General Electric Advanced Technology Manual

Chapter 4.9

Intergranular Stress Corrosion Cracking

Table of Contents

Tables

Figures

4.9 Intergranular Stress Corrosion Cracking

Learning Objectives:

1. Recognize the potential consequences of component failures caused by intergranular stress corrosion cracking (IGSCC) both during normal operation and accident conditions.

2. Recognize the mechanism for IGSCC and the conditions that make IGSCC more likely.

3. Identify the primary strategies used by licensees to limit the likelihood of IGSCC and ensure adequate system reliability and component integrity.

4. Recognize the factors used to establish a susceptibility ranking to shroud cracking.

5. Identify the inspection methods used to detect degradation from IGSCC.

4.9.1 Introduction

Corrosion is the weakening of a structural component as a result of a material deterioration caused by electrochemical reaction with the surrounding medium. The effects can be global or highly localized. Global effects are referred to as general corrosion. The localized effects usually involve some form of crack development. The water purity in early BWRs was well below that required today. Water impurities such as chlorides and sulfates accelerated corrosion of all types.

Intergranular stress corrosion cracking (IGSCC) is a form of corrosion found primarily in reactor pressure vessels. IGSCC occurs in a number of metallic alloys. In pressurized water reactors (PWR) IGSCC is found in a few different alloys. In BWRs it is almost exclusively found in stainless steel alloy 304. Stainless steel 304 was used for many structures in the General Electric BWR product lines. The structure of greatest concern is the core shroud located inside the reactor pressure vessel. Other important equipment pieces within or connected to the reactor pressure vessel are also areas of concern.

From a BWR perspective a simple explanation of IGSCC is that cracking forms along the grain boundaries in the heat affected zone of austenitic stainless in a reactor environment. BWR IGSCC requires three concurrent conditions:

- Susceptible Material – stainless steel 304
- Conducive environment – hot oxygenated environment
- Tensile Stress – stress left in the heat affected zone on welds

The elimination of any one of these conditions will prevent IGSCC from occurring.

4.9.1.1 Stainless Steel 304

When stainless steel 304 was installed in RPC internals and connections, IGSCC had not been evidenced as an issue. However, as BWRs accumulated significant operations time, IGSCC became evident at foreign and United States reactors. Until that time stainless steel 304 seemed to be an ideal candidate for use in reactors. It was inexpensive and had proven ideal in other environments. An important metal in stainless steels is chromium. In the welding of stainless steel in the reactor environment, chromium can migrate from grain boundaries and thus reduce the corrosion resistance at the grain boundaries. When a weld is cooled down through the temperature range from 1500 to 900 ^{o}F type 304 stainless steel undergoes a sensitization process characterized by chromium depletion at grain boundaries. The sensitization makes austenitic stainless steels susceptible to corrosion attacks. The presence of residual stresses in weld heat affected zones supplies the third requirement for SCC. Most of the SCC failures in BWR internals are found in weld heat affected zones.

4.9.1.2 Hot Oxygenated Environment

BWRs use large quantities of water for moderation and cooling. Due to the large assortment of radioactive species, many water molecules are split making oxygen a concern. The oxygen promotes corrosion. The dissolved oxygen increases the electrochemical potential of type 304 stainless steel and makes it vulnerable to corrosion attacks. BWRs now use hydrogen injection to scavenge free oxygen which suppresses many forms of corrosion including IGSCC.

4.9.1.3 Tensile Stress

Welding of stainless steel 304 can produce tensile stress. Tensile stress is a process where there is a residual force trying to pull the base metal from the welding material. When hot oxygen attacks a chromium depleted grain boundary in an area of tensile stress, the metal weakens and may pull apart creating a crack. The cracks may be microscopic at first. A relentless corrosion process in an irradiated environment has the potential for the cracks to aggregate and form a larger crack, which in time can lead to a complete material failure. Once IGSCC was identified and understood, welding methods were employed that reduced residual stress.

4.9.1.4 Susceptibility

Because most BWR vessel components are made of material that is susceptible to IGSCC, the industry has attempted to establish a susceptibility ranking for each plant which considers (table 4.9-2):

- length of operation,
- water chemistry/conductivity,
- material susceptibility,
- fabrication, and
- radiation fluence.

Shorter operational times, low conductivity reactor coolant water, the use of low carbon materials, minimal surface cold work, low weld residual stresses, and lower fluence levels reduce the likelihood of cracking.

4.9.2 Inspection Methods

There are two methods being employed to locate cracks and to estimate their lengths. The two methods are the specialized visual inspection (VI) and ultrasonic testing (UT).

Specialized visual inspections have primarily been performed on the outside diameter (OD) weld surfaces of the shroud. Inside diameter (ID) surfaces have also been performed, although the presence of other reactor vessel internal components have limited the inspective area or prohibited visual inspections altogether.

Ultrasonic testing examinations in some locations provide the only possible means of examination since the visual inspection accessibility of this region is blocked. One such area is the H2 weld location that is blocked by the core spray piping and spargers.

4.9.3 Most Susceptible Structures

4.9.3.1 Control Rod Drive (CRD) Stub Tubes

A few cases of IGSSC have been reported in the CRD stub tube penetration in the BWR fleet. In all cases, indications were found in furnace sensitized 304 stainless steel material. Weld stress is the only significant stress for this penetration.

If the sensitized regions or the weld between the penetration and housing developed IGSCC, there should be no operational impact since reactor water exists on both sides of the housing. In an extreme case where the housing could be considered deformed, the ability of the housing to support the fuel and the ability of the control blade to insert could be questionable. If a CRD stub tube requires a repair, the most likely repair would be a mechanical sleeve fitting.

4.9.3.2 In-Core Housings

In-core housings are the penetrations and enclosure for nuclear instrumentation inside the core region. An in-core housing IGSCC failure occurred in a foreign plant. The leak was a through wall crack in the in-core housing weld at the RPV bottom head penetration. The repair involved a larger weld surface area on the vessel bottom head. One concern about such a crack is that it might propagate into the bottom head.

4.9.3.3 Recirculation Inlet and Outlet Nozzles

IGSCC has been found in recirculation inlet nozzles. The initiation of IGSCC occurs in the weld which joins the safe ends to the nozzle attachment. A few instances have found some extensions of cracking into the stainless steel safe end nozzle material. IGSCC has also been observed in the 304 stainless steel thermal sleeve of a domestic BWR/3. In a worst case scenario, the crack may grow into the vessel, where service

induced crack growth might cause the crack to reach a critical size where lower temperature operation such as pressure testing could initiate brittle fracture. Margins in operating methods make this scenario unlikely, but the consequences would be severe.

4.9.3.4 Shroud-To-Shroud Support Weld

The shroud support consists of a horizontal plate (in four weld segments) welded on the inside of the vessel. A vertical ring is welded to the support plate which is in turn welded to the shroud. Structural support is added to the support plate by 22 gusset plates welded to horizontal plate and to the vessel wall. If IGSCC initiation occurred, service induced crack growth may cause cracks to grow into the RPV wall. IGSCC at the RPV wall surface could grow in size so that the lower temperature operations like pressure testing could initiate brittle fracture. Margins in operating methods make this scenario very unlikely, but the consequences would be severe from both a safety and economic point of view.

4.9.3.5 Core Shroud

The core shroud is a stainless steel cylinder assembly, Figure 4.9-1, that surrounds the core. The shroud provides the following functions/purposes:

- A barrier to separate or divide the upward core flow from the downward annulus flow.
- A vertical and lateral support for the core plate, top guide and shroud head.
- A floodable volume in the event of a loss of coolant accident.
- A mounting surface for the core spray spargers.
- A core discharge plenum, directing the steam water mixture into the moisture separator assembly.

The core shroud is welded to and supported by the baffle plate (shroud support plate). The upper surface is machined to provide a tight fit with the mating surface of the shroud head. Mounted inside the upper portion of the shroud, in the space between the top guide and the shroud head base, are the two core spray spargers. Typical cross sectional dimensions range from 14 feet to more than 17 feet in diameter with a wall thickness between 1.5 inches to 2 inches. Core shrouds were fabricated from 1.5 inches to 2 inches primarily for stiffness considerations for transport and installation. Boiling Water Reactor (BWR) shrouds are typically manufactured from either plate rings or forged rings of type 304 or 304L stainless steel. Fabrication of the plate portions of the shroud involves both axial and circumferential welds. Fabrication of the ring forging involves only circumferential welds. The circumferential welds in the shroud are identified according to their vertical location as shown in Figure 4.9-1, although the exact numerical notation may vary from plant to plant.

Numerous instances of shroud cracking have occurred in the BWR fleet. The first occurrence of cracking occurred in a BWR/4 located outside the United States. Cracking indications were observed in the circumferential beltline seam weld of the Type 304 stainless steel (with medium carbon content) core shroud. Circumferential crack indications with short axial components were observed in three locations on the inside surface of the shroud and were confined to the heat affected zone of the

circumferential weld. Short, axial indications were also observed on the outside surface of the shroud in the same heat affected zone. Multiple UT examinations have been performed after these indications were found, with the most recent exam finding significant crack growth over a single cycle. An evaluation of cracking was performed and found that the cracking was due to IASCC.

The second instance involved cracking at a domestic GE BWR/4. Crack indications were discovered during in-vessel inspection of reactor internals. Indications of cracking were circumferentially located in the top guide support ring parallel to the plane of the ring and adjacent to the H-3 weld. Indications were also found on the outside surface of the shroud
adjacent to the H-4 weld, oriented axially and measuring about on inch. Crack initiation was found to occur by IGSCC and was accelerated by IASCC contribution.

The third instance of cracking occurred in another domestic BWR/4. Indications were seen in both circumferential and axial directions at the H-3 and H-4 welds. In addition, circumferential indications were observed in the shroud plate associated with the vertical weld.

At the upper shroud elevations (H1, H2, and H3), lifting of a separated shroud is expected to occur due to differential pressure in the core being sufficient to overcome the downward force created by the weight of only a small portion of the remaining upper shroud assembly. As such, bypass flow through the gap created by the separation is sufficient to cause a power/flow mismatch indication in the control room. The main concern associated with cracks in the upper shroud region is during a steam line break. With a main steam line failure, the lifting forces generated may elevate the top guide sufficiently to reduce the lateral support of the fuel assemblies and could prevent control rod insertion.

At the lower shroud elevations (H4, H5), shroud lifting may not occur due to insufficient core pressure differential necessary to overcome the downward force from the weight of the shroud. As such, detection of bypass flow is not assured. The main concern associated with cracks in the lower elevations of the core shroud is the postulated recirculation line break. Recirculation line break loadings, if large enough, could cause a lateral displacement or tipping of the shroud which could affect the ability to insert control rod and may result in the opening of a crack. If the leakage were large enough, it could potentially affect the ability to reflood the core and maintain adequate core cooling flooding. In addition, the ability to shut down the reactor with the Standby Liquid Control System could be reduced.

Other concerns have been raised over the potential for damage to reactor vessel internals due to shroud displacement during postulated accident conditions. In particular, the possibility may exist for damage to the shroud support legs due to impact loading from the settling of the shroud after a vertical displacement. In addition, displacement of the shroud could cause damage to core spray lines.

The NRC developed a probabilistic safety assessment regarding shroud separation at the lower elevation for two plants, Dresden Unit 3 and Quad Cities Unit 1. The staff made conservative estimates of the risk contribution from the shroud cracking and concluded that it does not pose a high degree of risk at this time.

However, the staff considers a 360 degree cracking of the shroud to be a safety concern for the long term based on:

- Potentially exceeding the ASME Code structure margins if the cracks are sufficiently deep and continue to propagate through the subsequent operating cycle.
- The uncertainties associated with the behavior of a 360 degree through wall core shroud crack under accident conditions.
- The elimination of a layer of defense in depth for plant safety.

4.9.3.6 Access Hole Cover

The access cover is a 2 inch thick cover welded to the 2 2 inch thick shroud support. Extensive cracking has been found in several access hole covers in the BWR fleet. Cracking has occurred in creviced covers has initiated in the heat affected zone of the cover plate. Intermittent circumferential cracking has been the most common orientation of cracking.

In the worst case, access hole cover cracking could progress through wall and cause the cover to detach either partially or completely. A substantial flow path from the bottom head into the annulus region would be created, impacting core flow distribution during normal operation. The distribution would be detectable at significant levels. Such cracking would impact the boundary which assures 2/3 core coverage following a LOCA event. The consequence of cracking is high.

General Electric has replaced approximately 20 access hole covers to date with a cost of approximately $6 million per plant.

4.9.3.7 Jet Pump Riser Brace

The jet pump riser brace is connected to the riser pipe by a single bevel weld. At least one occurrence of IGSCC has been documented by General Electric. During visual examination at a BWR/4, a crack was found on the weld that attaches the riser brace yoke to the jet pump riser pipe. Cracking extended out of the heat affected zone of the weld and into the riser pipe. Although no definitive answer was reached, it is believed that the cracking initiated by an IGSCC mechanism and propagated by high cycle fatigue. At the crack location between the brace and the riser, a crack could have significant consequence on operation and safety. The brace is intended to provide structural support at the upper part of the jet pump assembly and lateral support to maintain jet pump alignment

4.9.3.8 Piping Cracks

Piping cracks from IGSCC was identified as early as 1965. In December of 1965, during a hydrostatic pressure test, a leak was observed in a 6 inch bypass line of the recirculation loop at Dresden Unit 1. Like the vessel penetrations and internals the cracks were found in the heat affected zone of welds in type 304 stainless steel. Table 4.9-3 lists the IGSCC incidents by line type in U.S. and Foreign BWRs. The data listed in the table is only good through January of 1979. Many of the cracks found after 1975 were due to the augmented inspections performed.

Following the discovery of cracks in recirculation piping, many utilities have replaced the 304 or 316 stainless steel with 316NG or 316 low carbon steel piping. This data is listed in Table 4.9-4.

4.9.4 Activities

BWR executives formed the BWR Vessel and Internals Project (BWRVIP) in June of 1994. One of the BWRVIP=s first challenges was to address integrity issues arising from service-related degradation of key components, beginning with core shroud cracking. BWRVIP also implemented a proactive program to develop products and solutions that bear on inspection, assessment, mitigation, and repair.

Through BWRVIP, utilities are developing, sharing, and implementing cost-effective strategies and products for resolving vessel and internals integrity and operability problems. BWRVIP also provides the regulatory interface on generic BWR vessel and internals matters. During the first year of BWRVIP activities, the following products were developed for the core shroud: Inspection and Flaw Guidelines, NDE Uncertainty and Procedure Standard, and Repair Design Criteria.

4.9.4.1 Hatch

The design of the Hatch Unit 1 core shroud modification consists of four stabilizer assemblies, which are installed 90 degrees apart. Each stabilizer assembly consists of a upper bracket, tie rod, upper spring, lower spring, lower bracket, intermediate support, and other minor components. The tie rods serve to provide an alternative vertical load path from the upper section to the tie rod assembly through the shroud support plate gusset attachments. These tie rod assemblies maintain the alignment of the core shroud to the reactor vessel. At the top guide elevation, the upper springs are designed to provide a radial load path from the shroud to the RPV. The lower springs are designed to provide a similar radial load path (from the shroud to RPV) at the core support plate elevation. The upper bracket is designed to provide attachment to the top of the shroud, and to restrain the upper shroud weld (weld H1). The middle support for the tie rods is designed to limit the radial movement of the tie rods. Wedges placed between the core shroud plate and the shroud prevent relative motion of the core plate with the shroud.

The stabilizer assemblies are designed to prevent unacceptable lateral or vertical motion of the shroud shell sections, assuming failure (360 degrees through wall) of one or more of the structural circumferential shroud welds. The functions of the components are as follows:

- upper brackets are designed to restrain lateral movement of the shell between welds H1 and H2, and the shell between welds H3 and H4
- the limit stops located at the middle of the tie rods are designed to restrain lateral movement of the shell between welds H4 and H5
- the lower springs contact the shroud, and are designed to restrain the shell segments between welds H5 and H6a, H6a and H6b, and welds H6b and H7
- the gussets, which were originally included as part of the shroud support design, are designed to preclude unacceptable motion of the shroud between welds H7 and H8

Materials for the stabilizer assemblies was selected to provide protection for the life of the plant. In addition, the material has a different coefficient of expansion than the core shroud and causes a compressive load when at normal temperature and pressure.

4.9.4.2 Hydrogen/Zinc Injection

Protection against IGSCC deals mainly with some form of primary water chemistry control process. Hot oxygenated water creates a corrosive environment in the BWR pressure vessel. Dissolved oxygen in water increases the electrochemical potential of type 304 stainless steel and makes them

The purpose of hydrogen water chemistry control is to suppress oxygen in the reactor water. By suppression the oxygen level in reactor water:

- general corrosion is controlled,
- characteristics of corrosion film layer in recirculation piping and reactor vessels are changed, and
- a reduction in the oxidation state of chromium is realized.

In response to the unacceptable degradation of reactor vessel components from Intergranular Stress Corrosion Cracking (IGSCC) a number of BWRs have adopted hydrogen water chemistry. Hydrogen water chemistry implies a low dissolved oxygen content coupled with low conductivity.

Hydrogen water chemistry appears to improve the margin for stress corrosion and corrosion fatigue of carbon and low alloy steels, but has a slight adverse affect on their overall corrosion kinetics.

Under hydrogen water chemistry, the dissolved oxygen in the recirculation systems decreases below the acceptable value for minimal corrosion of carbon steel piping. At very low levels of dissolved oxygen the protective corrosion film on carbon steel undergoes dissolution and produces accelerated corrosion of the base metal. Therefore, sufficient oxygen is added to the condensate system to maintain oxygen between 20 and 50 ppb.

Hydrogen water chemistry provides a reducing nvironment that not only lowers the oxidation potential of reactor water, but also favors formation of spinel. Spinel is a thinner, more adherent film, of a complex metal matrix consisting of iron, chromium, nickel, cobalt, manganese, copper and zinc.

Historically, the corrosion films on BWR components are a combination of hematite and spinel oxides. Higher fractions of hematite in the corrosion film lead to thicker and less protective oxides. This type of corrosion film tends to increase radiation buildup by permitting more corrosion products to enter solution. This tendency is counter balanced because hematite does not have a natural site for crystal formation by divalent ions, such as cobalt. Hematite has a lower cobalt concentration than corrosion films dominated by spinel structure. This means that the radioactive material buildup is not controlled solely by oxide layer thickness.

BWR chemistry without hydrogen water control provides oxidizing conditions in the reactor coolant. Under oxidizing conditions, stable oxygen-16 is activated to nitrogen-16 by a neutron-proton reaction. The resulting nitrogen-16 is primarily in the form of soluble nitrates (NO_3) and nitrites (NO_2) with a small amount in the form of volatile ammonia (NH_4).

Hydrogen water chemistry changes the BWR coolant to a reducing environment. Under reducing conditions, the chemical equilibrium shifts from nitrate/nitrite in favor of volatile ammonia. Nitrogen-16 carryover into the main steam system then increases by as much as a factor of five at full power. The carryover of nitrogen-16 results in significant increased dose rates in the turbine building during plant operation from 6.1 and 7.1 Mev gamma photons produced during radioactive decay. During outages, the dose rate from nitrogen-16 is not a factor since it is no longer being produced and it has a very short half-life of only 7.1 seconds.

The presence of zinc in the reactor coolant increase the spinel fraction in oxide formations on stainless steels. Spinel is a thinner (by a factor of six or more) more protective film oxide than hematite (Fe_2O_3). The corrosion protection provided by spinel based film is greater than that formed by divalent cations commonly found in BWRs. Zinc competes with cobalt for available crystal lattice sites in the spinel and under hydrogen water chemistry is the dominate divalent ion in the crystal matrix of Spinel; thereby, allowing little cobalt-60 buildup. It is hypothesized that the excess of zinc ions in a mixed metal oxide migrate to the vacant defect sites and block ion migration by other ions. This produces a quasi-stoichiometric oxide that is highly protective to the base metal.

Reducing the soluble cobalt-58 and cobalt-60 in the in the reactor coolant is an additional benefit. By reducing the long lived radioactive material that contribute to personnel exposure, BWRs see a positive impact in ALARA space.

4.9.4.3 Nobel Metals Injection

Noble metals, platinum and rhodium, injection has proven that it works. To reduce the hydrogen addition rate and lower the radiation levels, due to hydrogen addition, General electric developed NobleChem (Noble Metals Injection). The idea of noble metals injection is to increase the efficiency of hydrogen-water chemistry. There are about 20 BWRs in the fleet that are using or plan on implementing noble metals injection. The catalytic deposited layer provides the desired electrochemical corrosion potential levels for many components at a very low hydrogen injection level and extends hydrogen water control benefits to additional vessel internals. With the use of noble metals injection, approximately one-fifth of the hydrogen injection values used in traditional hydrogen injection are needed.

The general process adds a platinum and rhodium noble metal compound to the reactor water until the concentration is 40-100 ppb platinum and 20 - 150 ppb rhodium. Injection of the noble metal solution is into the Recirculation Loop A discharge line and the B RHR system downstream of the heat exchanger through existing small bore piping connections. The RHR system takes suction from the

recirculation loop B and is returned to the same loop. Consequently, the RHR system will provide the drive flow for the B loop jet pumps. Recirculation pump A will be in operation to provide drive flow to the other jet pumps. The two flows are balanced as equal as possible to assure distribution of the noble metals compound to each loop and circulate water in the vessel.

The process is normally applied during the normal cooldown sequence prior to refueling outage. The vessel water temperature at which the process will be applied is 265^0F $\pm$ 25ºF. The process requires the vessel water temperature to maintained for 48 hours. Decay heat will be used to maintain the water temperature at the desired process temperature. Excess heat is remove by operating the RHR system in the shutdown cooling mode. To prevent excessive deposition on the hotter fuel clad surfaces during treatment, the fuel cladding temperature needs to be within 20^0F of the bulk coolant temperature prior to starting the process.

Surfaces that come into contact with the reactor water during the process will have a target minimum loading of 1 microgram per square centimeter for platinum and 1/3 microgram per centimeter of rhodium with a maximum of 50 and 17 respectively. Surface samples of specimens tested in autoclaves have shown that the noble metal atoms present on the surface do not completely cover the surface but are distributed randomly across the surface. Consequently, the surface is not plated and the Pt/Rh layer is discontinuous. Based on General Electric laboratory data, if gaps larger than 0.1-1 mm in the noble metal coverage exist, they will not be protected locally. If cracks develop in these regions, the lower electrical chemical protection of the adjacent noble metal regions will arrest the cracks after a microscopic amount of crack growth.

General Electric has studied the behavior of stress corrosion cracks ranging in size from 20 micrometers to 40,000 micrometers and found that a mature crack is established in cracks less than 20 micrometers deep. There is widespread agreement that what produces the mature crack and usually aggressive crack chemistry is the difference in

corrosion potential between the crack/crevice mouth and crack/crevice interior, known as a differential aeration cell. Numerous studies have shown that essentially the entire potential gradient occurs very near the crack mouth-perhaps in the first 5%of the crack crevice. If this potential gradient is substantially eliminated by excellent hydrogen water chemistry or noble metals injection, then it makes no difference how long the crack/crevice is since the driving force that produces an aggressive crack chemistry is no longer present. These same characteristics have also been shown to exist under high flux irradiation conditions.

Platinum and Rhodium serve as sites for recombination of hydrogen and oxidants. The noble metal surfaces are chemically benign in the BWR environment and have little to no effect on the water concentration of hydrogen and oxygen.

During normal operation, the noble metal on the surface will prevent and mitigate stress corrosion cracking by reducing the oxidant concentration near the metal surface. The catalytic behavior of noble metals provides an opportunity to efficiently achieve a dramatic reduction in corrosion potential and stress corrosion cracking by catalytically reacting all oxidants that contact the catalytic surface with hydrogen. With stoichiometric excess hydrogen, corrosion potential decreases dramatically and crack initiation and growth are greatly reduced, even at high oxygen and hydrogen peroxide levels. Low hydrogen addition rates are necessary to provide sufficient hydrogen at the surface of noble metal treated components. Oxygen that diffuses to the component surface will immediately react with the excess hydrogen to form water. In this way the boundary layer of all noble metal wetted components is depleted of oxygen and maintains a very low corrosion potential. Noble metal utilizes very reactive surfaces to maintain oxygen deficient water in contact with reactor components. Moderate to high hydrogen water chemistry control, on the other hand, are brute force methods to reduce the oxygen content of the entire bulk coolant to be effective and increase the main steam line radiation levels.

Table 4.9-1 Shroud Cracking Experiences

Plant	Date of Operation	Summary
Brunswick 1	03/18/77	360° circumferential crack at H3 weld in the top guide support ring, 0.8" to 1.7" deep. Less significant circ. and axial cracks at H1 to H6. The H2 and H3 welds were repaired with 12 through-bolt clamps.
Brunswick 2	11/3/75	Significant cracking was observed during visual inspection (VI).
Dresden 3	11/16/71	The H1 through H7 welds have been inspected. 360° cracking on OD at weld H5, significant cracking indicated on the ID at weld H3. Safety evaluation (issued July 20, 1994) has allowed operation for no more than 15 months.
Duane Arnold	02/1/75	An ID examination was performed in accordance with the recommendations of GE SIL, discovering no indications of cracks. The plant has an L-grade shroud and uses hydrogen water control.
Fermi 2	01/23/88	Minor axial indications were discovered at H2 weld.
Hatch 1	12/31/75	The licensee installed a preemptive shroud repair in lieu of inspection and potential evaluation of identified flaws.
Hope Creek	12/20/86	A limited examination has been performed with the discovery of no cracks. The plant has L-grade shroud.
Millstone	03/01/94	Minor circumferential cracks present at H3, H4, and H5 weld locations. No repair has been implemented.
Monticello	06/30/71	Licensee completed a UT and enhanced VI of accessible welds. Minor indications observed at H2, H3, and H4
Nine Mile Pt. 2	03/11/88	Shroud is fabricated from low carbon stainless steel. Plant is outside the scope of GE SIL recommendations.
Oyster Creek	12/1/69	Licensee completed inspection in 1994 refueling outage. Minor circumferential indications on H2, H6a, and H6b welds. Extensive cracking on OD and ID of H4. Licensee is installing shroud repair.

Table 4.9-2 BWRVIP Susceptibility Rankings and Core Shroud Inspection Recommendations

Category	Inspection Recommendations	Plant Characteristics	Plants
A	No inspection necessary at this time.	Plants with 304 SS shrouds, <6 years hot operating time, and avg. conductivities #0.030uS/cm (0.030umhos/cm) during the first five cycles of operation.	None
		Plants with 304L SS shrouds,<8 years hot operating time, and avg. conductivities #0.030uS/cm (0.030 umhos/cm) during the first five cycles of operation.	Clinton, Fermi 2, Perry, Hope Creek, Limerick 2, Nine Mile Pt .2, Washington Nuclear Plant 2, River Bend
B	Limited inspection: top guide support ring, core support ring, and mid shroud shell circumferential welds; also the bimetallic weld if accessible.	Plants with 304L SS shrouds, ∃8 years hot operating time, and avg. conductivities #0.030uS/cm (0.030 umhos/cm) during the first five cycles of operation	Grand Gulf, LaSalle 1 & 2 Limerick 1, Susquehanna 1 & 2
C	Comprehensive inspection: Circumferential shroud welds H1-H7 (and H8 for BWR-2s)	Plants with 304SS shrouds and ∃ 6 years hot operating time, regardless of conductivity.	<u>Shrouds-weld, plate rings</u> Bruinswick 1& 2, Dresden 2 & 3, Hatch 1, Millstone 1, Oyster Creek, Nine Mile Point 1, Pilgrim, Quad Cities 1 & 2, FitzPatrick
			<u>Shrouds- Forged rings</u> Browns Ferry 1, 2 & 3, Peach Bottom 2 & 3, Vermont Yankee, Monticello, Cooper
		Plants with 304L SS shrouds, ∃8 years hot operating time, and avg. conductivities >0.030uS/cm (0.030 umhos/cm) during the first five cycles of operation	Duane Arnold, Hatch 2

Table 4.9-3 IGSCC Incidents by Line Type in U.S. and Foreign BWR's[a]

	Number of IGSCC Incidents		
System Component (Pipe Diameter)	Before July 1975	July 1975 to January 1979	Totals
Recirculation Bypass Line (4”)	30	12	42
Core Spray Pipe (10”)	16	17	33
CRD Small Bore Pipe (3”)	1	1	2
Reactor Water Cleanup (3” to 8”)	10	14	24
Large Recirculation System (12”)	0	13	13
Small Bore Pipe (3”) other than CRD & RWCU	0	6	6

[a] Cracking Incidents reported to the NRC

Table 4.9-4 Status of U. S. BWR Piping

Plant		Date of Operating License	Original Design Material	Replacements for Recirculation Piping	Replacements for RHR Piping[b]	Hydrogen Water Chemistry Implemented?
Design	Name					
BWR/2	Nine Mile Point 1	12/26/74	304SS or 316SS	Full, 316SS (low carbon)	Full, 316SS (low carbon)	Yes
	Oyster Creek	08/01/69	304SS or 316SS	None	None	Yes
BWR/3	Dresden 2	02/21/70	304SS or 316SS	None	None	Yes
	Dresden 3	03/02/70	304SS or 316SS	Full, 316NG	Full, 316NG	No
	Millstone 1	10/31/86	304SS or 316SS	None	None	Yes
	Monticello	01/09/81	304SS or 316SS	Full, 316NG	Full, 316NG	Yes
	Pilgrim	09/15/72	304SS or 316SS	Full, 316NG	Full, 316NG	Yes
	Quad Cities 1	12/14/72	304SS or 316SS	None	None	Yes
	Quad Cities 2	12/14/72	304SS or 316SS	None	None	Yes
BWR/4	Browns Ferry 1	12/20/73	304SS or 316SS	None	None	No
	Browns Ferry 2	08/02/74	304SS or 316SS	Part[a] (riser) 316NG	None	No
	Browns Ferry 3	08/18/76	304SS or 316SS	None	None	No
	Brunswick 1	11/12/76	304SS or 316SS	Part[a] (riser) 316NG	None	Yes
	Brunswick 2	12/27/74	304SS or 316SS	Part[a] (riser) 316NG	None	Yes

Table 4.9-4 Status of U. S. BWR Piping (cont.)

Plant		Date of Operating License	Original Design Material	Replacements for Recirculation Piping	Replacements for RHR Piping[b]	Hydrogen Water Chemistry Implemented?
Design	Name					
BWR/4	Cooper	01/18/74	304SS or 316SS	Full, 316NG	Full, 316NG	Yes
	Duane Arnold	02/20/74	304SS or 316SS	None	None	Yes
	Fermi 2	07/15/85	304SS or 316SS	None	None	Yes
	FitzPatrick	10/17/74	304SS or 316SS	None	None	Yes
	Hatch 1	10/13/74	304SS or 316SS	None	None	Yes
	Hatch 2	06/13/78	304SS or 316SS	Full, 316NG	Full, 316NG	Yes
	Hope Creek	07/25/86	316NG REC, RHR, RWCU	N/A	N/A	Yes
	Limerick 1	08/08/85	316NG REC,RHR, Core Spray, RWCU	N/A	N/A	Yes
	Limerick 2	08/25/89	316NG REC,RHR, Core Spray, RWCU	N/A	N/A	Yes
	Peach Bottom 2	12/14/73	304SS or 316SS	Full, 316NG	Full, 316NG	Yes
	Peach Bottom 3	07/02/74	304SS or 316SS	Full, 316NG	Full, 316NG	Yes
	Susquehanna Unit 1	11/12/82	304SS or 316SS	None	None	Yes
	Susquehanna Unit 2	06/27/84	304SS or 316SS	None	None	No
	Vermont Yankee	02/28/73	304SS or 316SS	Full, 316NG	Full, 316NG	No

Table 4.9-4 Status of U. S. BWR Piping (cont.)

Plant		Date of Operating License	Original Design Material	Replacements for Recirculation Piping	Replacements for RHR Piping[b]	Hydrogen Water Chemistry Implemented?
Design	Name					
BWR/5	La Salle 1	08/13/82	304SS or 316SSL[c]	None	None	Yes
	La Salle 2	03/23/84	304SS or 316SSL[c]	None	None	Yes
	Nine Mile Point 2	07/02/87	316NG for All Piping Systems	N/A	N/A	No
	WNP 2	04/13/84	304SS or 316SS	None	None	No
BWR/6	Clinton 1	04/17/87	316NG for REC, RWCU	N/A	None	No
	Grand Gulf 1	11/01/84	304SS or 316SS	None	None	Yes
	Perry 1	11/13/86	304SS or 316SS	None	None	No
	River Bend 1	11/20/85	316NG for REC	N/A	None	No

Notes:

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

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

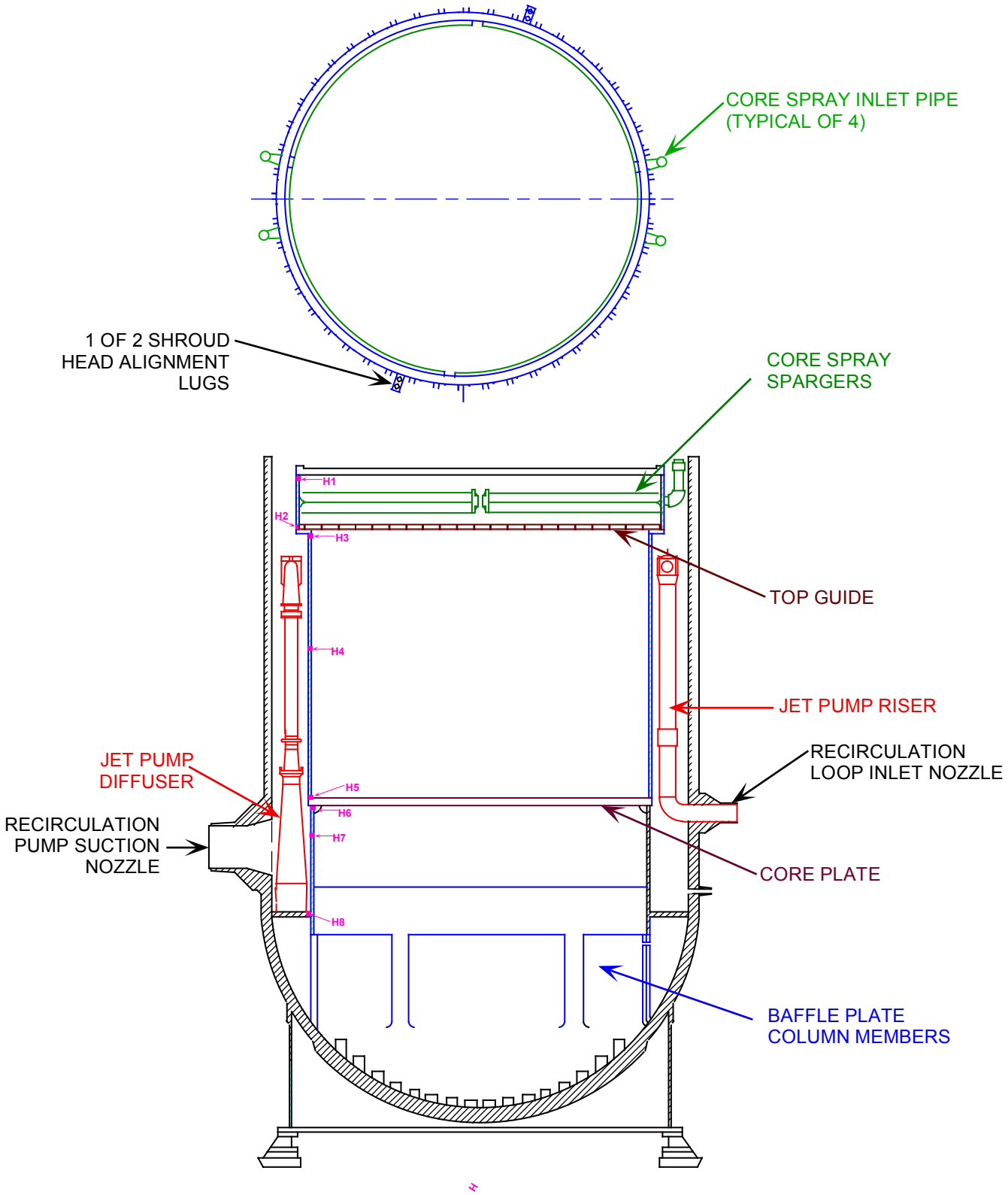

Figure 4.9-1 Core Shroud

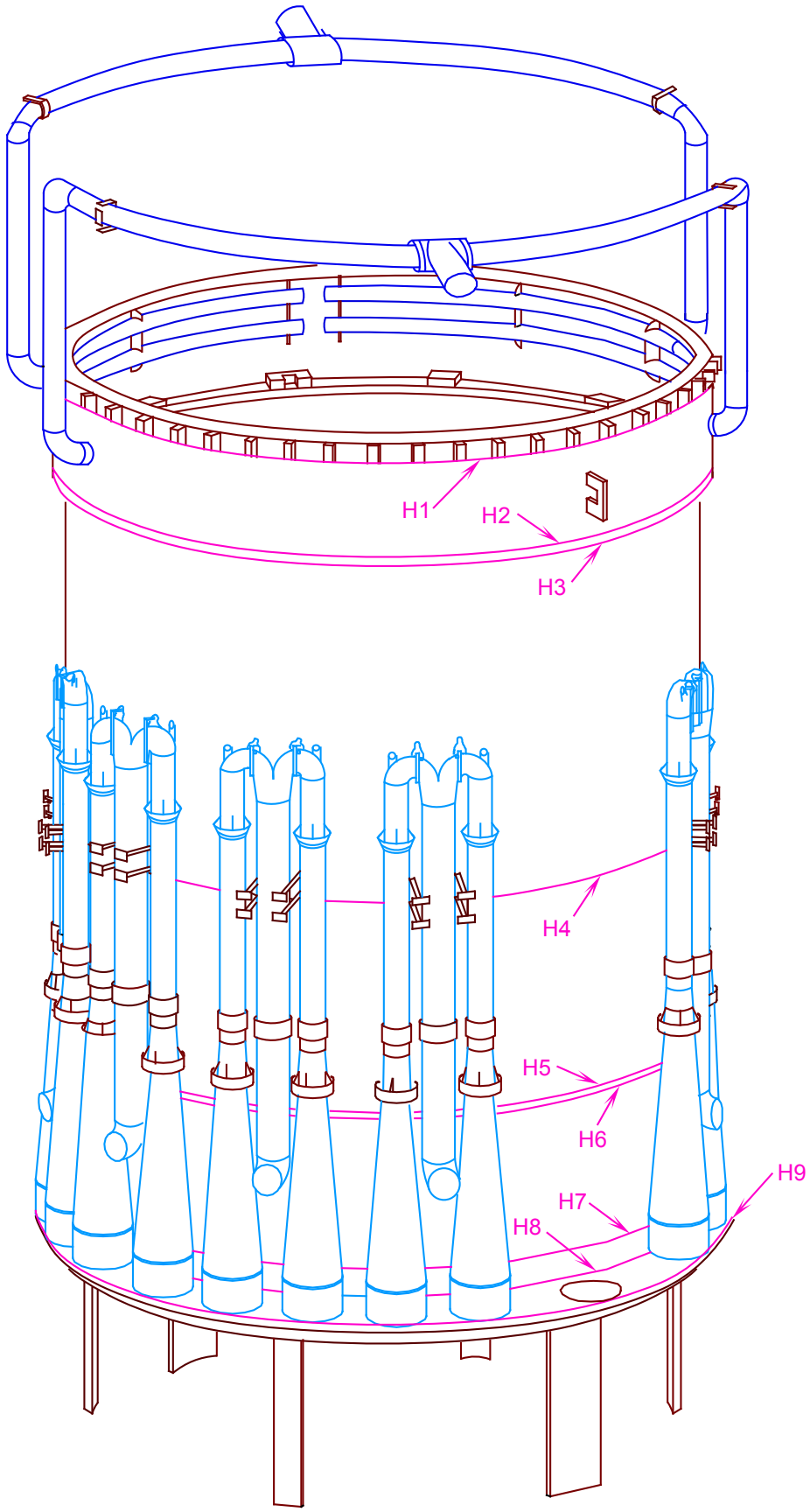

Figure 4.9-2 Core Shroud Weld Location

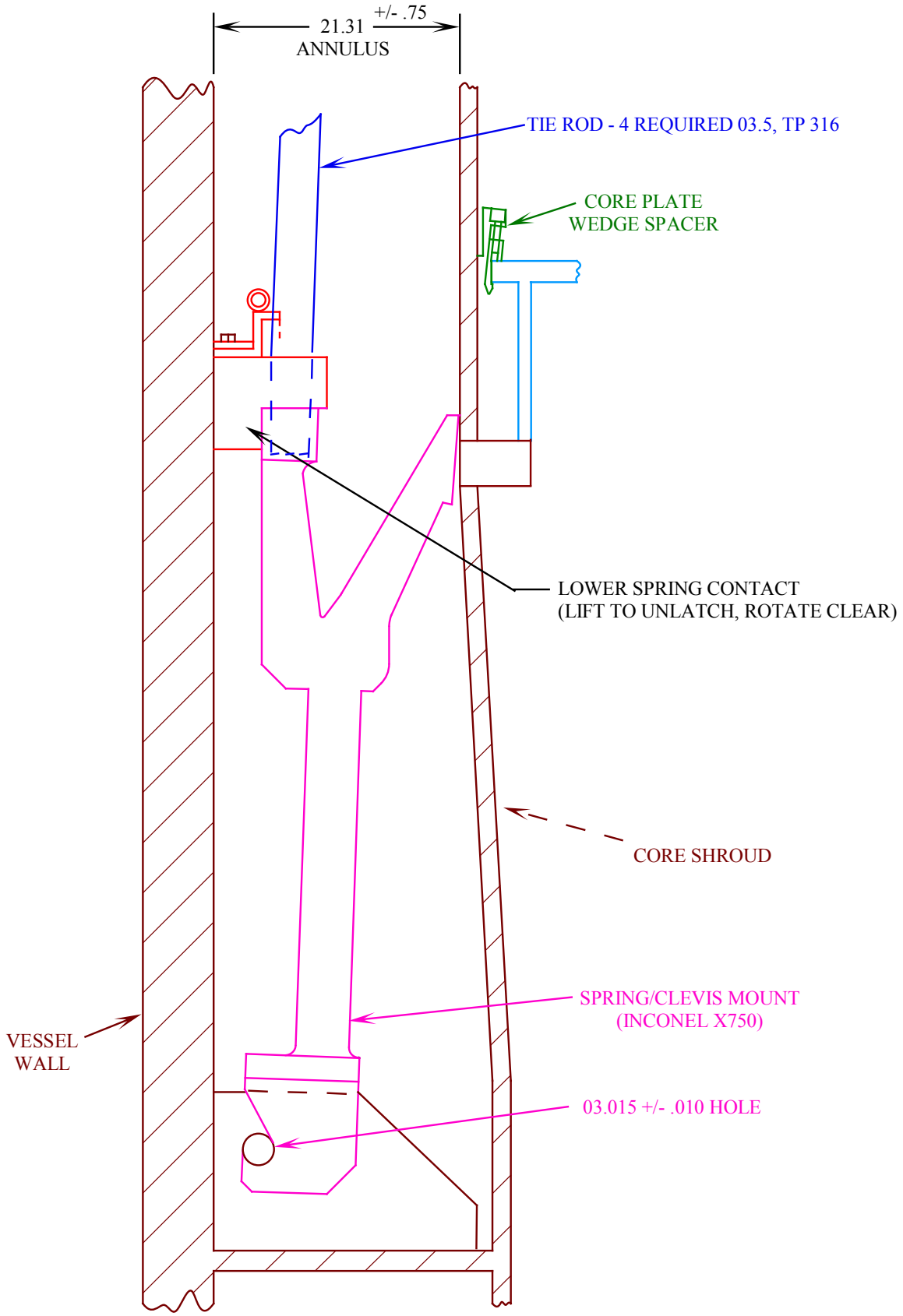

Figure 4.9-3 Lower Shroud Clamp

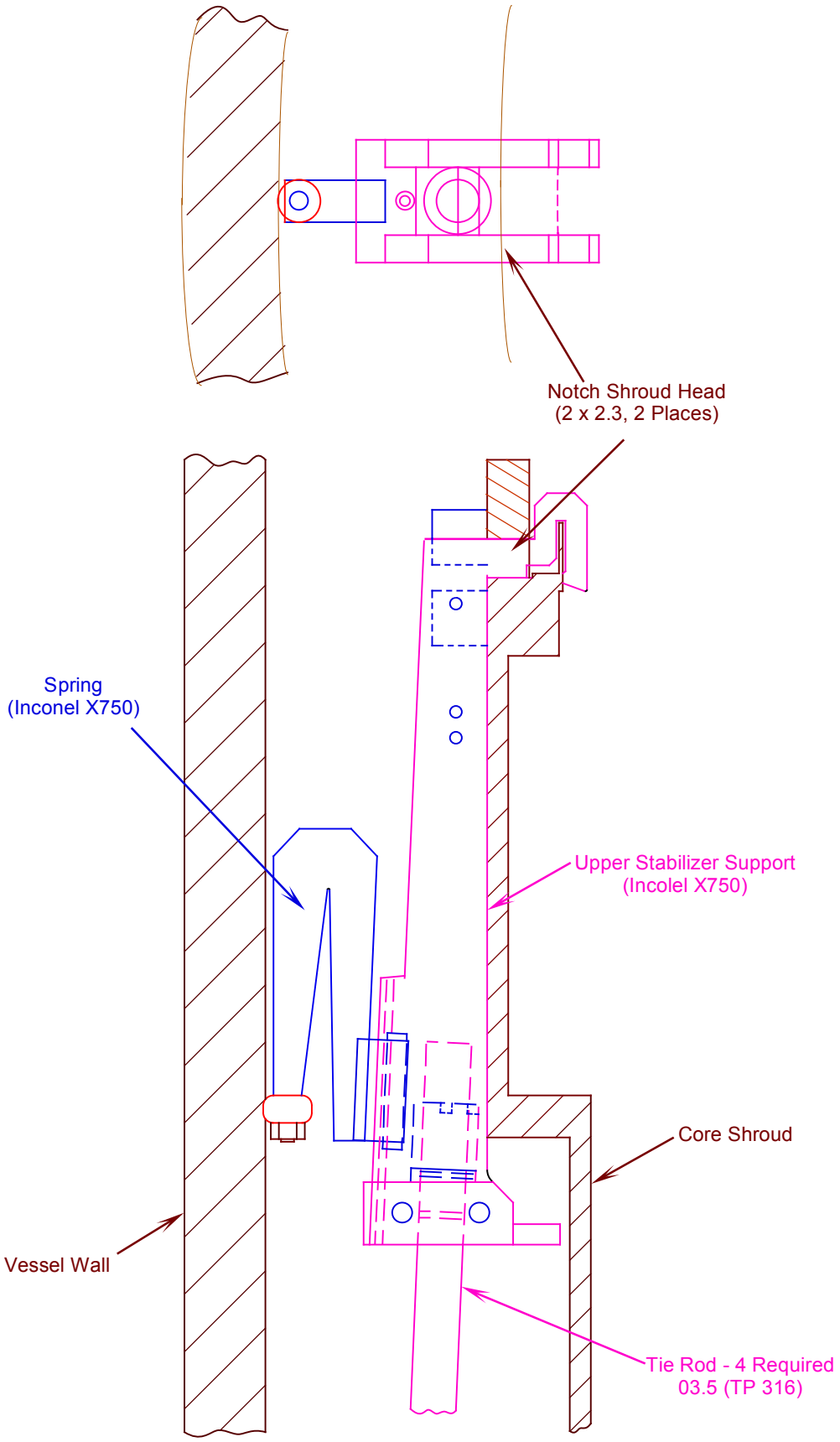

Figure 4.9-4 Upper Shroud Clamp