SMILE Project - Studies on Irradiation Embrittlement and Irradiation Assisted Stress Corrosion Cracking of Core Support Structures and Internals

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Abstract

The main objective of the SMILE (Studsvik Material Integrity Life Extension) project is to provide critical data and mechanistic understanding of materials ageing in support of plant ageing management, life extension programmes and operating license renewals [1)]. One of the tasks of SMILE concerns embrittlement and irradiation-assisted stress corrosion cracking (IASCC) of core support structures and reactor pressure vessel (RPV) internals. The present paper describes the approach and scope to achieve the objectives of this task. The library of aged materials harvested from recently retired Swedish boiling and pressurized water reactors, the various tests and examinations conducted to quantify irradiation-induced aging, including some results, are presented and discussed.

Introduction

Core support structures of light water reactors (LWR) are intended to remain in place for the entire design life of the plant and, consequently, they can be exposed to considerable neutron doses. In pressurized water reactors (PWRs), maximum neutron doses to the most highly irradiated zones can be of the order of ~80 to 100 dpa (displacements per atom) after an initial 40-year design life. In boiling water reactors (BWRs), maximum neutron doses to the nearest support structures are typically an order of magnitude smaller. Neutron irradiation gives rise to embrittlement of the stainless steels used for these components. Evidence of embrittlement begins to show at neutron doses between 0.1 and 1 dpa and many effects saturate between 5 and 10 dpa [e.g., 2)]. The embrittlement is characterized by significant increases in yield strength and large reductions in ductility [3)] as well as fracture toughness [4)].

Transmutation reactions caused by neutron irradiation also generate considerable quantities of hydrogen and helium. The diffusivity of hydrogen in austenitic alloys is quite rapid at LWR operating temperatures and, depending on component thickness, it will therefore equilibrate relatively quickly with any hydrogen dissolved in the aqueous coolant as well as with any hydrogen evolved by general corrosion. However, helium is highly insoluble in austenite so that it forms a large number of tiny bubbles of a few nanometres diameter with a small tendency to be more numerous on grain boundaries than within individual grains [5)]. The presence of helium on grain boundaries increases the risk of intergranular embrittlement.

Thick section core support components will also be subject to significant γ-heating so that internal metal temperatures may be markedly hotter than the surfaces exposed to the aqueous coolant. In PWRs, such high temperatures induced by γ-heating in combination with neutron doses greater than ~20 dpa also gives rise to a risk of void swelling occurring [e.g., 6)]. Although void swelling has not exceeded a few tenths of a percent in any samples of PWR irradiated austenitic stainless steel examined so far, there is strong evidence that void swelling is characterized by long incubation doses followed by relatively rapid void growth kinetics.

Core structural materials are, under certain circumstances, susceptible to environmentally assisted cracking (EAC), specifically to IASCC, in which neutron and/or γ irradiation contribute to the initiation and propagation of cracks. The neutron doses above which IASCC has been observed in austenitic stainless steels in BWRs on normal water chemistry (NWC) and in PWRs are respectively ~0.7 and ~3 dpa [e.g., 7)]. The difference in these apparent IASCC threshold doses is attributed to the difference in corrosion potential established in the oxygenated NWC of BWRs and the hydrogenated primary water chemistry of PWRs (and BWRs on hydrogen water chemistry, HWC). There are strong theoretical grounds and supporting observations that show that the embrittlement due to neutron irradiation discussed in the previous paragraph contributes directly to an increase in susceptibility to EAC mechanisms [e.g., 8)]. Radiation-induced segregation (RIS), especially at grain boundaries, must also be considered in assessing EAC susceptibility of austenitic alloys in LWRs. All these irradiation-induced phenomena are quite likely to extend progressively to near-core components with increasing irradiation dose associated with plant life extension.

Irradiation embrittlement and IASCC of core support structures and internals are covered in, respectively, sub-tasks 3.1 and 3.2 of the SMILE project [1), 10)]. The objectives of the former sub-task are to characterize the metallurgical and mechanical properties, including fracture toughness, of irradiated materials harvested from the internals of Swedish BWRs and a PWR retired after 40+ years of service. Particular attention will be paid to the maximum irradiation doses attained in the materials available, while the effects of void swelling, if any, will be evaluated on material taken from locations where the combination of dose and temperature in the retired PWR resulted in the highest probability for this phenomenon to occur. The objectives of sub-task 3.2 are to determine the combinations of metallurgical factors and residual stresses induced by high irradiation doses that enable IASCC growth, and to measure the effective fracture toughness in simulated LWR environments at the end of the crack growth rate (CGR) tests.

Reactor type	Component	Material	Material grades	Dose range (dpa)
BWR	Core shroud	Base and weld metal	304/308	~0.03 - ~0.5
	Core spray	Base and weld metal	316L, Alloy 82, XM-19	~0.01 – ~0.5
	Lower core plate	Base metal and weld	CF8, 304, X-750	<0.001 - ~0.3
	Riser pipe ECCS	Base metal and weld	316L	~0.5 - ~1
	Thermal shield	Base metal	316L	<0.01
PWR	Baffle plate	Base metal	304	<1 - >50
	Former plate	Base metal	304	<2 - >40
	Baffle to baffle bolt	Base metal	316	<1 - >40
	Baffle to former bolt	Base metal	316	<1 - >30
	Core barrel	Base and weld metal	304/308	<0.0001 - ~5
	Upper core support	Base metal	CF8	<0.01 - ~0.7
	Lower core support	Base metal	CF8	<0.01 - ~3.5
	Fuel alignment pin	Base metal	316	~1 - ~10
	Guide tube support pin	Base metal	316, X-750	~0.1 - ~0.5
	Flow mixing device	Base metal	CF8	<0.1

Table 1: **Library of materials from BWR and PWR RPV internals**

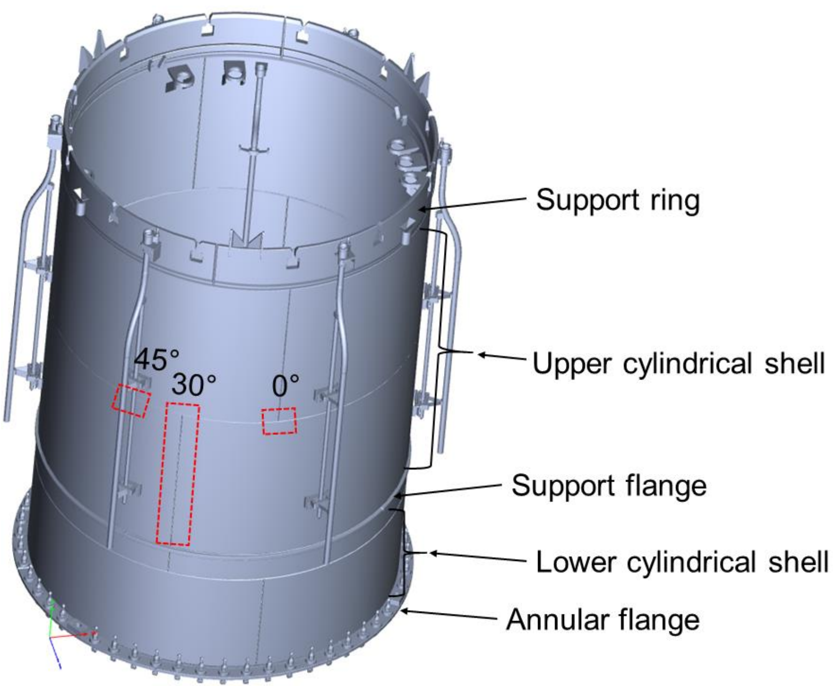
Library of Aged RPV Internal Materials

A library of materials harvested from the Swedish reactors Oskarshamn 1 and 2 (BWRs) and Ringhals 2 (PWR) is being established at Studsvik [10)]. These reactors were taken out of service after 40+ years of operation [1)]. Table 1 summarizes the materials from the RPV internals that are, or will be, included in the library. For each component, all relevant information available in the fabrication records (starting condition) is collected and the operational history (aging conditions) is described. In addition, the neutron dose, and where relevant gamma heating temperature, of the materials is being calculated, is described in more detail in [10]. Summaries of the information collected for the BWR core shroud and core spray system are provided as examples of the information to be contained within the library for all components in the next two sections.

BWR Core Shroud

Pieces of material have been harvested from the core shroud of Oskarshamn 2, a BWR designed by ASEA-ATOM (later ABB Atom, now Westinghouse Electric) and operated by OKG AB. The reactor had four external recirculation loops. The net electrical output was initially 570 MWe, but following power uprates the output was eventually raised to 638 MWe. Commercial operation started in 1974 and ceased in 2013 after it had been in service for ~31.7 effective full power years (EFPY).

The core shroud (now dismantled) was a large cylinder that surrounded the core and extended some distance above as well as below the core. Its main function was to direct the circulation of coolant through the core by separating the upward flow through the core from the downward flow in the down-comer outside the core shroud. The component was also part of the support structure for the core and the reactor pressure vessel (RPV) internals assembly. Figure 1 shows a schematic of the core shroud. The component was, from bottom to top, composed of the following major parts: an annular flange, a lower cylindrical shell, a support flange, an upper cylindrical shell, and a support ring. The support flange was located at about the level of the core inlet, and it was welded to the upper cylindrical shell, a part that surrounded the core. The upper cylindrical shell was made from four plates rolled into 180° seg-ments and joined by vertical welds forming two rings of equal size. The two rings of the upper cylindrical shell were joined by a horizontal weld located about level with the core mid-plane. The vertical welds in the upper cylindrical shell were located at 0 and 180° (upper ring), and 30 and 210° (lower ring).



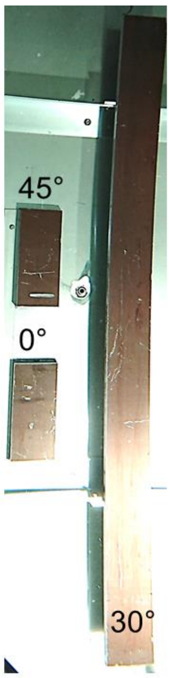


Figure 1: **Schematic of the core shroud from Oskarshamn 2 and the locations of the pieces of material harvested from the component (left), and a photo of the three pieces (right)**

As indicated in Figure 1, three rectangular shaped pieces of material were cut from the upper cylindrical shell of the core shroud, all containing a portion of the horizontal weld located at about the level of the core mid-plane. In addition, the pieces located at 0 and 30° contained vertical welds, including the junc-tions with the horizontal welds. The upper cylindrical shell was made of 25 mm thick plates joined by horizontal and vertical welds of the geometry shown in Figure 2.

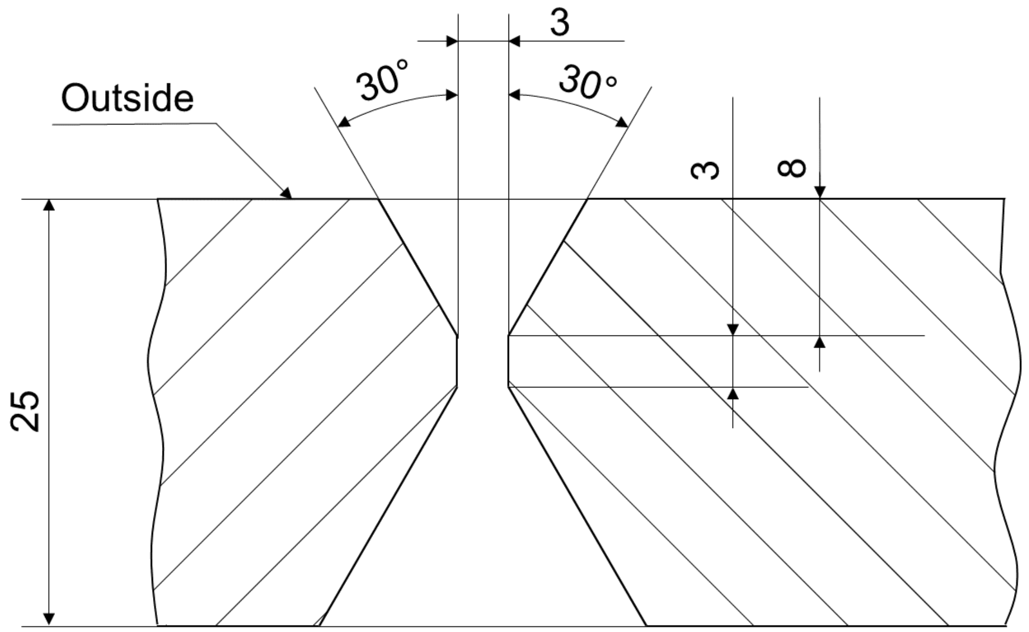


Figure 2: **Schematic of the horizontal and vertical weld geometries in the upper cylindrical shell of the Oskarshamn 2 core shroud**

The upper cylindrical shell was made of four solution annealed plates to Swedish standard SS2333-02, which is similar to Type 304 SS. All four plates were from the same heat and the chemical composition is shown in Table 2, whereas tensile properties as-received at room temperature and 300 °C are shown in Table 3. Corrosion testing in accordance with SIS 11 71 05 (replaced by SS-EN ISO 3651-2), ultrasonic testing, and visual inspections were performed on all plates. Ultrasonic testing and visual inspections included the plate edges prepared for welding. In addition, dye penetrant testing of the edges prepared for welding was performed. All plates fulfilled the requirements of these tests and inspections.

Material	Heat#	Analysis	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	Co	N
Plate	8-3007	Ladle	0.036	1.56	0.013	0.007	0.50	18.2	9.2	-	-	0.020	0.077
Weld	-	Check	0.029	1.63	0.016	0.011	0.53	19.6	9.5	0.03	0.06	0.022	-

Table 2: **Chemical composition (wt. %) of the plates of SS2333-02 and as-deposited weld metal used to manufacture the upper cylindrical shell**

Material	Room temperature			300 °C			
	YS MPa	UTS MPa	εt_{ot} %	YS MPa	UTS MPa	εt_{ot} %	RA %
Plate	279	579	59	140	441	41	67
Weld	448	611	37	284	415	22.5	61

Table 3: **Mechanical properties at room temperature and 300 °C of the plates of SS2333-02 (averages of three tests) and as-deposited weld metal used to manufacture the upper cylindrical shell**

The plates in the upper shell course were joined by manual metal arc welding (MMAW), also known as shielded metal arc welding (SMAW), using E19.9L-UC1 rutile coated electrodes of diameters 2.5, 3.25, 4 and 5 mm. To minimize distortion, weld beads alternated between the outside and inside. The welds were visually inspected and subjected to dye penetrant, ultrasonic and radiographic testing. When fabrication of the core shroud was completed, the component was stress relieved at 425 °C (±25 °C) for 24 hours. Dimensional inspections were performed after the stress relief treatment, first during test assembly of the internals at the workshop, and then after assembly at the reactor site. A few deviations were noted that either were accepted without action, or after minor adjustments. Thus, distortion caused by the stress relief treatment appears to have been small.

Information about the welds is available in reports from welding procedure qualification tests that were carried out and approved before welding of the core shroud took place. The plates used for the qualification tests were subjected to the same tests, examinations and treatments as the ones used for the upper cylindrical shell course, i.e., corrosion testing, ultrasonic testing, visual inspection, check of plate thickness, and dye penetrant testing of the edges prepared for welding. Following welding, all test welds were stress relieved at 400 °C (-0 °C, +50 °C) for 24 hours, after which they were subjected to radiographic testing and corrosion testing of specimens taken from three different locations in the thickness direction of the weld. All requirements of these tests and inspections performed before and after welding were fulfilled. The chemical composition of the weld (as-deposited) is shown in Table 2, while its mechanical properties are shown in Table 3. The ferrite content of the welds ranged from 9.0% to 10.6% with average and standard deviation values of 9.9% and 0.44%, respectively. Vickers hardness (HV10) measurements were made at three locations in the thickness direction of the weld; see Figure 3. These measurements showed that the hardness was quite uniform near the surfaces and that it was only slightly higher in the weld metal compared to the parent metal. A slightly higher variability in hardness of the weld along line 1 (L1) compared to line 3 (L3) is noted. At the centre of the weld and near the root pass, the hardness varied considerably, and the average was higher in the weld. However, peaks in hardness in the HAZ on either side of the weld were noted at all scanning locations.

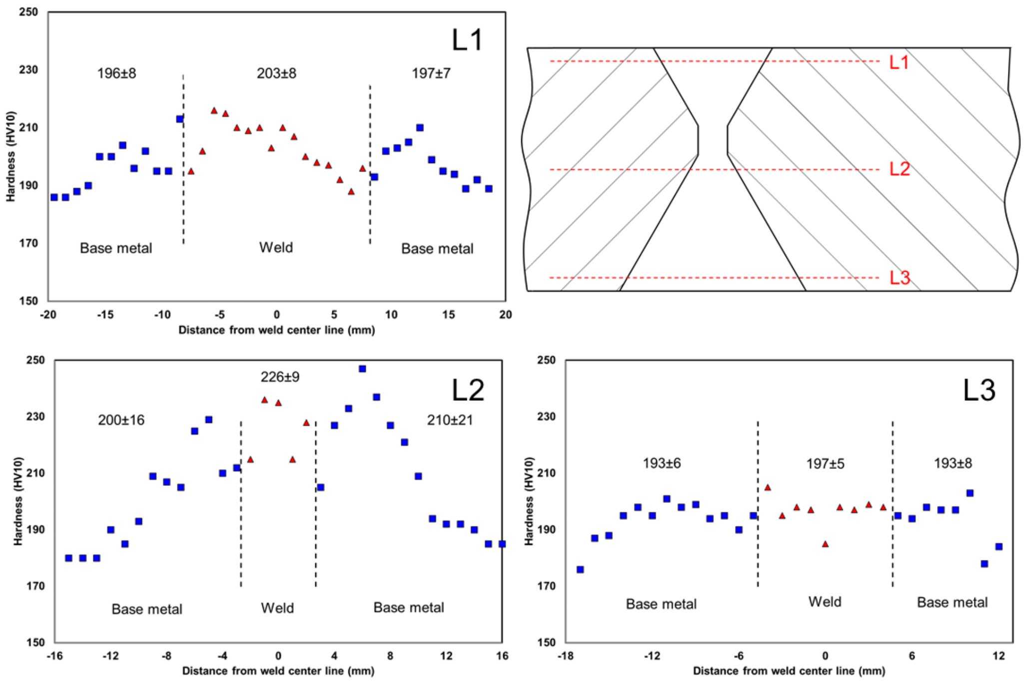


Figure 3: **Vickers hardness of the unirradiated welding procedure qualification test weld for the Oskarshamn 2 core shroud**

BWR Core Spray System

Pieces of material have been harvested from the core spray system of Oskarshamn 1, a BWR designed by ASEA-ATOM (later ABB Atom, now Westinghouse Electric) and operated by OKG AB. The net electrical output was initially 440 MWe, but following power uprates the output was eventually raised to 473 MWe. Commercial operation started in 1972 and the reactor was taken out of service in 2017, after it had been in operation for ~28.2 effective full power years (EFPY). Following completion in 1995 of an extensive inspection and repair project designated FENIX [11], OKG decided to replace several RPV internal components, including the core spray system which was part of the emergency core cooling system [12]. These components were replaced during the outage in 1998. The core spray system was in-service for ~11.9 EFPY until 2017.

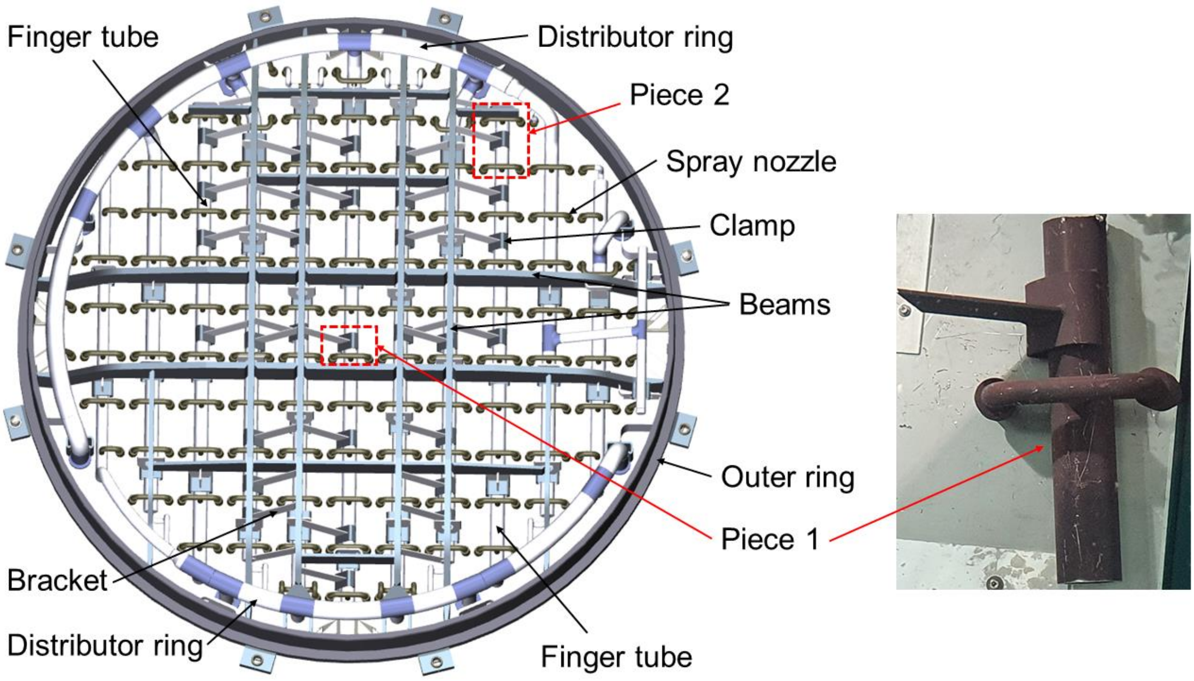


Figure 4: **Schematic of the Oskarshamn 1 core spray system including the locations of Pieces 1 and 2 (left) and a photo of Piece 1 (right).**

The replacement core spray system of Oskarshamn 1 was designed in two halves, where each half was connected to the external piping system by four individual riser pipes. The riser pipes were located around the inner surface of the core shroud and were connected to penetrations through the bottom dome of the reactor pressure vessel (RPV). Spring-loaded seals in the upper and the lower ends of the riser pipes connected with the core spray system and the RPV penetrations. Each half of the core spray system can be divided into a distributor ring, and straight finger tubes extending from the distributor half-rings; see Figure 4. Several spray nozzles were attached to each finger tube. The distributor half-rings and finger tubes were attached by means of brackets and clamps to a support structure consisting of an outer ring and a system of beams. All tubing and the support structure of the core spray system were made of Type 316L stainless steel, whereas brackets and clamps were made of XM-19, a nitrogen strengthened austenitic stainless steel also known as Nitronic 50. The brackets and clamps were manually welded to the tubing or support structure with Alloy 82 as filler metal.

As indicated in Figure 4, two pieces were cut from the core spray system. Piece 1 was cut from a central core position, whereas Piece 2 was cut from a peripheral core position. Cutting was made with a saw when the component was segmented for final storage. A photograph of Piece 1 is shown in Figure 4.

Hot rolled and solution annealed plates of XM-19 from two different ingots of the same heat were used for the clamps and brackets. The chemical compositions of the plates are shown in Table 4, while the mechanical properties at room temperature and 300 °C are presented in Table 5. Rockwell B hardness values, reported on the heat certificate, were 95 and 90 for Heats 67536-2F and 67536-1E, respectively. No intergranular fissures were observed following the intergranular corrosion test performed in accordance with ASTM A262, in which the samples were exposed to boiling copper/copper sulphate solution for 24 and 72 hours and then bent 180°. Average grain size measurements according to ASTM E112 were made on samples oriented longitudinal and transverse to the rolling direction. For both orientations the grain sizes were #6 (45 µm) for Heat 67536-2F and #4 (90 µm) for Heat 67536-1E.

Several heats of material and product forms of Type 316L were used for the fabrication of the core spray system; see Table 4 for information regarding product form and composition, and Table 5 for mechanical properties at room temperature and 300 °C. The materials were delivered in the solution annealed condition. All heats of Type 316L were tested for resistance to intergranular cracking according to ASTM A262, Practice E. No intergranular cracks were observed, and all heats of material passed this test. In addition, the plate and round bars were tested by UT, while the pipes were pressure tested at 8.0 MPa. The results of both tests were satisfactory.

Heat#	Material	Product form	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	Nb	Ti	Al	V	Co	N	B
67536	XM-19	Plate	.039	4.95	.023	.005	.46	21.11	12.19	2.14	.41	.15	.008	.009	.16	.16	.26	.0015
67536-2F	XM-19	Plate	.051	5.05	.026	.018	.47	20.94	12.32	2.12	-	.15	-	-	.15	.15	.26	-
67536-1E	XM-19	Plate	.051	5.04	.025	.018	.48	20.95	11.90	2.09	-	.15	-	-	.15	.16	.26	-
A3078	316L	Plate	.011	1.60	.015	.007	.58	16.60	10.38	2.08	-	-	-	-	-	.03	.07	-
437218	316L	Pipe	.014	1.59	.013	.001	.35	17.4	13.6	2.57	-	-	-	-	-	.03	-	-
438209	316L	Pipe	.012	1.65	.014	.001	.29	17.3	13.6	2.54	-	-	-	-	-	.03	-	-
33B8978	316L	Round bar	.020	1.53	.010	.010	.29	16.80	12.80	2.20	.05	-	.01	-	.01	.03	.02	-
33B8638	316L	Round bar	.016	1.53	.019	.005	.26	17.00	11.94	2.22	.04	.04	.004	.08	.03	.01	.03	-
033392[1)]	Alloy 82	Weld	.008	3.12	.005	.002	.20	20.30	72.85	-	.02	2.64	.32	-	-	.01	-	-
043567[1)]	Alloy 82	Weld	.007	3.09	.004	.003	.09	20.40	72.80	-	.02	2.66	.26	-	-	.01	-	-
043567[2)]	Alloy 82	Weld	.020	3.35	.013	.002	.14	20.3	64.8	.32	-	-	-	-	-	-	.047	-
503769[1)]	316L	Weld	.015	1,74	.019	.010	.50	18.65	11.15	2.58	-	-	-	-	-	.07	-	-
503713[1)]	316L	Weld	.010	1.83	.019	.004	.91	18.15	11.70	2.64	-	-	-	-	-	.06	-	-
[2)]	316L	Weld	.010	1.72	.016	.005	.61	17.67	12.6	2.56	-	-	-	-	-	.04	-	-

1) Weld electrode
2) As-deposited composition

Table 4: **Chemical composition (wt. %) of the materials used for Pieces 1 and 2 of the Oskarshamn 1 core spray system**

uMaterial	Heat#	Room temperature				300 °C			
		YS MPa	UTS MPa	εt_{ot} %	RA %	YS MPa	UTS MPa	εt_{ot} %	RA %
XM-19	67536-2F	440	805	47.0	69.0	278±7	653±9	42±1	61±2
XM-19	67536-1E	428±3	747±2	55±4	41±1	243±9	609±15	38±1	-
316L	A3078	287	590	62	-	159	-	-	-
316L	437218	273±5	558±10	48	-	168	461	41	-
316L	438209	275±6	612±5	50	-	190	477	36	-
316L	33B8978	284	542	54	74	196	400	34	68
316L	33B8638	278	532	58	76	192	392	38	72
Alloy 82	043567	460	757	-	-	-	-	-	-

Table 5: **Mechanical properties at room temperature and 300 °C of the materials used to manufacture the pieces cut from the core spray system**

Two heats each of Alloy 82 (Böhler NIBAS 70/20-IG) and Type 316L (Böhler EAS 4 M-IG) weld metal were used for the welds present in the two pieces cut from the core spray system. Compositional information about the weld materials is provided in Table 4. Welding procedure qualification tests were carried out for all types of welds in the component, and these tests provide information on the as-deposited composition of the weld metals; see Table 4. The welding procedure qualification tests also covered tensile (DIN 50120), bend (DIN 50121), and impact testing (DIN 50115) of specimens that were oriented transverse to the welding direction, as well as microstructural examinations of the weld joints. Tensile data for Alloy 82 Heat 043567 are shown in Table 5, while the ultimate tensile strength (room temperature) for the Type 316L welds ranged from 570 to 606 MPa. Impact energies for the Alloy 82 weld (Heat 043567) and HAZ were, respectively, 143±8 and 125±24 J. Corresponding values for a Type 316L weld and HAZ were, respectively, 92±5 and 90±1 J. Following welding, the test welds were subjected to dye penetrant and radiographic testing. All tests fulfilled the requirements.

The welds between the various tube sections constituting the nozzle tubing were automated, whereas all other welds in the pieces cut from the core spray system were manual. All welding followed welding procedure specifications specific for each type of joint design and combination of materials. An interpass temperature of ≤100 °C was specified for all welds, and the weld process was in all cases No. 141 according to EN ISO 4063 (GTAW, gas tungsten arc welding). Following welding, all welds were subjected to visual inspection, dimensional checks, dye penetrant testing, and in some cases also radiographic testing.

Examinations and Testing

The planned examinations and testing in sub-tasks 3.1 and 3.2 of the SMILE project are summarised in Table 6, while the objectives of these activities are outlined in the text of this section.

Component	Material	Dose	γ	NDT	Residual stress	Tensile	FT	LOM/ SEM	H/He	Density	CGR
Core shroud	308 Weld metal	Max.	✓	✓	-	✓	✓	✓	-	-	-
	308 Weld metal	Min.	✓	✓	-	✓	✓	✓	-	-	-
Core spray	XM-19	Max.	✓	-	-	✓	✓	✓	-	-	✓
Baffle plate	304 Plate	Max.	✓	✓	-	✓	✓	✓	✓	✓	✓
	304 Plate	<Max.	✓	✓	-	✓	✓	✓	✓	-	✓
	304 Plate	<Max.	✓	-	-	✓	✓	✓	✓	✓	-
Core barrel	Plate	Max.	✓	✓	✓	✓	✓	✓	-	-	-
	Weld metal	Max.	✓	✓	✓	✓	✓	✓	-	-	✓
	Weld metal	<Max.	✓	✓	-	✓	✓	✓	-	-	-
	Weld metal	Min.	✓	✓	✓	✓	✓	✓	-	-	-

Table 6: **Planned examinations and testing in sub-tasks 3.1 and 3.2 of the SMILE project**

Non-destructive testing (NDT) of some of the harvested pieces of material is being conducted at Studsvik's pool facility with the purpose of documenting any prior undetected manufacturing or service-induced defects, as well as aiding in the choice of positions for sampling/specimens for subsequent testing. The examinations comprise visual inspection, dye penetrant, replica moulding and ultrasonic testing depending on the component. For some pieces of material containing welds, residual stress measurements (hole drilling technique) are planned to be conducted to determine the stresses across the region of interest both laterally and in the depth direction. Gamma spectroscopy measurements of the materials are being performed, and the information will be used to benchmark the neutron dose calculations.

All materials included in the test matrix are subjected to microstructural examination in a light optical microscope, including micro hardness measurements, and scanning electron microscope (SEM), covering various thickness locations. These examinations are performed to characterize defects detected by NDT, and to document basic microstructures and weld geometries. Tensile tests are being conducted to determine the mechanical properties of the materials. Most tests are performed at 288 or 320 °C, but some testing will be performed at lower temperatures to determine the temperature at which the fracture mode of highly irradiated stainless steels transitions from brittle to ductile [9]. The fracture toughness (FT) at 288 or 320 °C of the materials is being determined by testing according to ASTM E1820, and in some specimens, the effect of hydrogen content will be evaluated by baking out hydrogen from deep traps under vacuum. Tensile and FT testing will for most materials cover multiple doses so that the dose dependency on the property can be assessed.

Helium and hydrogen analyses and density measurements will be made on highly irradiated baffle plate (from the retired PWR). This information together with transmission electron microscopy (TEM) studies of selected materials will be used to assess the presence of swelling, if any. Atom probe tomography (APT) will also be employed to characterize the effects of irradiation on selected materials and doses.

Crack growth rate (CGR) tests will be performed to determine the IASCC susceptibility of irradiated XM-19 from a BWR, as well as highly irradiated baffle plate and core barrel weld metal from a PWR. Regarding the baffle plate, testing at two different doses is planned, which will enable determination of the dose effect on the CGR. Testing of the BWR material will be conducted in NWC and HWC with on-the-fly changes between the two environments to quantify the mitigative effect of HWC, if any. Testing of the PWR materials is performed in primary water with on-the-fly changes in temperature and testing at different K levels, and the objectives are to determine the K and temperature dependencies on the CGR. All CGR tests will end, where possible, with a fracture toughness test in either BWR or PWR environment.

A summary of available results from the examination and testing of the BWR core shroud materials is provided below as an example of the eventual results to be developed for all components.

<u>Results of Examinations and Testing of the Oskarshamn 2 core shroud materials</u>
The three pieces cut from the core shroud were subjected to visual inspection, phased array ultrasonic testing (UT), time of flight diffraction (TOFD) testing, and eddy current testing (ET) in the pool facility at Studsvik. To the extent possible, eddy current and ultrasonic testing followed a procedure qualified for inspection of the intact Oskarshamn 2 core shroud at the reactor site. Deviations from the qualified procedure were the cut edges located relatively close to the weld that produced undesired signals, the longer manipulator arm holding the probes, and the manipulator arm only holding one probe at a time. In addition, ET scans were done on the inside as well as the outside surfaces of the pieces, while the qualified procedure only calls for inspection of one surface. Inspection by TOFD was done according to a best effort approach. No defects or indications of reportable size were observed in any of the pieces. However, when the detection threshold was lowered, one small (18 mm long and <0.5 mm) non-crack-like near-surface, or surface breaking, defect was noted by ET and TOFD (not UT) in the base metal of the piece cut at 0°. The defect was located rather close to the junction between the horizontal and vertical welds; see red arrows in Figure 5. Note that the defect was too small to have been considered for further analysis had it been detected during an in-service inspection campaign. Figure 5 shows the three pieces cut from the core shroud set up for inspection in the pool facility at Studsvik, and the ET and TOFD responses of the piece with the small defect. A metallographic sample will be prepared and examined to determine the nature of the defect.

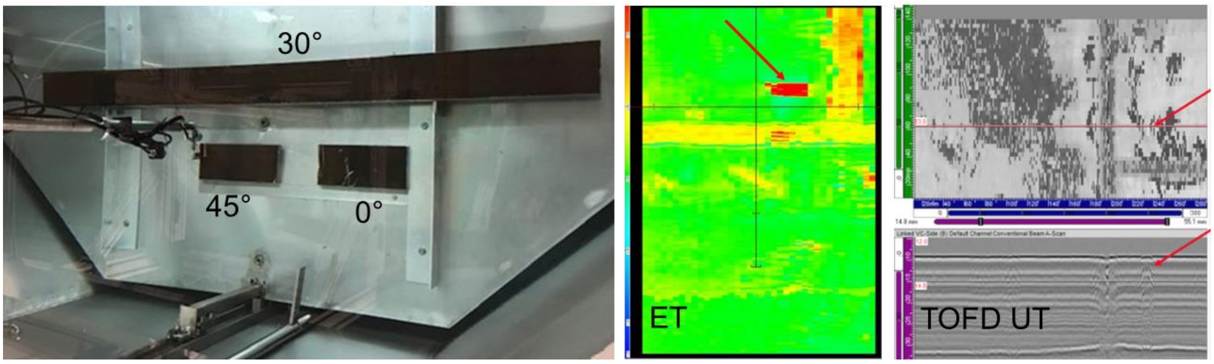


Figure 5: **Photograph of the three pieces cut from the core shroud, the manipulator arm holding the probes, and the inspection camera in the pool at Studsvik (left), ET (centre) and TOFD UT (right) responses of the piece located at 0°. The red arrows indicate the defect detected by ET and TOFD UT.**

Following coarse cutting under water at the pool facility, smaller pieces from the 0° and 30° azimuths have been transferred to a hot cell laboratory for machining of specimens, examinations, and testing. Compact tension (CT) and tensile specimens, as well as samples for metallographic examinations and gamma spectroscopy have been prepared; see Figure 6. As shown in this figure, 10 mm thick (B) and 20 mm wide (W) CT specimens with side grooves having depths that are 10% of thickness on each side have been fabricated such that cracking should extend in the weld metal. These specimens will be tested for fracture toughness at 300 °C in the T-L and T-S orientations relative to the welding direction.

Tensile specimens with a total length of 20 mm will be machined from the weld, whereas the metallographic samples cover the complete weld and base metal on either side of it.

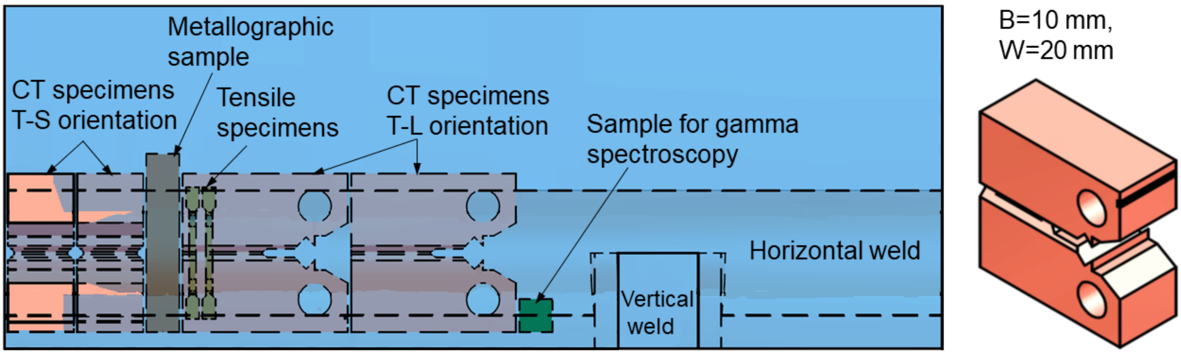


Figure 6: **Schematic showing the samples and specimens machined from the piece located at 30° and the CT specimen to be used for fracture toughness testing. The horizontal dashed lines indicate the weld fusion lines**

The metallographic samples of the horizontal core shroud weld at 0 and 30° were embedded in epoxy resin. Following grinding and polishing, Vickers micro hardness (HV 50 gf) testing was carried out using an Anton Paar MHT 10 hardness instrument integrated in a Leica Mef 4 light optical microscope installed in a hot cell. The hardness measurements were made along three lines at different locations in the thickness direction of the plate, consistent with the hardness measurements made on the test weld prepared for the welding procedure qualification test. Following the micro hardness measurements, the sample was ground and polished again and then electrolytically etched in 10% oxalic acid.

Micrographs of the sample located at 30° in the etched condition are presented in Figure 7. This figure covers the microstructure of the weld metal and the heat affected zone (HAZ). Figure 7 shows that the microstructures of the plate and weld were largely uniform and normal for stainless steel weld joints. The microstructure of the sample located at 0° was very similar, indicating the quality of the weld was uniform. Data from the micro hardness measurements are presented in Figure 8 with neutron dose estimates given in the figure caption. This figure shows that the hardness profiles were similar to the ones measured on the unirradiated welding procedure qualification test weld, i.e., the hardness was relatively uniform near the plate surfaces with a slightly higher hardness in the weld metal compared to the parent metal. Like the unirradiated test weld, the hardness at the centre of the weld and near the root pass varied considerably. The average hardness was higher in the weld, but peaks in hardness in the HAZ on either side of the weld were noted. A higher hardness was noted in the sample located at 30°, consistent with the higher neutron dose at this azimuth.

Summary

The approach and scope of work to study irradiation embrittlement and IASCC susceptibility of core support structures and internals in the SMILE (Studsvik Material Integrity Life Extension) project are described in the present paper. Information from the fabrication records regarding the Oskarshamn 2 core shroud and the Oskarshamn 1 core spray system, an outline of the planned testing, as well as initial results from the examination and testing of materials from the core shroud, are presented in this paper.

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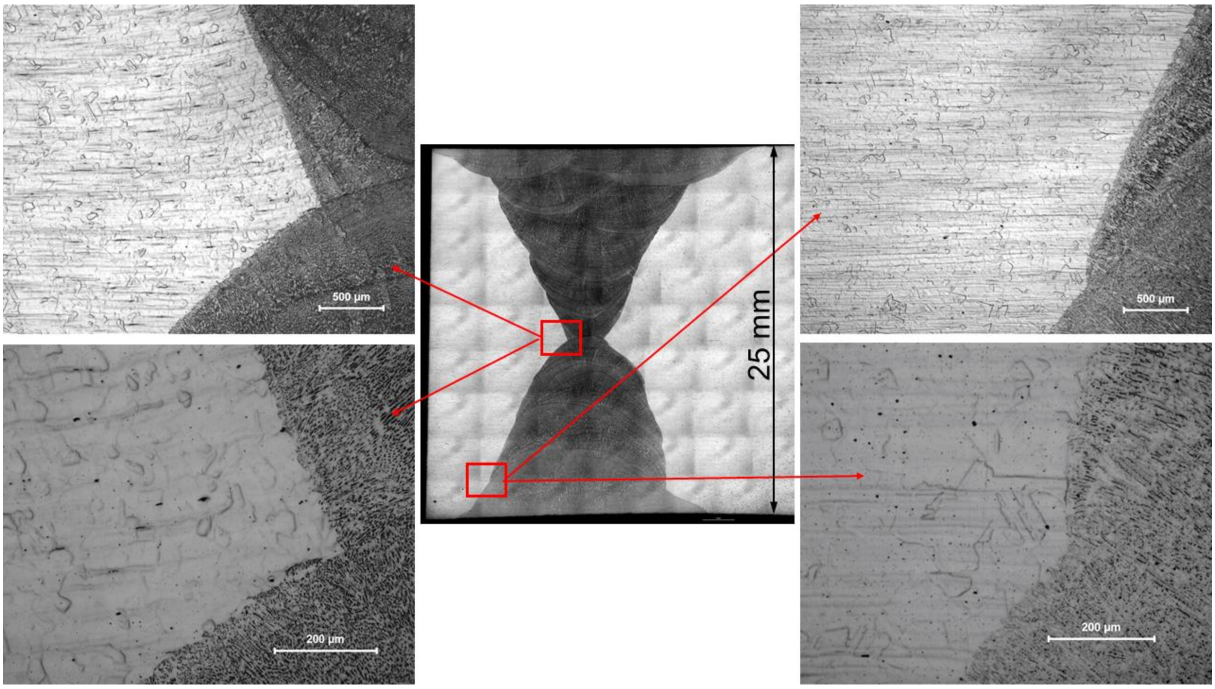

Figure 7: **Microstructure of the horizontal core shroud weld in the piece located at 30°.**

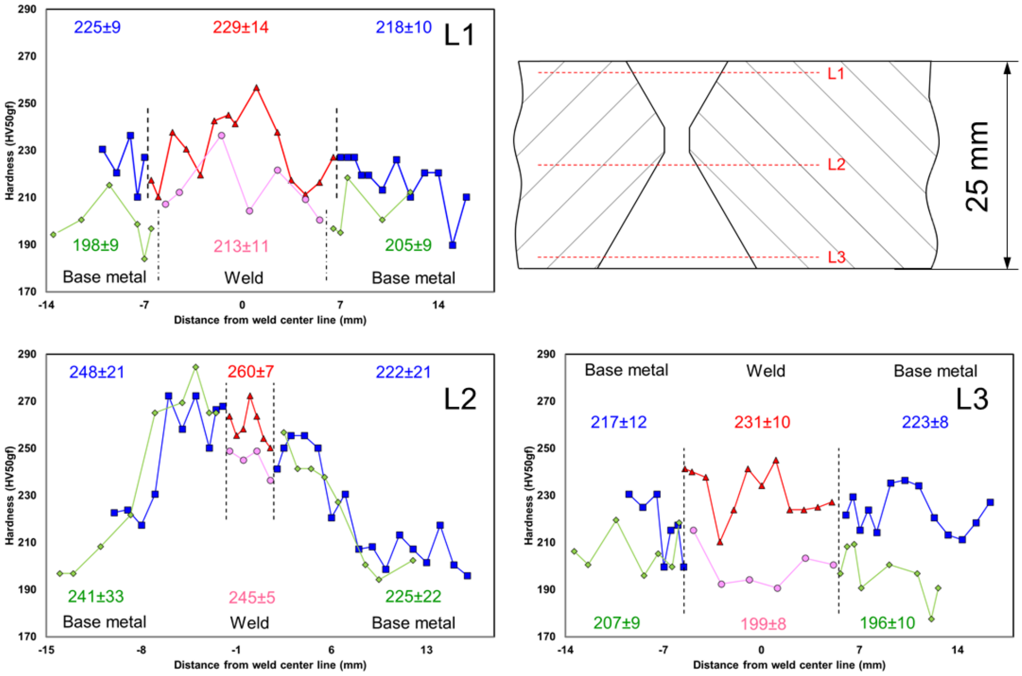

Figure 8: **Vickers micro hardness (50 gf) of the horizontal core shroud weld at low (0°, ~0.05 dpa, green and pink data) and high dose (30°, ~0.5 dpa, blue and red data).**

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