination of samples after several shutdowns,that short crack extensions had occurred during each shutdown period driven by a combination of thermal effects from cladding and a sulfide scale that imposed a wedge-opening load.The area of short crack extension nearest the crack tip,which was attributed to the mostrecent shutdown prior to the extraction of metallurgical samples,was found to contain no sulfide scale on the surface,as compared with the earlier cracking which did contain a sulfide scale on the surface.

Other instances of surface-breaking cracks that propagated into the Cr-Mo base metal have been reported on steels with tensile strengths above the modern maximum limits of 690 MPa(100 ksi)for conventional steels and 760 MPa(110 ksi)for advanced steels.Cracking associated with internal attachments of reactors built in the late 1960s from high strength(approx.830 MPa [120 ksi]tensile strength)2/$_4$Cr-1Mo has been documented. W Cracking due to severe sigma phase embritlement of 300 series SS weld overlay at internal supports and support cone-to-shell connections has been reported.20 Ring joint groove cracks (discussed below)and other cracks at areas of very high stress intensity have been found to propagate into the Cr-Mo base metal.

Cracking at the corners of RTJ flange ring grooves,as discussed in 6.8,is frequently encountered.Contact loads between the gasket and ring groove create locally high stresses in the groove corners,commonly leading to crack formation.Similar to ratcheting,successive joint assemblies result in progressively larger crack opening widths,which will inevitably promote crack growth beyond the overlay and into the Cr-Mo flange base material.The loading on RTJgroove cracks is almost exclusively strain limited in nature,and little primary stress exists to promote unarrested fracture(i.e.the cracks will grow due to repeated gasket loading,but often will not grow due to pressure loading).Extensive industry experience and repeated FFS assessments in industry have demonstrated that RTJ groove cracking can often be considered temporarily fit for service—but all known studies acknowledge that RTJ groove cracks will continue to grow under repeated loading with RTJ gaskets,and therefore will eventually require repair.

As discussed in 6.8,many owners have modified the flanges or flange faces to change the original ring joint configuration to a raised face using a flat compressed gasket without filling in the ring groove.The raised face flanges are not prone to RTJ cracking.Additional flange repair guidance(including guidance on flange conversion)is provided ASME PCC-2 Article 305.Raised face flanges are also now commonly used for new reactor fabrication rather than RTJ flanges.

8.2.2 FFS of Embedded Crack-like Flaws

Embedded crack-ike flaws are observed more frequently in welds in older vessels that are in-service inspected using a UT technique,but originally during fabrication were only inspected using RT.Many embedded crack-like flaws are believed to be the result of lack of sidewall fusion.An assessment of embedded flaws using the guidance in Part 9 of API 579-1/ASME FFS-1 and only considering applied and residual welding stresses often shows that embedded crack-like flaws are acceptable and do not pose a risk forfast brittle fracture orslow stable crack propagation,provided that proper operational startup and shutdown procedures,including the MPT curve, are adhered to.

However,an increase of stress intensity at an embedded flaw resulting from the potential buildup of molecular hydrogen gas during shutdown of a reactor vessel indicates that crack propagation during a shutdown is a distinct possibility;133 hence,this was studied by the Aging Reactor JIP.The metallurgical analysis for one example,described in 6.4,concluded that an embedded flaw had grown during shutdowns.13 As a result of the concern for growth of embedded flaws during shutdowns,two separate laboratory testing programsl¹6 were conducted on Cr-Mo weld samples with embedded crack-like flaws.These samples were exposed to conditions of high temperature and high hydrogenpressure more severe than experienced by a heavy wall reactor vessel in hydroprocessing service.It was expected that the molecular hydrogen pressure buildup in the embedded crack-ike flaws in these test samples,after the samples were cooled down from the high temperature and hydrogen pressure exposure,would be higher than could occur in an actual embedded flaw during shutdown of a heavy wall reactor in hydroprocessing service.However,in both laboratory testing programs,the embedded crack-ike flaws did not propagate after multiple exposures to the severe conditions.These laboratory results are consistent with the majority of field experience where,over time,multiple ultrasonic examinations of embedded crack-ike flaws have shown that they do not propagate.Despite these test results and the majority of field experience, manyowners continue to periodically inspect known embedded flaws to verify that growth is not occurring.

8.3 FFS Performed on Ground-out Cracks

Some owners of heavy wall vessels choose to grind out cracks before they can propagate deeply into the vessel wall.In some cases,this has been an opportunity to remove samples containing cracks for metallurgical investigation.In the ground-out areas orareas where boat samples or scoop samples are extracted,the resulting excavation is ground smooth and can be qualified by API 579-1/ASME FFS-1,Part 4(Assessment of General Metal Loss)or API 579-1/ASME FFS-1,Part 5(Assessment of Local Metal Loss).A ground-out region of metal loss is typically easier to characterize,monitor,and qualify with an FFS assessment as opposed to a crack. Additionally,it provides an opportunity to perform a repair as outlined in Section 9 thatdoes not require a PWHT and provides protection from sulfidation corrosion.

As discussed below in 8.4,a heavy wall reactor vessel can tolerate very deep and long blended grooves while still meeting the acceptance criteria contained in Part 5 of API 579-1/ASME FFS-1.This is due to the pressure-retaining supportprovided by the large volume of metal surrounding a groove in heavy wall vessels.However,the resulting groove frequently exposes the Cr-Mo steel below the cladding.Exposed Cr-Mo steel can experience high-temperature sulfidation corrosion during operation and may need to be protected,although opinions on the need or timing for this vary depending on the experience of the owner.Methods to protect exposed Cr-Mo steel at the bottom of a blended groove are discussed in Sections 9 and 10.

8.4 FFS Performed Due to Corrosion or Other Metal Loss

Metal loss typically is not observed on heavy wall reactor vessels.External corrosion is highly unlikely because reactor vessels operate at temperatures well above the range where corrosion under insulation(CUI)occurs.On the inside surface,reactors exposed to a corrosive environment that contains H_2S are overlayed or clad with a 3mm to 6mm(e in.to /in.)thick layer of 300 series SS,which is highly resistant to sulfidation corrosion from H_2S.However,300 series SS is not immune to sulfidation corrosion,so wall loss is possible on cladding and internals (particularly screens),and especially on older weld overlays that may have a diluted chromium level. In rarely reported instances,localized metal loss has been observed in reactors where a fluidized or ebullating catalyst bed resulted in erosion.

In the event that metal loss occurs either on the inside or outside surface of the reactor vessel,the guidance contained in API 579-1/ASME FFS-1 can be used determine if a repair is needed.Part 4 of API 579-1/ASME *FFS-1(Assessment of General Metal Loss)is used for the assessment of general corrosion and is typically more* conservativethan Part 5,which is used for the assessment of localized corrosion.In general,heavy wall vessels can tolerate a significant amount of metal loss and still meet the criteria ofAPI 579-1/ASME FFS-1,becausethe surrounding uncorroded portions of the vessel provide pressure-retaining support for the corroded portions.211

9 Weld Repairs Not Requiring Postweld Heat Treatment

9.1 Repairs to Shallow Cracks in Stainless Steel Weld Overlay and Cladding

In the situation where cracks are confined to the internal SS overlay or cladding,it may be possible to grind out the crack and fill the resultant groove with matching SS welding material without affecting the underlying Cr-Mo steel.In general,it is possible to weld on the SS without affecting the underlying Cr-Mo steel as long as the remaining stainless steel is at least 4.5 mm(3/16 in.)thick when a conventional welding procedure with a typical welding consumable diameter is used.This thickness is based on API Recommended Practice 582,and thinner layers can be used if the weld procedure qualification can demonstrate that no new HAZ is formed in the base metal.There is successful experience qualifying a low heat input welding procedure(typically using a gas tungsten arc welding [GTAW]process)with as ittle as 1 mm(0.04 in.)SS overlay or cladding thickness.

Rigorous testing of the procedure and welder is typically performed and involves the use of small diameter electrodes to minimize the heat input.Examination of the weld coupons involves laboratory hardness testing of the Cr-Mo steel immediately below the overlay or cladding and performing metallography on cross sectional samples to demonstrate that the underlying Cr-Mo steel is not adversely affected by the weld repair.Table 4 provides typical shielded metal arc welding(SMAW)welding parameters used in procedures from one owner depending on the remaining overlay or cladding thickness.

Table 4—Typical SMAW Welding Parameters for Welding over SS Cladding or Overlay and Avoiding Formation of a New HAZ in the Underlying Base Metal

Clad or Overlay Thickness	1G and 4G Positions		2G and 3G Positions	
	Maximum Current (amps)	Maximum Heat Input(J/in)	Maximum Current(amps)	Maximum Heat Input(J/in.)
>3.2 mm(1/8 in.)	150	18,000	90	18,000
1.6 to 3.2 mm(1/16 to 1/8 in.)	90	12,000	60	12,000
<1.6 mm(1/16 in.)	SMAW generally is not performed without PWHT.Automated low heat input welding process such as GMAW have been used on remaining thickness as low as 1 mm with proper qualification.			

Difficultyhas often been encountered when attempting to weld toexisting cladding or overlay.Overlay or cladding back-welds may be embrittled due to sigma phase formation from previous PWHT.For example,welding onto service-aged type 347 SS cladding or overlay can prove especially difficult and may require the use of a higher ductility fller metal,such as E/ER 309L,prior to depositing new E/ER 347.Deposition of a butter layer of the higher ductility filler metal can aid in the overall weldability of repair joints.

9.2 Repairs to Ground-out Cracks in Stainless Steel Overlay that Expose the Cr-Mo Steel

In some situations,grinding out the overlay crack will expose the Cr-Mo base metal,either because the cracks have gone completely through the overlay and stopped at the base metal,or because the cracks have grown to the base metal while grinding.Also,on rare occasions,cracks in stainless steel overlay can propagate into the Cr-Mo steel shell.Exposure of the Cr-Mo base metal can be verified by spraying it with a copper sulfate solution. Even if the crack has penetrated the base metal,an FFS assessment of the blended groove often will show that it is acceptable per the guidance in Parts 4 or 5 of API 579-1/ASME FFS-1.

However,exposed Cr-Mo steel may need protection against sulfidation.The JIP on Aging Reactors identified four repair procedures without PWHT that can be used to protect the exposed Cr-Mo steel from the sulfiding environment.In each of thefour procedures,welding is confined to the austenitic stainless steel without affecting the underlying Cr-Mo steel,thus eliminating the need for PWHT.Each of these approaches is discussed below.

Several owners have had positive,long-term experience even when they did not protect the exposedCr-Mo steel from sulfidation.These companies have generally found that only minimal metal loss has occurred.Each owner will need to decide,based on their experience,the experience of others,and/or subject matter expert advice, whether or not to protect exposed Cr-Mo base metal from high-temperature sulfidation at operating conditions.

Some published repair procedures have allowed welding of 300 series SS overlay directly to the Cr-Mo base metal without subsequent PWHT,with careful restrictions.In such cases,welding was only allowed when less than 0.8 mm(1/32 in.)of Cr-Mo had been removed in a very narrow area.Additional restrictions on preheat, interpass temperature,and heat input were required in this case.201 Prior to the execution of such a repair,a thorough evaluation of the metallurgical implications and localjurisdictional requirements,including evaluation of welded test coupons,is warranted.See 9.5 for a more complete discussion of this topic.

9.2.1 Method 1—Application of a Thin Strip of Stainless Steel

Figure 20 is a sketch of a ground-out area where the Cr-Mo steel beneath the stainless steel is exposed.The ground-out area is covered with a thin strip of stainless steel that is welded to the remaining intact stainless steel overlay.This repair strategy has been used in clad or overlayed reactor vessels with surface-breaking cracks at the toe of fllet attachment welds on internal supports and in weld overlay in areas adjacent to internal supports.

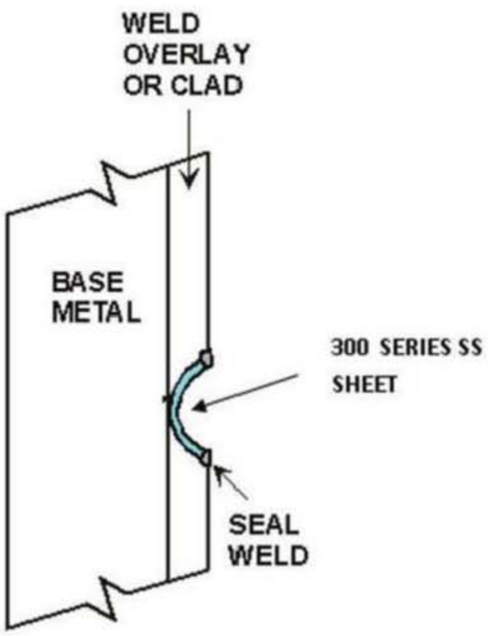


Figure 20—Applying a Thin Strip of 300 Series SS to Blend Ground Groove to Protect Exposed Cr-Mo Steel from Sulfidation

The SS strip,which conforms to the shape of the blended groove,is commonly called a "taco patch"because of its shape.Some dificulty can be encountered when trying to shape a patch to match the double contour of an excavation.In some cases where additional corrosion of the underlying base metal could be tolerated,flat patches have been used without flling the void underneath.Experience has generally shown that the amount of corrosion that occurs on the base metal beneath the patch is not a significant concern.

The fllet weld between the SS strip and the cladding is made using a matching SS welding consumable,and the weld is kept at least 4.5 mm(3/16 in.)from the Cr-Mo steel base metal to avoid heating the Cr-Mo to a temperature where an HAZ would be created and PWHT would be required.A small gap is left in the fllet weld to allow for venting of the space under the strip during future shutdowns and to avoid buildup of molecular hydrogen gas.

The integrity of the SS patch protecting the Cr-Mo steel is normally checked when first applied and periodically when the reactor is shut down for maintenance.When first installed,the patch attachment welds are typically inspected using PT.Repair patches are often visually and/or PT inspected during subsequent turnarounds to insure they are still ntact.In addition to inspection of the patches from the inside surface of the reactor vessel, some operators perform TOFD or PAUTexamination from the outside surface of the reactor to insure no through-wall cracks are propagating from the groove and no significant metal loss is occurring due to sulfidation.

9.2.2 Method 2—Filling of the Groove with Dense Refractory Covered by a Stainless Steel Strip

Figure 21 shows a repair strategy that involves flling the excavation with a dense refractory and then covering the filled groove with a SS strip or welding the SS strip and thenflling the void with refractory.This repair strategy typically has been used in cases where cracking into the Cr-Mo shell as been deep and/or the ground-out area extensive(but acceptable as per FFS).Examples have occurred at the grind-out of cracking associated with the filletweld attaching the outlet collector or catalyst support cone(also sometimes called the"elephant stool")to the bottom head of the reactor.

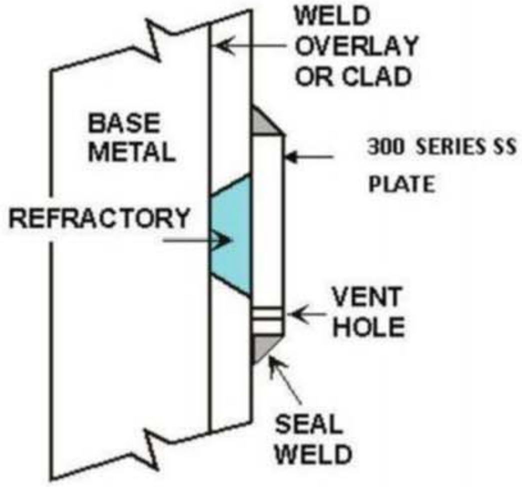


Figure 21—Filling the Blend Ground Groove with a Dense Refractory and Covering It with a Stainless Steel Strip to Protect Exposed Cr-Mo Steel from Sulfidation

The SS strip is fillet welded to the SS overlay or cladding just as with the patch described in 9.2.1.If the strip is welded first,a hole is tapped into the stainless sheet and a fitting inserted,through which a pumpable refractory can be added to completely fill the void.The fitting is removed and either the drilled hole or a gap in the fllet weld is left open to provide a vent,so that any hydrogen gas generated in the filled groove can vent to the process stream.A dense,high alumina pumpable refractory is typically used to minimize the hot,sulfidizing process environment from reaching the exposed Cr-Mo surface.

The Cr-Mo steel at the base of the groove can be inspected during turnarounds for reinitiation of cracking or corrosion using the external techniques described in Section 7,such as TOFD and/or PAUT.

9.2.3 Method 3—Filling the Groove with Stainless Steel Wire and Welding Over It

Figure 22 shows a repair that involves filing the groove with unwelded SS welding wire and then welding over it with a SS welding consumable that matches the overlay.This approach is commonly referred to as slugging.This repair strategy has been used in clad or overlayed reactor vessels with surface-breaking cracks at the toe of fillet attachment welds on internal supports and to repair weld overlay where it was removed from internal supports. It also has been particularly useful in the repair of cracking in the base of RTJ grooves and has been used in situations where an RTJ groove face was converted to a raised face flange,to use a spiral-wound or groove metal gasket,and to eliminate the ring gasket.

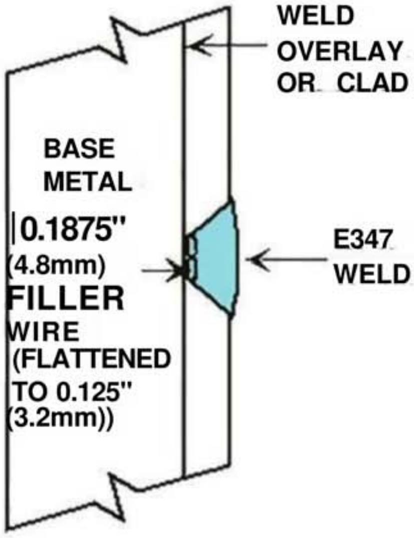


Figure 22—Protecting the Exposed Cr-Mo Steel from Sulfidation by Filling the Groove with SS Welding Wire and Welding Over It with an SS Consumable While Avoiding Welding on the Cr-Mo Steel

In these situations,any excavation exposing the Cr-Mo base metal is filled with tightly packed stainless steel wire.New weld metal is deposited where the "slugged"welding wire meets the existing stainless overlay in a manner such that no HAZ is introduced into the base metal.The remainder of the excavation is filled flush or to a slight excess with stainless steel weld deposit.When using this strategy for repair of RTJ groove cracking, some companies simply fll the RTJ groove completely with new weld deposit to match the appropriate raised-face dimension.This new raised face surface is subsequently machined to accommodate a flat spiral-wound or groove metal gasket.

As with the previous two repair methods,all welding associated with this repair is typically made to the intact existing cladding or weld overlay,ata distance from the Cr-Mo steel base metal interface that will avoid creating a new base metal HAZ and thus requiring PWHT as discussed in 9.1.

Thistype of repair,when made on the reactor shell,can be inspected during future shutdowns from the outside surface using either TOFD or PAUT to check if cracking has reinitiated.When this repair is applied to ring joint grooves,specialtyultrasonic techniques described in Section 7 can be used to confirm that no crack propagation or new cracking from the base of the wire-filled excavation is occurring (see 7.5.4).

9.2.4 Method 4—Protecting Exposed Cr-Mo Steel with Metal Spray Coating

Figure 23 shows the details associated with covering the exposed base metal with a SS metal spray coating after grinding out a crack.Typically,the metal coating isapplied using a high-velocity oxy-fuel(HVOF)technique. Application of a metal spray coating to protect the exposed Cr-Mo steel is not a frequently used strategy to repair heavy wall Cr-Mo reactor vessels,at least partly due to the logistics of performing a proper metal spray application on short notice when the need arises unexpectedly during a turnaround.This repair strategy is highly dependent on the skill and experience of the personnel operating the spray equipment.It is important that the company contracted to applythe metal spray coating has experience and has demonstrated an ability to apply a reliable and long-lasting SS spray-applied coating.

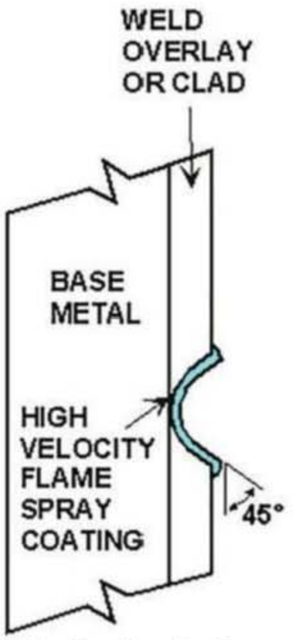


Figure 23—Protecting the Exposed Cr-Mo Steel at the Base of the Groove from Sulfidation with a Spray-applied SS Coating

Metal spray coatings have shown a tendency to fail in service,in some cases in less than five years.Even with proper application procedures and skilled operators,some companies limit the maximum service of metal spray coatings used for heavy wall reactor repairs to five years.After this time,the metal spray coating can be reapplied,or an alternative repair strategy can be implemented.

The repair area can be visually inspected during down times to assess the condition of the metal spray and check for crack reinitiation,with follow-up PT if needed.Crack initiation and growth can also be checked from the outside using TOFD or PAUT.

9.3 Examples ofWeld Repairs Performed without PWHT

As part of the JIP on Aging Reactors,several case histories were collected of repairs performed on reactor vessels,both with and without PWHT.These case histories appear in Attachment 1.Many of these repair case histories involve repairs performed using the general guidance included in Section 8 that did not require a PWHT after the repair was performed.There are also repair case histories in Attachment 1 that require a PWHT as discussed in Section 10.Additional case histories compiled by the Japanese Welding Engineering Society are detailed inAttachment 2.

If the excavated area is very extensive and the requirements in Parts 4 or 5 of API 579-1/ASME FFS-1 cannot be satisfied for the existing design conditions,including advanced Level 3 analysis with finite element analysis (FEA),then rerating the vessel can be considered.The design conditions would need to be reduced so thatthe excavation meets the requirements in API 579-1/ASME FFS-1,and hence,would avoid a repair that requires PWHT.Although this is an option,it is often not practical for refinery operations.

9.4 Considerations for Welding on Older Stainless Steel Weld Overlay

Some have reported difficulties welding on older versions of SS weld overlay in reactor vessels.Older versions of 300 series SS have been found to have relatively higher levels of sigma phase present,as discussed in 5.5.In addition to high levels of sigma phase,a SS weld overlay in a heavy wall reactor will have a high level of tensile residual stresses when cooled to ambient temperatures,as discussed in 5.9.These two effects generally have been cited in contributing to cracking problems when trying toweld on old SS weld overlay.Others have reported that when they have welding problems on the older versions of 300 series SS,they are able to successfully perform welds by minimizing heat input and interpass temperature during welding.Minimizing heat input and interpass temperature typically is achieved by using smaller diameter welding consumables and using a stringer bead welding sequence.

9.5 Considerations for Using Controlled Deposition Welding to Avoid PWHT

This document does not provide guidance or experience related to the use of controlled deposition welding (CDW)on heavy wall Cr-Mo reactor vessels to avoid the need for PWHT,for the reasons given below.CDW has been used effectively to repair 1/$_4$Cr-/$_2$Mo and 1Cr-$_2$Mo steel coke drums and catalytic reforming unit vessels,i.e.services with no or significantly lower hydrogen partial pressures compared with the heavy wall hydroprocessing vessels covered by this technical report.A non-PWHT'd CDW procedure is generally not considered an appropriate repair practice for heavy wall reactorvessels for the following reasons.

—Most heavy wall Cr-Mo reactor vessels in hydroprocessing service are constructed from 2/$_4$Cr-1Mo or Cr-Mo steels with higher alloy levels that make the steels more hardenable than 1/$_4$Cr-Y$_2$Mo or 1Cr-Y$_2$Mo steels.It has been demonstrated that a repair weld deposit using a CDW procedure on 2/4Cr-1Mo or Cr-Mo steels with higher alloy levels cannot meet maximum acceptable hardness limits specified for repair welds.

—CDW procedures for 1/$_4$Cr-$_2$Mo and 1Cr-/$_2$Mo steels have been developed that result in good mechanical properties in the weld deposit and HAZ;however,the resulting CDW weldments have not been tested for their resistance to hydrogen embrittlement.As aresult,a CDW procedure without PWHT developed for 1/$_4$Cr-V$_2$Mo or 1Cr-/$_2$Mo has not been employed to repair a heavy wall vessel in high-temperature high-pressure hydrogen service.

-ACDW repair without PWHT is not expected to provide relief of residual stresses from the repair welds.CDW techniques have been demonstrated to reduce the as-welded hardness of the repair area and HAZ.However, research including physical measurement of residual stresses via metal removal,neutron diffraction,and/or X-ray diffraction have demonstrated that the residual stress field is not significantly reduced by a CDW weld procedure,compared to a normal weld procedure without precise deposition control.2223

10 Weld Repairs Requiring PWHT

10.1 General Considerations for Performing Weld Repairs that Require PWHT

Due to the complexity and risks associated with a field PWHT,all possibilities for repairing without the need for PWHT are usually carefully considered before proceeding with a repair that does require PWHT.As described in Section 8,the resulting blended groove from removing a crack can often be shown to meet the requirements of Part 5 of API 579-1/ASME FFS-1,making it acceptable to leave without a weld buildup repair.Typically,the only other consideration for those repairs is the need to protect any exposed Cr-Mo steel at the base of the blended groove from sulfidation.If a weld repair is deemed to be required,Section 9 details various situations and methods where an effective repair can often be made without PWHT.

PWHT of a Cr-Mo weld repair is complicated and requires a high level of planning and expertise in execution, especially when done in the field compared with the initial PWHT,which is typically done in the fabricator's shop.

The owner typically makes sure that all the following important considerations associated with Cr-Mo steel weld repairs are reviewed and evaluated before the final repair plan is developed and executed to ensure the risks are adequately mitigated.

10.2 Contractor Performing Weld Repair

It is important to choose welding and heat treatment contractor(s)familiar with and preferably experienced with performing weld repairs on Cr-Mo heavy wall reactorvessels.Their expertise would ideally involve all aspects of the repair,including the following:

—experience with reviewing weld procedures for the repair of Cr-Mo heavy wall reactor vessels and detailed knowledge of the requirements in Section IXof theASME Code andAPI Recommended Practice582,Welding *Guidelines for the Chemical,Oil,and Gas Industries;*

—development and/or review of repair welding procedures by a Welding Engineer certified by the IW,the American Welding Society(AWS),or the equivalent;

—understanding ofthe effects that heat treatment performed as part of the repair could have on mechanical properties,including potential degradation of strength and toughness;

—familiarity with WRC Bulletin 452,Recommended Practices for Local Heating of Welds in Pressure Vessels, especially in the required heating bandwidths and thermocouple placements;

—familiarity with API documents for fabricating heavy wall Cr-Mo vessels,including API 934-A and API 934-B for vessels fabricated from 2/$_4$Cr-1Mo,3Cr-1Mo,and V-modified varieties of these steels,and API934-C and API934-D for vessels fabricated from 1/$_4$Cr-/$_2$Mo and 1Cr-$_2$Mo;

—familiarity with all state,local,and/or other jurisdictional requirements affecting the repair,including any additional requirements for ASME BPVC,Section VIII,Division 2 vessels,if applicable.

The welding contractor typically prepares a detailed written plan for the repair that includes the following,as a minimum:

—detailed repair procedure including repair location and sequence of steps.This includes preheating,welding, preheat maintenance,intermediate dehydrogenation heat treatment(DHT)or intermediate stress relief(ISR), PWHT,NDE,etc.;

—excavation methods such as grinding,air arc gouging,and machining,including the extent or limits of excavation;

—welding procedure specifications (WPS)and supporting procedure qualification records (PQR);

—PWHT procedure,including the requirements for insulating and supporting the reactor vessel while at the PWHT temperature;

—welders'and welding operators'qualifications (WPQ);

—quality control procedures,including NDE and test procedures as well as PMI and traceability procedures;

—inspection and test plan (ITP),including acceptance criteria and definitions of inspection hold points;

—material test certificates for new welding consumables and base metal,as applicable.

In situations where local welding contractors do not have experience with performing repairs on Cr-Mo heavy wall reactor vessels,the owner can contact the original vessel fabricator to provide assistance.Often,heavy wall vessel fabricators perform field repairs on vessels they originallyfabricated.

10.3 Heat Treating Guidelines Associated with Cr-Mo Weld Repairs

Heat treating is an essential part of a Cr-Mo weld repair,and PWHT is performed to ensure the Cr-Mo base metal has good mechanical properties (includingfracturetoughness)after the weld repair is performed.In addition to a final PWHT,the repairs may require an ISR and/or DHT during a weld repairif the repair area needs to be cooled to below the preheat temperature before PWHT(for reasons such as to perform NDE).The vanadium-modified Cr-Mo reactor steels are more sensitive to possible issues in the non-PWHT condition,but both standard and vanadium-modified Cr-Mo steels need to use the proper steps.Guidance for heat treating heavy wall vessels is found in API934-A and API 934-B for vessels fabricated from 2$_4$Cr-1Mo,3Cr-1Mo,and V-modified varieties of these steels,and API934-C and API 934-D for vessels fabricated from 1/$_4$Cr-V$_2$Mo and 1Cr-Y$_2$Mo.

The PWHT temperature and hold time employed for a repair weld typically complies with the guidance in either API 934-A or API 934-C,as appropriate based on the material of construction.

Local PWHT is generally employed for repair welding and is typically required to use a full circumferentialsoak band,heat band,and gradient control band in accordance with 6.4.3.4 of ASME BPVC,Section VII,Division 2.Local,spot PWHT(commonly called a bulls-eye heat treatment)is generally not allowed,as it can introduce high thermal stresses and result in excessive residual stresses,cracking,or distortion.Other considerations with regard to local PWHT are given in Article 2.14 ofASME PCC-2 and in WRC Bulletin 452.The widths of the soak band,heating band,and gradient control band are generally required to meet or exceed the guidelines given in WRC Bulletin 452,along with the required thermocouple placements.

The use of external electric heating elements is generally preferred when heat treating heavy wall vessels in the field to maintain the necessary metal temperature control.Attempted heat treatment of repair welds with internally applied heating elements can result in failure of the power supply lines for these elements,complicating the PWHT process.All ISR heat treatments and final PWHT requiring metal temperatures above 540℃(1000°F) typically follow the guidance in WRC Bulletin 452.It is prudent to have all heat-treating equipment and thermal insulation onsite and ready for application before a weld repair is executed in the field.

It is important to consider the tempering effects that can occur during all heat treatments conducted as part of a welded repair.Typically,the original fabrication and material testing of heavy wall reactor vessels specified simulation of at least one extra PWHT and in many cases two extra PWHTs to account for possible repairs or modifications to the vessel in the future.After simulating these heat treatments,the strength and toughness of the base materials andwelds were tested.It is important to check,ifavailable,the mill certs,fabrication records, and records of previous weld repairs,including PWHT time at temperature,for the vessel to determine the effects of additional heat treatment(s)during the planned weld repair.This may require an assessment using a time-temperature tempering parameter such as the Larson-Miller parameter.Hardness readings can also be used to estimate the strength levels of the base materials or welds in the repair area before doing the repairs.

The possibility of changes in mechanical properties due to PWHT are some of the first considerations when planning a weld repair that requires PWHT.This includes concerns associated with uneven heating during PWHT.If there is a riskthat the repair heat treatment(s)may cause the tensile strength of the vessel base metal or weld metal to drop below the minimum acceptable level,it may be prudent to consider other options,such as rerating the vessel to a lower design pressure and leaving a large blend ground groove as-is(or applying one of the strategies presented in Section 9).If there are questions regarding the mechanical properties of the base metal,it may be possible to use test blocks of the original material that occasionally are installed in new reactors to provide test specimens throughout the life of the reactor.Specimens from the test block can be used to determine the starting mechanical properties of the base metal before initiating the weld repair.Then,the block (or a portion of the block)can be exposed to heat treatments simulating those to be applied during the repair and tested to determine the expected mechanical properties of the base metal after all weld repair heat treatments are completed.

In addition to the heat treatments typically required to comply with the requirements in the relevantAPI fabrication documents listed earlier(which apply to newly fabricated vessels),a prerepair dehydrogenation heat treatment to remove soluble hydrogen that is charged intothe reactor shell during normal operation is also done.This step is also called a hydrogen bakeout and is performed prior to conducting a weld repair to avoid welding defects from dissolved hydrogen in the base metal.For example,a prerepair DHT of 300℃to 350℃(570°Fto 660°F) for four hours may be extended to a longer time for thicker walled reactors with severe operating conditions. This heat treatment is normally performed before removing defects in Cr-Mo steels,unless the defects are in or directly under the SS overlay.In this situation,it is beneficial for the SS overlay to be removed,because it impedes hydrogen outgassing during the prerepair DHT due to the low diffusivity and high solubility of hydrogen in SS.In situations when the weld overlay is not removed prior to a DHT,heating normally is performed from the weld overlay side of the reactor wall.

As with a DHT possibly applied during the repair,a prerepair DHT is not performed at high enough temperature to adversely affect mechanical properties nor require compliance with WRC Bulletin 452.

AfteraPWHTis performed on a repairweld,hardness testing as required inAPI934-AorAPI934-Cas appropriate is typically conducted to ensure that the hardness levels of the repair meet the same requirements as fora newly fabricated vessel.

11 Consumables,Testing,NDE,and Recordkeeping of Weld Repairs

11.1 General

The guidance in this section is applicable to repairs described in both Sections 9 and 10.

11.2 Specific Weldability and Weld Consumable Considerations

All welding consumables for repairs are typically required to meet the requirements in API 934-A for 2$_4$Cr-1Mo,3Cr-1Mo,and vanadium-modified grades of these steels,and API 934-C for 1/$_4$Cr-Y2Mo and 1Cr-$_2$Mo,as applicable.Most welding contractors will work closely with a fabricator of heavy wall Cr-Mo vessels to ensure that the proper welding consumables are selected.

When poor weldability of a SS weld overlay due to aging is expected,a bead-on-plate test on the SS weld overlay can be performed in the following manner:

1)lay down a weld bead at least 50 mm(2 in.)in length on the surface of the overlay in the area to be repair welded or an adjacent area;

2)inspect the bead by PT and/or UT for crack-like indications.

NOTE In cases where only PT is used for the inspection,PT is typically performed on the weld bead and surrounding area and after the weld bead is ground off to check for any cracking that may have occurred under the bead.

If the bead-on-plate test demonstrates that the material has poor weldability,such that subsequent welding is expected to create additional flaws or propagate existing flaws,the deposition of a butter layer,using a higher ductility SS filler metal,such as E/ER 309L,or the use of the metal thermal spray coating option described in 9.2.4 canbe considered.

Other specific welding guidelines found in API Recommended Practice 582 are typically required as good practice,including surface preparation,interpass temperature limits,listed welding consumable combinations for stainless-to-low-alloy steel welds,etc.

11.3 Testing,NDE,and Inspection Hold Points Associated with Repair Welds

All tests associated with a weld repair are typically performed in accordance with the requirements ofAPI 934-A for 2/$_4$Cr-1Mo,3Cr-1Mo,and all V-modified versions of these steels,and in accordance with the requirements of API 934-C for 1/$_4$Cr-/$_2$Mo and 1Cr-/2Mo.In addition,the guidance given inASME PCC-2,Part 5,Article 2.5 are generally followed when applicable.

11.3.1 Confirmation of Defect Removal

For all repair strategies discussed in Sections 9 and 10,the defect is first removed before a weld repair is executed.In these cases,the surfaces including the excavated defect and the immediate surrounding area are typically examined by MT or PT.The examination usually includes the groove or other new surface created where the defect was removed,and an area at least 50 mm(2 in.)surrounding the removed defect.Depending on the type of flaw and its location,it may be necessary to also perform UT(including the advanced techniques described in Section 7)in the same area to detect any embedded flaws beneath the ground area.Any associated embedded flaws could propagate during repair welding.This UT is often performed fromthe opposite surface of the material from the repair.

11.3.2 Testing and NDE of Cr-Mo Base Metal and Weld Deposits

For repairs that include the deposition of new Cr-Mo fller metal,inspection is typically performed after repair welding,but before applicationof the SSoverlayand PWHT.The repaired area is usually examined inaccordance with 8.4.1 of API 934-A.PAUT in lieu of RT,meeting the requirements ofASME BPVC,Section VIII,Division 2, 7.5.5,is most commonly used.MT is typically also done.After PWHT,if PWHT is being done(which is after the application of the SS overlay),it is typically required that the PAUT be repeated.

In addition,forrepairs involving the welding of new austenitic overlay directly to the underlying Cr-Mo (requiring PWHT),all repaired areas are typically examined by straight-beam UTin accordance with ASME BPVC,Section V,SA-578,supplementary requirement S7,and by PT.Both inspections are generally performed before final PWHT and again after final PWHT.

The chemical composition of the completed weld deposit representing each different welding procedure is typically checked prior to proceeding with the next procedure or performing PWHT.

For reactors with no SS overlay or cladding,after final PWHT,hardness determinations are generally made for each pressure-retaining weld in accordance with the requirements ofAPI934-A for 2V$_4$Cr-1Mo,3Cr-1Mo,and all V-modified versions of these steels,and in accordance with the requirements of API 934-C for 1/$_4$Cr-Y$_2$Mo and 1Cr-$_2$Mo.

If the repair involves replacement of a portion of the cylindrical shell or head with new material,impact testing of production test plates would also be required per the ASME Code.

In the unlikely scenario that a hydrostatic pressure test is performed,all accessible surfaces associated with the repair are typically required to be examined by PT again after the pressure test.If a bare(not clad or weld overlayed)reactor is weld repaired and pressure tested,MT instead of PT would be used.Additionally,any external weld surfaces associated with the repair,such asthe completed weld in a flush patch repair,are typically inspected with MT rather than PT.An alternating current (AC)yoke method is generally used to prevent arc strikes.

11.3.3 Testing of Weld Overlay

The chemical composition and ferrite content of the weld overlay are typically required to be verified to meet the requirements in API 934-A or API 934-C.Frequency of the analysis (i.e.number of tests per area of weld)can be determined by the owner considering the size of the repair area,but at least one analysis for each welding process and each lot of welding consumable is typically performed.

Forrepairs that are non-PWHT and only include welding onto the existing SS overlay,the completed surface of the overlay is typically inspected with PT.

11.3.4 NDE Personnel,Methods,and Acceptance Criteria

NDE personnel are typically required to be qualified in accordance with 8.1 of API 934-A.As discussed in 7.3, a reactor repair inspection is typically considered critical enough to justify the use ofASNT Level ll or Level III qualified NDE technicians.Also,some owners go beyond industry certificationandrequire the use of performance qualified NDE personnel who have passed company-specific qualfication tests,which helps insure proficiency in the use of the NDE technique and interpretation of its results.

Unless otherwise specified,the methods and acceptance criteria for NDE required as part of a weld repair at least can be the same as those identified in API934-A for new fabrication welds.

If new base metal is to be installed,it is typically examined in accordance with 8.2 of API 934-A.

11.3.5 Hydrostatic Testing After a Weld Repair

A hydrostatic test or pressure test in general is seldom performed following weld repairs on a heavy wall reactor vessel,even when the weld repair is followed by a PWHT.The possible effects of temper embrittlement and hydrogen embritlement generally make a pressure test impractical and unadvisable.When a pressure test cannot be performed,NDE in lieu of a pressure test in accordance with Article 5.2 of ASME PCC-2 may be employed with owner and jurisdictional approval.The use of NDE in lieu of a pressure test is the most frequently followed practice in cases when a weld repair is performed.

In the unlikely scenario that a hydrotest is required,precautions are generally taken to avoid additional damage to the reactor vessel and repair area.The hydrostatic test pressure may or may notbe the same as that required for the reactor when originally fabricated.Detailed requirements for hydrostatictesting are given inASME PCC-2, Article 5.1,6.1.

In all reactors with SS overlay or cladding,water used for a pressure test is generally required to have low chloride levels to ensure the 300 series SS overlay and internals are not affected by chloride SCC.Some owners limit chloride levels in hydrotest water to 50 wppm.It is essential that all of the water is drained from any equipment that is filled with water during the pressure test.Remnant water from a pressure test has been known to cause chloride SCC in the reactor vessel itself and in downstream SS piping when the unit is restarted.For repairs that include SS liners with vent holes/gaps,it may not be possible to completely drain the void behind the liner following a hydrotest,which is another reason that hydrotesting may not be desirable.Draining of the catalyst dump nozzle,side entry thermowell nozzles,and othersimilar deadlegs or low points is also considered critical.

Precautions are typically taken to prevent freezing of the pressure test water when testing in low ambient temperature conditions.Even with newer reactors,it is important that the temperature of the pressure test water meets the reactor's MPT to account for possible temper embrittlement and hydrogen embrittlement.

11.4 Recordkeeping and Future Inspection of Weld Repairs

Information and data from all weld repairs are typically required to be maintained as part of the inspection and/ or maintenance records associated with the reactor vessel.The Japan Welding Engineering Society (JWES)has developed a formthat can be used todocument weld repairs on heavywall pressure vessels.A blank version of this form and several examples of repairs that have been performed and documented using this form are shown in Attachment 2.This form provides complete documentation of a repair performed on a heavy wall reactor vessel.

As deemed necessary,a plan is often developed forfollow-up inspection of each weld repair.Depending on the damage mechanisms involved,future inspections for cracking or corrosion of the repair area may be justified over several subsequent turnarounds.This is especially appropriate when the repair location is in a high-stress area where cracking may be more likely,such as at an internal support ring.

11.5 Examples of Weld Repairs with and without PWHT

Attachment 1 contains multiple examples of weld repairs performed with and without a PWHT.This compilation was developed by the JIP on Aging Hydroprocessing Reactors.Attachment 2 contains examples of repairs that were compiled by the JWES.

12 Attachment 1—Documented Repair Case Histories from the JIP on Aging Reactor Vessels

The JIP members submitted information on repairs performed within their respective companies.Some of the information received was very limited as to the details of the repair procedure.The minimum ultimate tensile strength (UTS)is indicated only if it was different from the normal 70 ksi(483 MPa)to 75ksi(520 MPa).In all cases,the material of construction is conventional $2/_4$Cr-1Mo unless otherwise noted.The available information relative to the various repairs is summarized as the following case histories(CH#).

CH#1-RTJ groove cracks were repaired on two reactor nozzles (WNFtype)by weld buildup and conversion to a nubbin/groove joint.The procedure involved a Cr-Mo buildup and SS overlay restoration.The reactor was built in 1974 and repaired in 1988.The nozzle was outgassed prerepair for 20 hours at 425℃(800°F)and postrepair at 350℃(660°F)for one hour.The repair was PWHT at 700℃(1292°F)for two hours(P=35,600).Above 300℃(572°F),the maximum heating and cooling rates were 56°℃hr(100 °F/hr).The repair required 20 days.

CH#2-RTJ groove cracks were repaired on the 18 in.(460 mm)outletnozzle by replacing the weld neckflange (WNF)fitting.A single modified J-groove bevel was used.The reactor was built in 1970 and repaired in 1991. The pre-and postrepair outgassing were both at 385℃(725°F)for two hours.The repair was PWHT at 695℃ (1285°F)for eight hours(P=36,500).The heating rate was 66℃hr(118°F/hr)above 425℃(800°F).The cooling rate was 82°℃hr(148°F/hr)to 425℃(800°F)and then still air cooled to 93℃(200°F).The nozzle was locally hydrotested at 22.6 MPa(3281 psi).The repair required 22 days.

CH#3-RTJ groove cracks were repaired on the reactor outlet nozzle pipe flanges(WNF type)and other pipe flanges.Some cracked flanges had previously been repaired.The reactors were built in 1967 and repaired in 1987.The nozzles were prerepair outgassed at 315℃(600°F)for 10 hours and no postrepair outgassing was indicated.The 1987 repairs were PWHT at 675℃(1250°F)for two hours(P=34,800).The heating and cooling rates were 222C/hr(400°F/hr).

CH#4-Overlay cracking repair procedure using 2.4 mm(23/32 in.)fller wires twisted to make a braid.The braid was placed in the bottom of the ground groove to protect the Cr-Mo base metal.The groove with the braid was welded to protect the base metal from the corrosive environment.No PWHT was required.

CH#5-Overlay cracking repair in which the overlay had cracked whenever it was welded.A channel cover from the feed/efluent exchanger was high-velocity flame coated with TP 316 SS in grooves where the cracks were removed.The cover was coated in 1987 and inspected in 1994.No coating deterioration or corrosion was noted. No PWHT was required.

CH#6-Crack in reactor wall started on ID and propagated 60%through the wall thickness.The reactor was built in 1968 and repaired in 1980.The reactor was outgassed on shutdown.The repair was PWHT without cooling from preheat temperature.The repair was PWHT at 620℃(1150°F)for 4 hours(P=33,700)so the 759 MPa(110 ksi)UTS would not be reduced.A through-wall crack occurred after 20 months of service,requiring extensive repairs (see CH#7).Several(9)cracks initiated where lugs were welded to the insidewall to hold heating elements in place.These cracks propagated from the inside surface to approximately midwall.

CH#7-Severe cracking of a previous repair required the installation of a flush window patch in a 5 in.reactor wall.The reactor was built in 1968 and this repair was made in 1982.The reactor was outgassed at 315℃ (600°F)for six hours during shutdown.The repair was given an intermediate stress relief at 607℃(1125°F)for three hours before cooling below the preheat temperature.The PWHT was at 662℃(1215°F)for 19.5 hours (P =35,700).The heating rate was 28°℃/hr(50°F/hr)and the cooling rate was 55℃/hr(100°F/hr).

CH#8-A midwall crack in a circumferential weld on a 127 mm (5in.)thick reactor was repaired.The reactor was built in 1968 and repaired in 1992.The reactor was prerepair outgassed at 315℃(600 °F)for six hours during shutdown.The postrepair outgassing was at 425℃(800°F)for four hours.The repair was PWHT at 653℃(1210°F)for 16 hours(P=35,700).The heating rates and cooling rates were 28℃hr(50 °F/hr)below 204℃(400 °F).The heating and cooling rates were 42°℃hr(75°F/hr)above 204℃(400°F).The specified maximum thermal gradient was 275°℃/m(150°F/ft).The repair required 19 days and the heating contract cost was approximately $135,000.

CH#9-RTJ groove cracking required repairs to the reactor's inlet and outlet nozzles.The repair involved a Cr-Mo weld buildup and SS overlaying.The nozzles were 230 mm(9 in.)thick and the head was 114 mm(4.5in.)thick. The reactor was built in 1968.The outlet nozzlewas repaired in 1984 and the inlet nozzle was repaired in 1986. The reactor was outgassed at 315℃(600 °F)for six hours during shutdown.The postrepair outgassingwas at 315℃(600°F)for four hours.The PWHT was at 653℃(1210°F)for seven hours(P=34,800).The heating and cooling rates were 28℃hr (50°F/hr)below 150℃(300°F)and 42℃/hr(75°F/hr)above 150℃(300°F). The maximum thermal gradient was 182°℃/m(100°F/ft).The head minimum UTS was 621 MPa(90 ksi)and no derate was required.The RTJ grooves were last inspected in 1992 and no defects were detected.

CH#10-RTJ groove cracking on the outlet nozzles of two reactors was repaired.The nozzle and head thickness were 190 mm.(7.5 in.)and 129 mm(5.062 in.),respectively.The reactors were built in 1966 and repaired in 1987.The reactors were outgassed during the shutdown.The postrepair outgassing was at 315℃(600°F)for four hours.The PWHT was at 667℃(1235°F)for nine hours(P=35,600).The heating and cooling rates were 28°℃hr(50°F/hr)below 150℃(300°F)and 42℃hr(75°F/hr)above 315℃(300°F).The maximum thermal gradient was 182C/m(100°F/ft).The repair required~11 days.

CH#11-A high-velocity flame coating of Tp 316 SS was applied where a section of cladding was removed on a $1/_4$Cr-$_2$Mo reformer pretreater reactor.The repair was completed in 1996.No PWHT was required.The repair required 1 day at a cost of $33,000.

CH#12-Cracking at an internal support overlay had propagated into the base metal of a $1/_4$Cr-/2Mo reactor and was repaired in 1988.The repair involved a Cr-Mo weld buildup and an overlay restoration.The repair was PWHT'd at 650℃(1200°F).

CH#13-Overlay cracking was found in 1982 and 1988(in different locations).The repair wasmade using 4.8mm (3/16 in.)filer wire flattened to 3.2 mm(1/8 in.)in the bottom of the groove where the cracks were ground out. The filler protected the base metal when the groove was welded flush with the overlay.No PWHT was required.
CH#14-Leaks in loose nozzleliners required repairs.The repair involved the replacement of the defective nozzle liners.The new Tp 347 liners were designed and installed so all welding was to existing overlay.No PWHT was required.

CH#15-RTJ groove cracking was repaired and converted to a RF-type joint design on a WNF nozzle.The procedure involved a Cr-Mo buildup and overlay to eliminate the RTJ groove.The repair was PWHT'd but no details were provided except that the thermal gradient was limited to 275℃/m(150 °F/t)up to 315℃(600°F).
CH#16-RTJ groove was converted to a RF design when a minimum of3.2mm(1/8 in.)overlay existed toprotect the base metal.The groove was built up with E-347 with heat input controlled to 18,000 Joules/in.No PWHT was required.

CH#17-Repaired~200 overlay cracksand a bottomnozzle RTJ groove crack.The reactor was builtin 1973 and repaired in 1993.The shell and head thickness were 115mm(4.538 in.)and 79 mm(3.112 in.),respectively.The prerepair outgassing was600°F for four hours.The preheat was only 93℃(200°F),probably due to personnel working inside the reactor.The repairswere PWHT'd at 690℃(1275°F)for five hours(P=35,950).The heating and cooling rates above 315℃(600 °F)were 56°℃/hr(100 °F/hr)and 28℃hr(50°F/hr),respectively.Two small racks were found in the overlay after the PWHT,which were repairedwithoutanother heattreatment.The repair required less than 21 days and cost $496,000.

CH#18-Overlay cracking was repaired using lap patches.The reactor was built in 1974 and repaired in 1993. No PWHT was required.

CH#19-Overlay cracks(approx.70)in the bottom head near the catalyst support cone hold-down pad and ring were repaired.A360°crack in the bottom nozzle RTJ groove was determined acceptable for service after the crack was ground out.The cracks in the overlay were ground out and lap patched.No PWHT was required.

CH #20-Cracking at the Cr-Mo buildup pad for the catalyst support cone was repaired on three reactors.The reactors were buit in 1969 and repaired in 1973.Landing pads were removed,overlayed,and PWHT'd at 690℃ (1275°F)for 11 hours(P=36,500).Cracks were found in two reactors in 1993 and repaired as CH#21.

CH#21-In 1993,one of the repaired reactors had cracks in the overlay that ran into the base metal.This required a Cr-Mo buildup,overlay,and PWHT.An intermediate heat treatment at 650℃(1200°F)for two hours was applied following the repair before cooling.The repairs were PWHT'd at 690℃(1275°F)for four hours (P =35,800).Some cracks were detected in the overlay after the PWHT,which were ground out and repaired with shingle patches.Recent inspections showed no metal loss under the shingle patch.Some overlay cracks were present but did not go to base metal.

CH#22-Overlay cracking in two 7-in.-thick reactors at weld pads and tray attachment welds required repairs. The reactors were built in 1969 and repaired in 1973 and 1993.One reactor had the cracks removed,weld repaired,and PWHT'd.The 1993 repair was given a 650℃(1200°F)intermediate stress relief for two hours. The repairs were PWHT'd at 690℃(1275°F)for four hours(P=35,800).The cracks in the other reactor were not removed after an FFS analysis.The cracks had not grown in depth but some had become longer(at the ID surface)based on a subsequent inspection.

CH#23-Through-wall cracks in nozzle welds and cracks in top head seam welds required repair.The two reactors were built in 1965 and required repairs in 1966.The reactor shell was 145 mm (5.69 in.)thick and the heads were 95 mm(3.75 in.)thick.All girth and nozzle welds were heat treated to lower the UTS from 828 MPa (120 ksi)to 704 MPa(102 ksi).Subsequent inspections showed no problems.The reactors were retired in 1989.

CH#24-Through-wall cracks in nozzle welds and cracks in top head seam welds required repair.The three reactors were built in 1965 and required repairs in 1966.The reactor shell was 138 mm(5.445 in.)thick.Al girth and nozzle welds were heat treated to lower the UTS from 828 MPa(120 ksi)to 704 MPa(102 ksi).Subsequent inspections through 1990 showed no additional problems.

CH#25-Through-wall cracks in nozzle welds and cracks in top head seam welds required repair.The three reactors were built in 1965 and required repairs in 1966.All girth and nozzle welds were heat treated to lower the UTS from 828 MPa(120 ksi)to 704 MPa(102 ksi).In 1969,cracks were found in the tray support that had propagated through the wall in one reactor.Two reactors required weld repairs(CH#26).

CH#26-Three reactors repaired in CH#25 experienced tray support cracking,onewhich propagated through a long seam causing a leak.Two reactors required weld repairs and PWHT.The reactors received an intermediate stress relief at 607℃(1125°F)for three hours.The final PWHT was at 653℃(1225°F)for 12 hours,which lowered the UTS to 650 MPa(94 ksi).Periodic inspections from 1971 through 1989 revealed no additional problems.

CH#27-Cracking in the overlay was repaired by seal welding an 11 gage sheet that was contoured to fit the groove left by crack removal.The reactor was built in 1966 and repaired in 1990.Initial seal welds were made using GTAW process and some HAZ cracking was experienced.New patches were stitch welded using SMAW process and cooled to ambient between stitches;no cracking was experienced.The repairs were inspected in 1992 and some cracks were found under the patch.The cracks were removed and a new patch was installed. A1995 inspection revealed additional overlay cracking in the area of a previous repair.Two cracks could not be accessed for repair and will be monitored.

CH#28-Overlay cracking was found in 1995 at the catalyst cone support in one reactor but not in the sister reactor.The reactors were built in 1966.The head thickness was 138 mm (5.44 in.).The cracks did not go into the base metal.An FFS analysis indicated that no immediate repair was required.The defects will be monitored.

CH#29-Cracks in a quench nozzle weld were found during the first UT inspection and repairs were required. The reactor was built in 1980.The nozzle in the 108 mm(4.25 in.)thick shell was repaired in 1993.The reactor was prerepair outgassed at 300℃ to 400℃(572°F to 750°F)for two hours.A postrepair intermediate stress relief was at 610℃(1130°F)for five hours.Afinal PWHT was at 690℃(1275°F)for eight hours(P=36,300). The PWHT used gas heat on the ID and resistance heaters on the OD.The repair took eight days.

CH#30-Overlay cracking in 102 mm(4 in.)quench nozzle was repaired in 1992 and 1995.The reactor was built in 1983.The 1992 repairinvolved seal welding a TP316 SS liner over the cracks.The 1995 inspection revealed that the overlay cracks had grown in length and went to the base metal.The overlay and cracks were removed and an Inconel 625 liner was seal welded to the overlay.The nozzle was drilled to vent the area behind the liner to atmosphere.No inspections have been made of the 1995 repair to date.

Ch#31-Overlay cracking at the toe of the supportbeam weld required repairs.The reactor was built in 1969 and the first repairs were required in 1986.The crack was removed,the cavity flled with refractory,and a 10 gage vented Tp 304 cover was welded to the overlay.A 1988 inspection revealed additional cracks at the end of the cavity that were removed and repaired as in 1986.A1994 inspection revealed no problems.

Ch#32-RTJ groove cracks propagated into base metal were repaired on the 438 mm(17.25 in.)outlet nozzle by replacing the RTJ WNF.The cause of the crack was presumed to be sigma phase embrittlement.Tp 347 final layer in the gasket groove of the replaced flange did not receive PWHT.A single modified U-groove bevel was used.The reactor was built in 1970 and repaired in 1991.The pre-and postrepair outgassing were both at 350℃ to 400℃(662°F to 752°F)for two hours.Welding using the SMAW process was applied.The repair was PWHT'd at 690℃(1275°F)for eight hours(P=36,300).The heating rate was 65°℃/hr(118°F/hr)above 427℃(800°F).The cooling rate was 48℃hr(86°F/hr)to 427℃(800°F)and then still air cooled to 93℃ (200°F).The nozzle was locally hydrotested at 22.6 MPa(3281 psi).

Ch#33-RTJ groove cracks were repaired on both the 600 mm (23.625in.)inlet nozzle and mating flange.The cause of the crack was presumed to be sigma phase embritlement.The gasket groove was converted to an RF-type joint on a WNF nozzle.The reactor was built in 1970 and repaired in 2001.The procedure involved aCr-Mo buildup and overlay to eliminate the RTJ groove.The pre-and postrepair outgassing were both at 300℃ to 400℃(572°Fto 752°F)for six hours.Welding using GTAW and SMAW processes was applied.The repair was PWHT at 690℃(1275°F)foreight hours(P=36,300).The heating rate was 55°℃hr(100°F/hr)above 427℃ (800°F).The cooling rate was 55°℃/hr(100°F/hr)to 427℃(800°F)and then still air cooled to 93℃(200 °F).

Ch#34-RTJ groove cracks were repaired.Hydrogen embrittlement and chloride SCC were presumed to be the cause of the cracks.The reactor was built in 1994 and repaired in 2001 as the result of FFS analysis.The procedure involved the weld overlay of Tp 347 and required no PWHT on the RTJ groove because the remaining weld overlay thickness of more than 2.5 mm (0.1 in.)was confirmed by UT inspection.

Ch#35-The reactor was built in 1993 but never put in service.Modification of the internal support ring was necessary because of an internal process change.Repair using TP 347 weld overlay was needed because the Cr-Mo base metal was exposed with no protection from hot H_2S corrosion.The repair procedure involved overlay buildup over the filler rod on a small area(12 mmx8mm)using GTAW process.TOFD UT inspection was done before and after repair.The problem in the repaired part is that a void is leftunder the weld overlay.

13 Attachment 2—Repair Documentation Table and Several Examples as Prepared by the Japan Welding Engineering Society (JWES)

Repair Record Blank Format

Record X-XXXXXX

<table>
<tr><td>Title</td><td colspan="6">XXXXX</td></tr>
<tr><td rowspan="2" colspan="2">Type of Reactor</td><td rowspan="2"></td><td>DT</td><td>℃</td><td rowspan="2">Owner</td><td rowspan="2"></td></tr>
<tr><td>DP</td><td>kg/cm²G</td></tr>
<tr><td rowspan="2" colspan="2">Operation History</td><td rowspan="2"></td><td>Base Metal</td><td colspan="3">(Thk:mm)</td></tr>
<tr><td>WOL/Clad</td><td colspan="3">(Thk:mm)</td></tr>
<tr><td colspan="2">Type of Flaws and Locations</td><td colspan="5"></td></tr>
<tr><td colspan="2">Note</td><td colspan="5">—</td></tr>
<tr><td rowspan="2" colspan="4">Repair Welding Procedure</td><td colspan="2">Data</td><td></td></tr>
<tr><td colspan="2">Contractor</td><td></td></tr>
<tr><td>Step</td><td>Work</td><td colspan="5">Details</td></tr>
<tr><td>1</td><td></td><td colspan="5"></td></tr>
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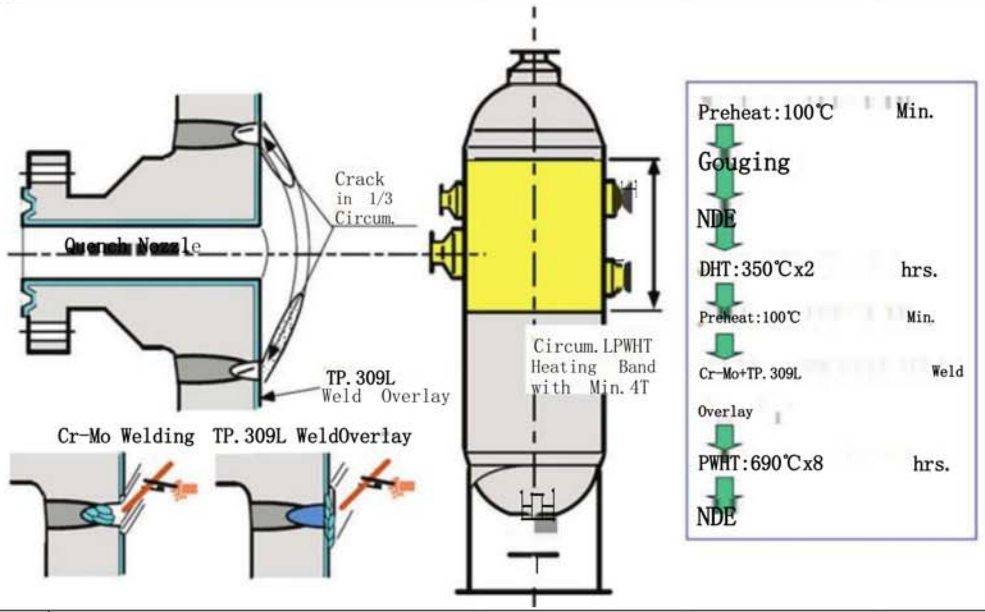
<tr><td>13</td><td>I</td><td colspan="5"></td></tr>
<tr><td colspan="7">Sketch of Repair</td></tr>
<tr><td colspan="7"></td></tr>
<tr><td>Reference:</td><td colspan="6">—</td></tr>
</table>

Record 1-Repair of Nozzle Attachment Welds

Title	Repair of Nozzle Attachment Welds				
Type of Reactor	Unicracking reactor	DT	427 ℃	**Owner**	—
		DP	116 kg/cm² G		
Operation History	Operated for 10 years	**Base Metal**	21/4Cr-1Mo (Shell thk:18 mm)		
		WOL/Clad	Type 309(thk:6mm)		
Type of Flaws and Locations	Deep and long welding defects in 4 in.-(I.D.)-quench nozzle attachment welds				
Note	—				
Repair Welding Procedure		**Data**	—		
		Contractor	JSW		

Step	**Work**	**Details**
1	Detection of flaws	NA
2	Removal of flaws	Arc air gouging and grinding with preheat at minimum 150℃(302F)
3	Confirmation of removal	NA
4	DHT	350℃(662° F)for two hours
5	Welding	Welding in the gouged area of Cr-Mo base metal under preheating at minimum 200℃ (392° F)using SMAW process with E9016-B3
6	Inspection	Dry powder MT after completion of Cr-Mo welding, maintaining preheating temperature
7	DHT	350℃(662° F)for two hours
8	Inspection	UT, wet MT and PT on the welds at ambient temperature
9	Weld overlay	Weld overlay maintaining preheat at minimum 100℃(212° F)using SMAW process with E309L
10	Inspection	PT on the overlay welds at ambient temperature
11	PWHT	PWHT at 690℃(1274° F)for eight hours - Soaking band was applied to full circumference of the reactor.
12	Finishing	Smooth up by grinder not to leave any sharp edges with R less than 50 mm
13	Inspection	MT, UT, and PT on the whole area welded and/or PWHT'ed

Sketch of Repair



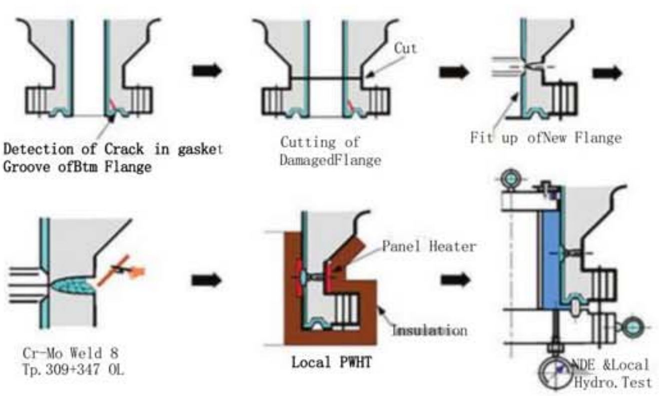
Reference:	8-1

Record 2—Full Replacement of Bottom Nozzle

Title	Full Replacement of Bottom Nozzle				
Type of Reactor	Hydrocracking reactor	DT	425℃	Owner	—
		DP	143 kg/cm² G		
Operation History	Operated for 5 years	Base Metal	21/4Cr-1Mo (Nozzle neck thk:94 mm)		
		WOL/Clad	Type 309/347(thk:6.4 mm)		
Type of Flaws and Locations	Cracking in gasket groove of 24 in.-(L.D.)bottom MH nozzle due to overtightening				
Note	—				
Repair Welding Procedure		Data	—		
		Contractor	JSW		

Step	Work	Details
1	Detection of Flaws	MT and UT from outside and PT from inside of the old nozzle neck
2	DHT	350℃(662° F)for two hours
3	Cutting	Nozzle was cut off at the old nozzle neck using arc air gouging under preheating at minimum 150℃ (302° F).
4	Groove Preparation	Arc air gouging and grinding and apply MT
5	Inspection	MT on gouged area
6	Welding	Fit-up of new flange Preheating at minimum 150℃ (302° F)from inside of the nozzle First side (outside)welding of Cr-Mo base metal using SMAW process with E9016-B3 Back gouging,grinding,and dryMT maintaining preheating temperature Second side (inside)welding of Cr-Mo base metal on the back gouged area using SMAW process with E9016-B3
7	DHT	350℃ (662F)for 12 hours
8	Inspection	Wet MT both on inside and outside surtaces at ambient temperature
9	Weld Overlay	First layer under preheating at minimum 100℃(212° F)using SMAW process with E309L. PT on the overlay welds at the ambient temperature. Second layer without preheating using SMAW process with E347L.
10	Surface preparation	Smooth up by grinder
11	Inspection for WOL	PT
12	Inspection for Weld Joint	RT,UT and MT
13	PWHT	PWHT at 690℃(1274° F)for eight hours Electric heater and insulation blanket were applied to both the inside and outside of the nozzle.
14	Finishing	Smooth up by grinder
15	Inspection	MT and UT from outside and PT from inside of the new nozzle neck
16	Pressure Test	Partial hydrostatic test for installed nozzle and the welds
17	Inspection	RT,UT,and MT

Sketch of Repair



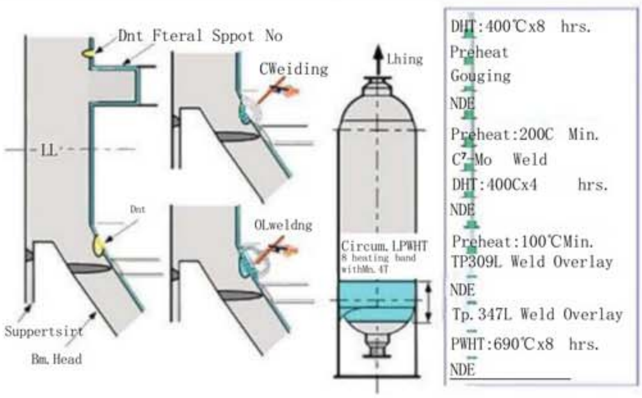
Reference:	8-1

Record 3—Repair of Dents inside Bottom Head

Title	Repair of Dents Inside Bottom Head				
Type of Reactor	Black oil conversion reactor	**DT**	454℃	**Owner**	—
		DP	216 kg/cm²G		
Operation History	Operated for 10 years	**Base Metal**	21/4Cr-1Mo (Shell/head thk:279/139 mm)		
		WOL/Clad	Type 309/347(thk:8 mm)		
Type of Flaws and Locations	Dents inside bottom head near support skirt due to erosion/corrosion				
Note	—				
Repair Welding Procedure		**Data**	—		
		Contractor	JSW		

Step	**Work**	**Details**
1	Detection of flaws	NA
2	DHT	400℃(752F)for 12 hours
3	Inspection	UT from outside of reactor
4	Groove preparation	Arc air gouging and grinding maintaining preheatat minimum 150℃ (302F) Grinding off the weld overlay adjacent to the gouged area
5	Inspection	MT on the ground surface maintaining preheat temperature
6	Welding	Welding in the groove area ofCr-Mo base metal maintaining preheat at minimum 200 ℃ (392F)using SMAW process with E9016-B3
7	Inspection	– Grinding and dry powder MT after completion of Cr-Mo welding on the ground surface maintaining preheat at minimum 200℃ (392F)
8	DHT	400℃(752° F)for four hours
9	Weld overlay	First layer overlay welding on the portion of repair weld and peal back,maintaining preheat at a minimum of 100℃ (212F)using SMAW process with E309L electrodes PT on the first layer weld overlay aftergrinding at ambient temperature Second layer overlay welding using SMAW process with E347L electrodes without preheating Grind smoothly
10	Inspection	– UT and PT the overlay welds and Cr-Mo base metal
11	PWHT	Specified the heating band (width and temperature profile)and the insulation blanket width during PWHT operation by FEM thermal stressanalysis – Lifted by crane during PWHT operation to prevent buckling ofsupport skirt (reactor weight:250 ton) PWHT at 690℃(1274F)for eight hours with minimum width of 4Twith a full circumferential soak band(T:reactor wall thickness)
12	Inspection	MT,UT,and PT on the whole area welded and/or PWHT'ed

Sketch of Repair



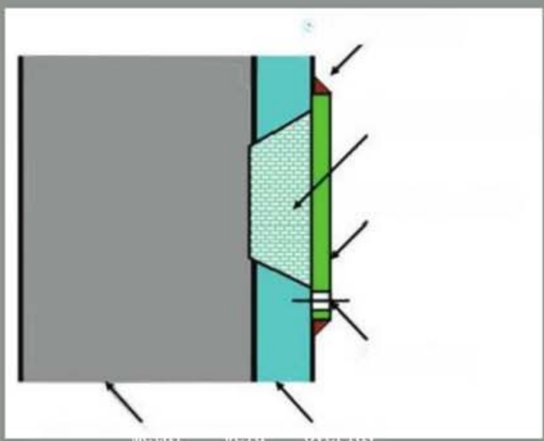
Reference: –

Record 4—Repair of Clad Plates and Weld Overlay

Title	Repair of Clad Plates and Weld Overlay				
Type of Reactor	HDC reactor	DT	469 ℃	Owner	—
		DP	166 kg/cm²G		
Operation History	Operated for 35 years	Base Metal	21/4Cr-1Mo (Shell thk:262 mm)		
		WOL/Clad	Type 347(Thk:3.2 mm)		
Type of Flaws andLocations	Weld overlay cracking on shell internal surface				
Note	—				
Repair Welding Procedure		Data	—		
		Contractor	JSW		

Step	Work	Details
1	Inspection	Road-on-motal toet UT from inside and outside to ensure the crack location and no propagation into base metal
2	Removal of flaws	Excavation of the crack to cover all the cracks in weld overlay Grinding followed by PT Installed refractory(AA22S)in ground-out groove to prevent corrosion - Refractory dry-out.
3	Welding	Fillet welding of patch plate (3-6 mm thk.,TP.347 or TP.321)covering the refractory fill.Patch plate,which contained a hole to vent hydrogen,was fillet welded to the weld overlay using GTAW process with ER347
4	Inspection	PT for fillet weld

Sketch of Repair



Bead-on Test

UT

Excavation of Cracks

PT

Fill of Refractory

SS Fillet Weld of Patch Plate

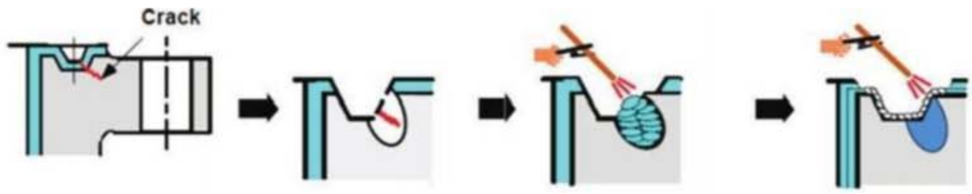
PT of Fillet Weld

Reference: 8-1

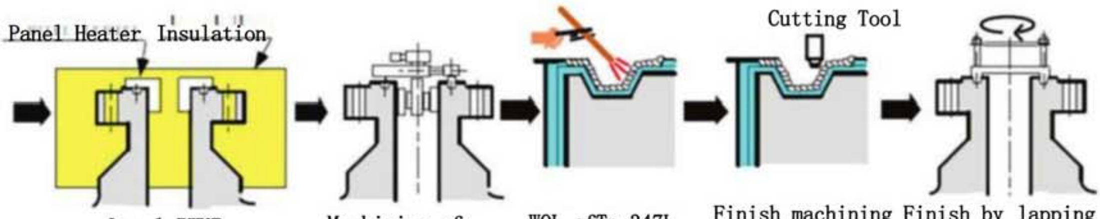
Title	Repair of Gasket Groove				
Type of Reactor	Hydrocracking reactor	DT	432℃	Owner	--
		DP	146 kg/cm² G		
Operation History	Operated for 22 years	Base Metal	21/4Cr-1Mo(Thk:---mm)		
		WOL/Clad	Type 309/347(Thk:5.3 mm)		
Type of Flaws and Locations	Crack by hydrogen&sigma phase embrittlement in RTJ gasket groove of top 18-in.-(I.D.) manhole nozzle				
Note	--				
Repair Welding Procedure		Data	--		
		Contractor	JSW		

Step	Work	Details
1	Detectionof flaws	PTand UT
2	Removal of flaws	Machining by portable facer
3	Confirmation of removal	NA
4	DHT	400℃(752F)for 12 hours minimum
5	Welding	- vvelaing repair oT CF-MO maintaining preneat at minimum 15U℃(3U2°)Dy SMAWV Grinding
6	Inspection	MT
7	Weld overlay	Repair of Tp.309/Tp.308 SS weld overlay maintaining preheat at minimum 100℃ (212° F)by SMAW Grinding
8	Inspection	PT
9	PWHT	PWHT at 690℃(1274F)for eight hours Panel heater and insulation blanket were applied.
10	Machining	Machining of gasket groove
11	WOL	Tp.347L SS weld overlay on the final layer of gasket groove without preheating
12	Finishing	Machining of the finalgroove shape by portable facer Finishing of the seating surface by lapping ring
13	Inspection	Dimension test (outer gage and groove gage) PT

Sketch of Repair



Detection of Crack in gasket Groove by NDE (PT&UT) → Removal of Crack by Machining or Grinder → Preheat, Cr-Mo Weld, Grinding, NDE(MT) → Preheat, WOL of Tp.309+308L, Grinding, NDE(PT)



Local PWHT → Machining of Gasket Groove → WOL ofTp.347L → Finish machining of Final layer → Finish by lapping of Gasket Groove, NDE (DT&PT)

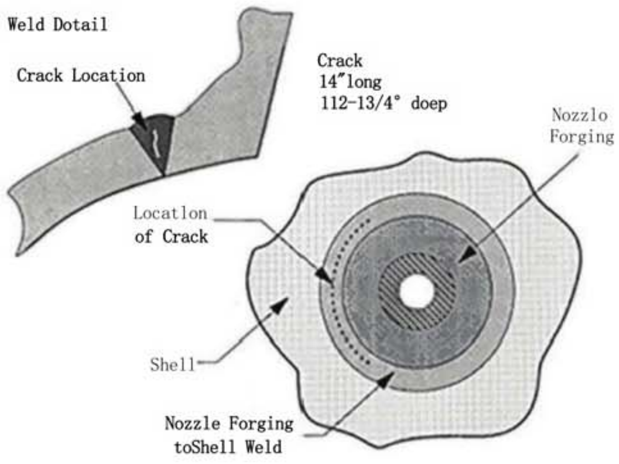
Reference: -

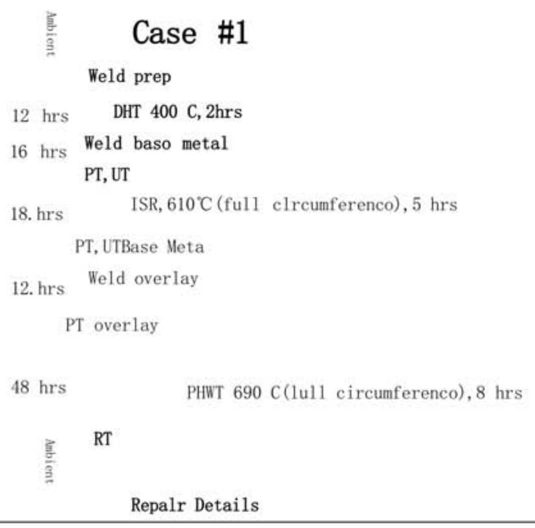
Record 6—Repair of Crack in Nozzle Weld

Title	Reactor Nozzle Repair				
Type of Reactor	Hydrocracker	DT	400℃	Owner	—
		DP	105 kg/cm²		
Operation History	~10 years in service	Base Metal	21/4Cr-1Mo (thk:4.5 in.)		
		WOL/Clad	Type 308L (thk:3/16 in.min)		
Type of Flaws and Locations	Reactor nozzle weld cracking				
Note	---				
Repair Welding Procedure		Data			
		Contractor			

Step	Work	Details
1	Flaw detection	UT from outside
2	DHT	400℃
3	Defect removal	Grinding and air arc gouging
4	Verificationof defect removal	MTof gaged area
5	Repair welding	-150℃ preheat -Base metal welding:SMAW,E-9018-B3,150℃ minimum preheat maintained during entire welding -PT,UT
6	ISR	610℃(full circumference),five hours
7	Inspection	PT,UT
8	Weldoverlay	-Preheat first layer 100℃,other layers no preheat -First layer with E-309L,SMAW -PT inspection at ambient temperature -Subsequent layers with E-308L,SMAW
9	Surface preparation	Grinding
10	Inspection weld overlay	PT
11	PWHT	690℃,full circumference,eight hours
12	Inspection	PT,UT

Sketch of Repair





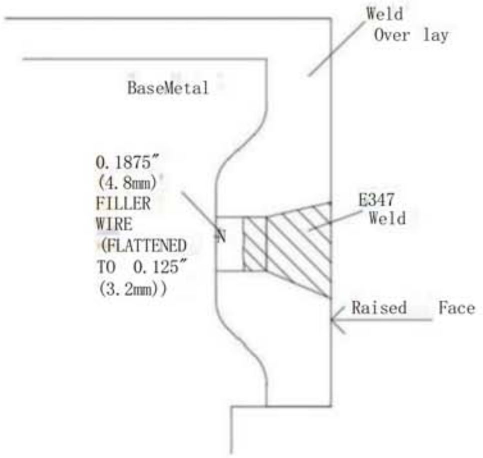
Reference:	—

Record 7-Repair of Outlet Nozzle RTJ and Conversion to RF Flange

Title	Repair of Bottom Outlet Nozzle RTJ Groove Crack and Conversion to RF Flange				
Type of Reactor	Unicracking reactor	DT	835F	**Owner**	
		DP	2215 psi		
Operation History	Operated for 35 years	**Base Metal**	21/4Cr-1Mo		
		WOL	Type 309/347 SS WOL(thk:varied in RTJ)		
Type of Flaws and Locations	Cracking at root of RTJ ring;intermittent around circumference;confinedto overlay				
Note	--				
RepairWelding Procedure		**Data**			
		Contractor			

Step	**Work**	**Details**
1	NDE	PT to identify crack locations;UT examine to check depth of cracking.
2	Removal ofcracks	Hand grinding several separated cracks with frequent checking with PT to monitor for complete crack removal.
3	Confirmation of removal	Visual and PT
4	Measure depth of ground out grooves	Depth gage and ruler to confirm depth of grinding into base metal did not exceed pre-determined amount(max 1/4-in.into base metal).
5	Apply preheat	Apply and maintain 150F preheat throughout welding.
5	Insert braided wires	Type 308/309 welding wires twisted together and resulting braids pushed into ground out areas to fill to a minimum of 1/8-in.beyond the base metal-to-overlay interface.
6	Tack welding	Tack welded the wires in place being certain to stay away from base metal and weld only to overlay.
7	Visual inspection	Visual inspection by PV inspector to assure preparation for weld-out.
8	Initial welding and weld-out to fll grinding grooves	Bridge across braided wire using lowheat input GTAW welding(T 347 wire)to create a solid base for weld-out.Apply layers of Type 347 to fillin grinding grooves using GTAW.
9	Inspection	Visual and NDE(PT)of weld repair at that point to assure no cracking.
10	Welding	Apply initial layer of GTAW Type 347 followed by additional layers of GTAW/SMAW weld build-up to fill in flange ring and achieve specified dimensions.
11	Preliminary machining	To achieve flat surface.
12	NDE	UT of repair area from nozzle ID to assure no crack indications.
13	Final welding	Apply final passes of T347 to bring flange face above nominal dimensions for raised-face flange.
14	Machining and inspection	Automatic machining to achieve required flange face dimensions and finish.Intermediate PT,visual,and PMI.
15	Finalinspection	Visual,PT,UT,PMI,dimensional,surface finish.
16	Convert mating pipe flange (Type 347L)to RF	Weld-out ring and face to required RF dimension.No preheat.Visual and NDE (PT)as welding progressed.Final machining and inspection(visual,PT,dimensional,surface finish).

Sketch:



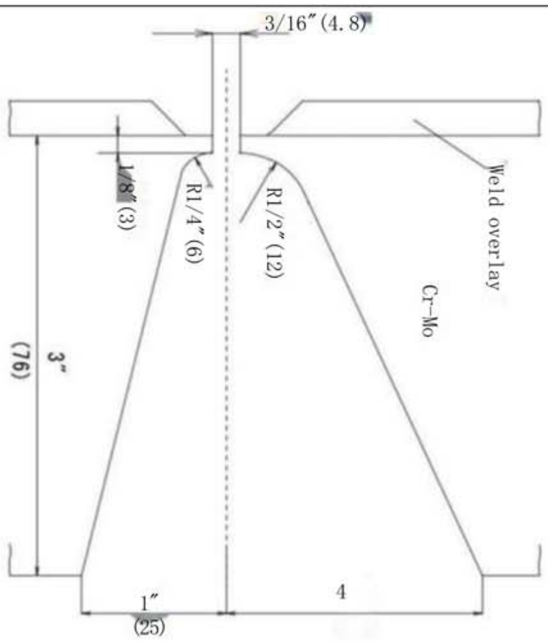
Reference	—

Record 8—Full Replacement of Bottom Nozzle Flange

Title	Full Replacement of Bottom Nozzle Flange				
Type of Reactor	Unicracking reactor	DT	835F	Owner	
		DP	2215 psi		
Operation History	Operated for 20 Years	Base Metal	21/4Cr-1Mo;nozzle thk:3 in.		
		WOLClad	Type 309/347 SS WOL;thk:3/16"		
Type of Flaws and Locations	5-in. long crack at base of RTJ ring groove that extended approx.1 in. into the base metal				
Note	---				
Repair Welding Procedure		Data			
		Contractor			

Step	Work	Details
1	Preheat for cutting	Heat to 212F and hold throughout cutting and weld prep machining.
2	Remove flange	Cut off old flange using air-arcing at predetermined location.
3	Weld prep	Bevel the weld joint and strip the overlay back sufficient for base metal welding.
4	Outgassing	Increase temperature to 700F to 750F and hold for two hrs.minimum.Air cool to ambient.
5	Inspection	VT of beveled surfaces.$CuSO_4$test to assure complete WOL removal.
6	Preheat for welding	Heat to 400F and hold through all base metal welding.
7	Welding	Welded using E9018-B3 electrodes with matching GTAWroot.
8	DHT	Increase temperature to 700F to 750F and hold for two hrs.minimum.Aircool to ambient.
9	Inspection	RT,UT,and MT of weld.
10	Preheat for WOL	Heat to 220F and hold.
11	Apply WOL	Apply one layer of T309L followed by second layer of T 308L.Min total thickness:3/16 in.
12	PWHT	Increase temperature to 1285F+/-25° F and hold for eight hrs.Air cool to ambient.
13	Inspection	Hardness check,UT,RT,MT,PT.
14	Preheat forfinal WOL pass	Heat to 150F and hold.
15	Apply WOL	Apply layer of T347.
16	Hydrotest	Local hydrotest of flange weld only.Water temperature:175F to 190F.
17	Inspection	UT,MT,PT.

Sketch of Repair



Notes:Values in parentheses represent mm.

Reference:	

14 Industry Codes,Standards,and References

API510,Pressure Vessel Inspection Code:In-service Inspection,Rating,Repair,and Alteration

API 579-1/ASME FFS-1,Fitness-for-Service

API Recommended Practice 571,Damage Mechanisms Affecting Fixed Equipment inthe Refining Industry

API Recommended Practice 572,Inspection Practices for Pressure Vessels

API Recommended Practice 582,Welding Guidelines for the Chemical,Oil,and Gas Industries

API Recommended Practice 934-A,Materials and FabricationRequirements for 21/4Cr-1Mo,21/4Cr-1Mo-1/4V, *3Cr-1Mo and 3Cr-1Mo-1/4V Steel Heavy Wall Pressure Vessels for High-temperature,High-pressure Hydrogen Service*

API Technical Report934-B,Fabrication Considerations for Vanadium-Modified Cr-Mo SteelHeavy Wall Pressure Vessels

API Recommended Practice 934-C,Materials and Fabrication of 11/4Cr-1/2Mo Steel Heavy Wall Pressure *Vessels for High-pressure Hydrogen Service Operating at or below 825°F(440℃)*

APITechnical Report 934-D,Technical Report on the Materials and Fabrication /ssues of 11/4Cr-1/2Mo and 1Cr-1/2Mo Steel Pressure Vessels

API Recommended Practice 939-C,Guidelines for Avoiding Sulfidation (Sulfidic)Corrosion Failures in Oi *Refineries*

API Recommended Practice 941,Steels for Hydrogen Service at Elevated Temperatures and Pressures in *Petroleum Refineries and Petrochemical Plants*

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ASME Boiler and Pressure Vessel Code (BPVC),Section III,Materials,Part D,Properties

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ASME PCC-2,Repair of Pressure Equipment and Piping,Part 2,Article 2.14,Field Heat Treating of Vessels

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BS 7910[3],Guide to Methods for Assessing the Acceptability of Flaws in Metallic Structures

JWES-CP-0902E[4] /WRC Bulletin 566[5],Guidelines for Repair Welding of Pressure Equipment in Refineries and
Chemical Plants

NACE Standard Practice SP0170[6],Protection of Austenitic Stainless Steels and Other Austenitic Alloys from *Polythionic Acid Stress Corrosion Cracking During Shutdown of Refinery Equipment*

WRC Bulletin 240,Hydrogen Embrittlement of Austenitic StainlessSteel Weld Metal With Special Consideration Given To The Effects Of Sigma Phase

WRC Bulletin 452,Recommended Practices for Local Heating of Welds in Pressure Vessels

WRC Bulletin 519,Stainless Steel Weld Metal—Prediction of Ferrite Content:An Update of WRC Bulletins 318 *and 342*

2 ASTM International,100 Bar Harbor Drive,P.O.Box C700,West Conshohocken,Pennsylvania 19428,www.astm.org.

3 BSI,389 Chiswick High Road,London,W44AL,UK.

4 Japan Welding Engineering Society,4-20,Kanda Sakumacho,Chiyoda-ku,101-0025 Tokyo,Japan.

5 Welding Research Council,345 East 47th Street,Room 801,New York,New York 10017,www.forengineers.org.

6 AMPP(the Association for Materials Protection and Performance,formerly NACE International),15835 Park Ten Place, Houston,TX77084,www.ampp.org.

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