SAFETY REVIEW STATUS OF MAJOR STRUCTURAL MATERIALS FOR SMART

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ABSTRACT

Recently the full design of SMART such as small and moderate size reactor was completed and its design reports are under reviewing by domestic regulatory authority. Major structural materials of SMART were selected based on the operating experience of existing commercial reactors and to produce data for the verification of design suitability some tests for materials changed environmentally were conducted in HANARO reactor. In this report, reactor pressure vessel materials, reactor vessel internals, steam generator materials, and so on as structural materials of the primary system are summarized and materials reliabilities in aging aspect are evaluated and finally additional research to be necessitated are introduced. Also the irradiation embrittlement characteristics of Alloy 690 materials for steam generator tubes tested under neutron irradiated environment are reported.

INTRODUCTION

SMART (System-integrated Modular Advanced ReacTor) is a small nuclear reactor for electricity generation and seawater desalination. 330 MWt as the standard thermal power is suitable for the countries with limited electric grid size or limited financial resources and suitable for a city with 100,000 people. SMART is an integrated pressurized water reactor of which reactor vessel contains major primary components, such as core, steam generators, reactor coolant pumps, and a pressurizer. The standard design of SMART was completed at the end of 2010, and it is expected that the Standard Design Approval (SDA) will be obtained at the end of 2011 from MEST (Ministry of Education, Science and Technology) of Korea.

Now, the design report of SMART is reviewing to acquire the Standard Design Approval (SDA) at the end of this year. Separated effect tests and small size thermo-hydraulic test are finished. Codes and design methodologies were verified by 2010. The review of safety has been started from early 2010. Supplementary test will be performed in 2011, if required during SDA audit. Test using Integral Test Loop (ITL) will be finished by 2012 so that the results of the tests will be applied to the design improvement.

The primary system of SMART consists of core, 8 helical coiled steam generators, 4 canned motor reactor coolant pumps, and a pressurizer. These major components are integrated in a single pressure vessel. The reactor coolant is circulated through the core and steam generators by the reactor coolant pumps. The heat generated in the core transfers to the secondary system through steam generators. Fig. 1 shows the reactor vessel assembly and its flow patterns.

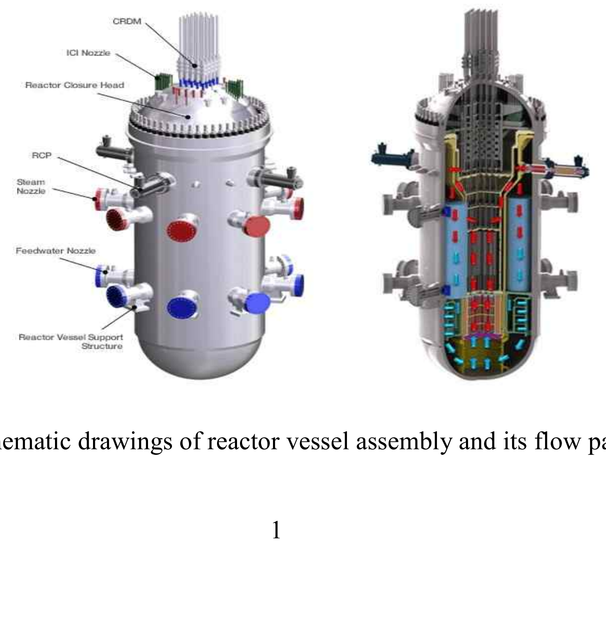


Fig. 1: Schematic drawings of reactor vessel assembly and its flow patterns

The materials selected for the reactor pressure vessel, reactor vessel internals, steam generator materials, and reactor coolant pumps as major structural materials of the primary system are described in this reports and the evaluation results of materials reliabilities in aging aspect based on the operational experiences of commercial nuclear power plant are discussed. Also the mechanical properties of Alloy 690 materials for steam generator tubes tested under neutron irradiated environment are introduced.

MATERIALS SELECTION FOR SMART PRIMARY SYSTEM

Reactor Pressure Vessel Assembly

Reactor pressure vessel assembly is consisted of reactor vessel, stud bolt assembly, and flow distribution cylinder. It is the protective container to prevent the outside release of radioactive materials and act as the role of supporting all kind of I&C components to protect from various operational load occurred during the normal operation and the upset operation conditions, and the faulted condition. The materials of reactor pressure vessel is the low alloy steel of MDF A508 Grade 3 Class 1 as shown in Table 1 and the inside of reactor pressure vessel is cladded with austenitic stainless steel materials to prevent the corrosion due to the contact with the reactor coolant.[1]

Table 1: Materials selected for reactor pressure vessel assembly

Assemblies	Components	Materials
Reactor Pressure Vessel	Reactor Pressure Vessel	MDF A508 Gr, 3 Cl 1
	Cladding	Austenitic Stainless Steel
Stud Bolt Assembly	Stud bolt, Nut, Washer	MDF 540 Gr, B24 Cl 3
Flow Distribution Cylinder	Cylinder, Stiffener, Inner Ring, Flange, Snubber Lug	Alloy 690
	Shim Block	MDF A479 521800

The cods and standards of materials for SMART are based on the KEPIC (Korea Electric Power Industry Code).

Reactor Closure Head Assembly

Reactor closure head assembly is made up of reactor head, CRDM nozzle installed through the reactor head, pressurizer heater nozzle, in-core instrument nozzle, and various instrument nozzles. Reactor head is combined to reactor pressure vessel with stud bolts and formed the pressure boundary of reactor pressure vessel assembly. The materials of reactor head is the low alloy steel of MDF A508 Grade 3 Class 1 and the inside of reactor head is also cladded with austenitic stainless steel materials. All kind of nozzles are made of Alloy 690 materials.

Reactor Internals

Reactor internals is a structure located in reactor pressure vessel. It supports the reactor cores, guides the control rods, provides the flow path of reactor coolant and cools the cores and the control rod assembly, and supports the steam generator cassettes structurally. Additionally it reduces the flow induced vibration by the reactor coolant, mitigates the radiation and the heat released outward the reactor pressure vessel, and protects the cores and the control rod assembly against the external load. Reactor internals are composed of ICI guide support structures, core support barrel assembly, flow mixing header assembly, and upper guide structure assembly. They are installed to and withdrawn from the reactor pressure vessel as the type of individual assembly. Most reactor internals is made of type 304 austenitic stainless steel which has the high strength and the excellent corrosion resistance.

Reactor Pressure Vessel Assembly Support Structure and Reactor Pressure Vessel Upper Structure Assembly

Reactor pressure vessel assembly support structure is the structure by which reactor pressure vessel assembly is fixed to supports of containment building. It is designed to maintain the structural integrity against the static and dynamic loads occurred due to the accidents such as the earthquake and the pipe rupture. Reactor pressure vessel upper structure assembly has the role that supports and fixes a number of instrument cables and power cables connected to intermittent panels not to sag in intermediate part. So it is designed to be arranged systematically to perform the maintenance and the repair work easily. The materials of studs and nuts used in assembling of two above components are SA-638 Gr. 660 and MDF A194 or MDF A193 individually.

Steam Generator

Steam generators are consisted of twelve steam generator cassettes and installed in the plenum space between core support cylinder and reactor pressure vessel. The shape of each steam generator cassette is the cylinder with the steam header in the upper side and the feed-water header in the lower side. It has the tube bundle with more over 250 tubes wound helically and this tube bundle is connected to the feed-water header and the steam header inside the cylinder. The reactor coolant induced through the upper part of steam generator flows downward outside of tubes and drains through the lower part of steam generator, and the feed-water induced to feed-water header in the lower part of steam generator cassette flows upward through inside of the tube and converts to the steam due to the heat of the reactor coolant and is finally released outside of the reactor through the steam header in the upper part of the steam generator cassette. The design characteristics of this steam generator are as follows;

(a) The flowing-through type of steam generator which its tubes were wound helically to have the maximum heat transfer surface in the limited space was selected.

(b) Alloy 690 as the tube materials was chosen with considerations of the corrosion resistance, heat transfer property, and strength and so on.

(c) That the diameter of inner cylinder of cassette is 530 mm with respect to the minimum diameter for the in-service inspection was determined

The materials of major components of steam generators are shown in Table 2.

Table 2: Materials selected for steam generators

Assemblies	Components	Materials
Steam Generator	SG Inner & Outer Cylinder	MDF A240 TP304
	Feedwater Header, Steam Header, SG Tube	Alloy 690
	Tube Support Strap, Tube Support Arm, Tube Orifice, SG Cover, Thermal Insulating Cylinder	MDF A240 TP304

Reactor Coolant Pump

In the design concept of the reactor coolant pump, the flow capacity and the water peak pressure of pump, the start-up at room temperature, link with the reactor, inertial slow moving, material requirement are considered as the important design requirement. In SMART, canned motor type of 4 reactor coolant pumps are installed in the lateral of the upper part of the reactor pressure vessel. The predominant corrosion resistant material is used for the pressure boundary components as pump casing and flange. The material with the excellent electro-magnetic and mechanical properties is used in the rotating shaft and the material with the good erosion resistance in the high pressure and high temperature conditions is used for the bearing and the high strength material is used as the stud for the flange assembling. Most components and parts of reactor coolant pump are made of austenitic type 304 or 316 stainless steels except the bolts and nuts of which the material is MDF A540 Grade B24 Class 3.

MATERIALS EVALUATION FOR SMART PRIMARY SYSTEM

Irradiation Embrittlement

A low alloy steel (MDF A508 Grade 3 Class 1) selected as the materials of reactor pressure vessel and reactor closure head for SMART is nearly same to the material (MDF A508 Grade 2 Class 1) used as the reactor pressure vessel and reactor closure head of Kori unit 1 for 32 years in respect of the chemical composition. In addition, the heat treatment condition for the formation of mechanical properties is applied to two materials identically. So they have the individual same ranges of ultimate tensile strength, yield strength, ductility ratio, and area reduction ratio. The differences of chemical compositions are that the compositions of C, Cr, Mo in Grade 2 are 0.02%, ~0.2%, 0.1% more than those in Grade 3, besides the composition of Mn in Grade 3 is 0.5~0.7% more than that in Grade 2.

The evaluation of irradiation embrittlement for MDF A508 Grade 3 Class 1 as the SMART reactor pressure vessel material can be substitute with the evaluation results for the irradiation embrittlement of Kori unit 1 reactor pressure vessel steel having operating experiences for 32 years and the expectation results of irradiation embrittlement with consideration to 40 operating years in the future. As the evaluation results of ΔRT_{NDT}, RT_{PTS}, and USE for Kori unit 1 data, the margin of irradiation embrittlement for 40 operating years will be maintained well. Therefore, based on the operating experience of commercial nuclear power plant for 32 years and the expected evaluation results for 40 years, it is understood that MDF A508 Grade 3 Class 1 steel will have no problem in using as SMART reactor pressure vessel and reactor closure head material.

Most of the ICI support structure assembly, the upper & lower guide structure assembly, and flow mixing header assembly are fabricated by using MDF A182, A240, A312, and A479 Type 304 and 304LN materials. Reactor internals can be sensitive to the embrittlement and the reduction of ductility. Based on the operating experience of a domestic commercial nuclear power plant, in the irradiation embrittlement of reactor internals material, its reliability is like to be maintained though the operating period is 30 years (about 22.3 EFPY) and the eatimated neutron fluence is $5x10^{22}$ n/cm^2. And also the results of visual test performed for the general reactor internals of Kori unit 1 was good.

Alloy 690 which substituted to Ti Alloy having used as the original design material has no field and laboratory experience data. On the other hand, Alloy 690 will have considerable effects by the neutron irradiation if the environment of the steam generators and flow distribution cylinders located in SMART is considered and the data production for the basic verification of the mechanical property change is required.[2]

IASCC (Irradiation Assisted Stress Corrosion Cracking)

In PWR environment, the relationship of the IASCC (irradiation assisted stress corrosion cracking) occurrence (%) and the neutron fluence (Dose) for the cold worked TP 316 stainless steel and the solution annealed TP 304 and TP 347 stainless steels shows that the threshold fluences with temperatures are about 9 dpa at 290℃, about 4 dpa at 320℃, and about 3 dpa at 325℃. So we note that the IASCC of operating environment in nuclear power plant can occur at more than 4 dpa.

The calculated neutron fluence for the baffle former bolts of Kori unit 1 is $5x10^{22}$ n/cm^2 and their bolts should be controlled carefully because its value exceeds over the critical value. According to the operation experiences, the split pin of control rod guide tube broke at $5x10^{21}$ n/cm^2 of neutron fluences and the failure of the baffle former bolts occurred at $1x10^{22}$ n/cm^2 of neutron fluences.

There are two criteria in evaluating the IASCC for the reactor internals of Kori unit 1 generally. At first, in case of water chemistry, the occurrence probability of the IASCC is extremely low if the halogen and the sulphur concentrations are less than 0.05 ppm, respectively and the dissolved oxygen concentration is also less than 0.05 ppm. Such water chemistry values don't exceed over 0.05 ppm, but the neutron fluence is high, the IASCC can occur. Secondly, in PWR environment, the occurrence condition for the IASCC depends on the neutron fluence. As mentioned before, the threshold neutron fluence of PWR reactor internals is $5x10^{21}$ n/cm^2. As the results of the neutron fluence analyses, the occurrence probability of the IASCC for the other parts of the reactor internals except the inner wall of the core barrel will be extremely low nevertheless the 50 operating years pass. Accordingly, the inner wall of the core barrel, baffle and former, baffle former bolts are required to control the aging degradation including the enhanced inspection because the excess of the critical neutron fluence for the IASCC with the irradiation embrittlment at the point of 40 operating years.

Alloy 690 has few field and laboratory experience data for the IASCC. Alloy 690 will have also considerable effects by the neutron irradiation if the environment of the steam generator located in SMART is cosidered and the data production for the basic verification of the IASCC resistance change is required.[3]

MECHANICAL TESTS OF IRRADIATED ALLOY 690

The test background is to obtain the data for the material characteristics of irradiated Alloy 690 by installing of steam generators inside of the reactor pressure vessel in SMART.

The material of test specimens is Alloy 690 as the tube material of steam generator in SMART. Its chemical compositions are shown in Table 3 and its mechanical properties in Table 4. Their measured values were satisfied with the ASME code requirements.[4]

In this test, various conditions of the neutron fluences were chosen; $1\text{x}10^{23}$ n/m^2, $1\text{x}10^{24}$ n/m^2, $1\text{x}10^{22}$ n/m^2. According to the result of the internal research, three test conditions were determined based on the value of the neutron fluence ($1.1\text{x}10^{22}$ n/m^2) which is exposed during the lifetime of the steam generator in SMART. The specimens were irradiated in Hanaro research reactor in KAERI. After the irradiation, the capsule were taken to pieces and divided into classes in hot cell of IMEF (irradiated materials examination facility). Then the tensile test and the fracture toughness test were performed.

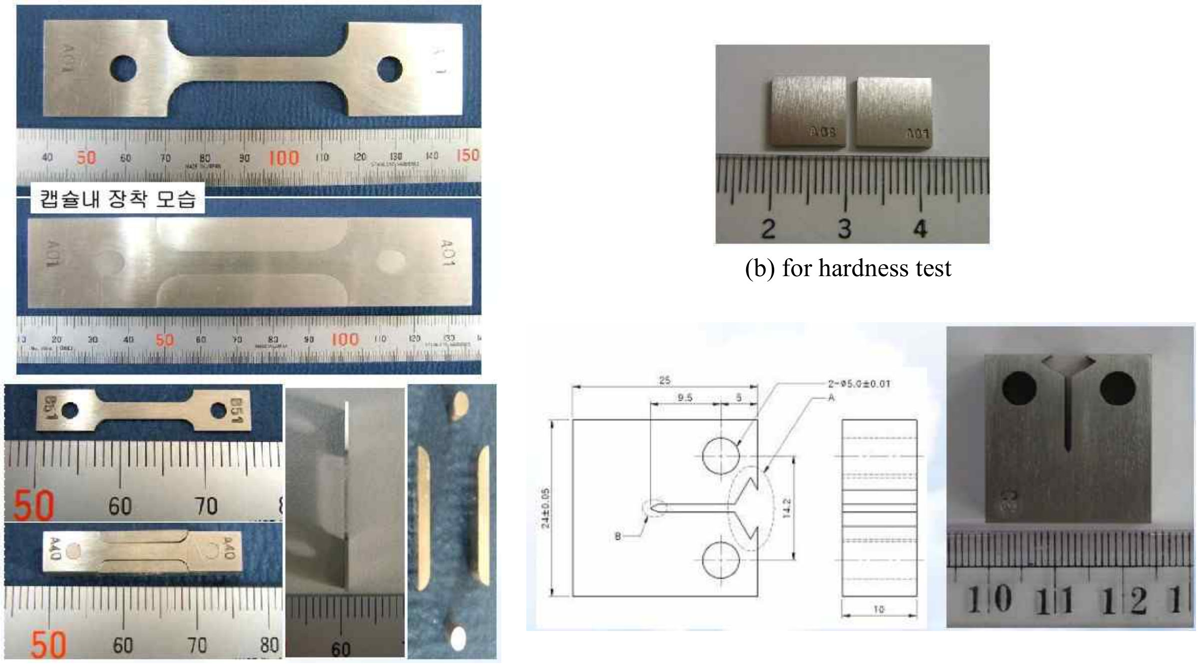
The mechanical test temperatures are the room temperature (20℃), the operating temperature (250℃), and the design temperature (360℃). The number of test specimen is the minimum two and the test methods are the tensile test, the fracture toughness test, the thermal prorety test, and the hardness test (at room temperature only). The various irradiated specimens are shown in Fig. 2. Before and after the tests, the status of irradiated specimens is shown in Table 5 and 6.

Table 3: Chemical compositions of Alloy 690 Heat A

No.	C	Ni	Cr	Fe	Si	Mn	S	Cu
Alloy 690 Heat A	0.02	60.70	28.97	9.13	0.28	0.21	<0.002	0.01

Table 4: Mechanical properties of Alloy 690 Heat A

No.	Yield Strength (MPa)	Tensile Strength (MPa)	Elongation (%)	Remarks
Alloy 690 Heat A	287	658	57	



(a) for tensile test

(b) for hardness test

(c) for fracture toughness test

Fig. 2: Various irradiated specimens for mechanical tests

Table 5: Status of 1st irradiated specimens before and after the tests

Specimen Type	Size(mm)	1st Capsule (09M-02K)		Status of Test			Preliminary Stock	Remarks
		Amount of Heat	Total Amount	Test Amount	Test Temp.	Total No. of Test		
0.4T CT	24x25x10	A(16),B(2),C(2)	20	A(9),B(1),C(1)	RT,250,360	11	A(7),B(1),C(1)	
Standard Tensile	108x25x2.5	A(14)	14	A(9)	RT,250,360	9	A(5)	
Small Tensile	26x5x0.5	A(20),B(10),C(10)	40	A(9),B(7),C(6)	RT,250,360	22	A(11),B(3),C(4)	B(1) was failed during the test
Hardness	10x10x2	A(10),B(9),C(9)	28	A(2),B(2),C(2)	RT	6	A(4),B(3),C(3)	
TEM	ø3x0.1	A(10),B(10),C(10)	30	-	-	-	30	Using the Hv specimen
Thermal Diffusivity	10x10x2	A(4),B(4),C(4)	12	A(4),B(4),C(4)	RT~360℃	12	-	
Total Amount of Specimen		132	132	-	-	60	72	

Table 6: Status of 2nd irradiated specimens before and after the tests

Specimen Type	Size(mm)	2nd Capsule (10M-01K)		Status of Test			Preliminary Stock	Remarks
		Amount of Heat	Total Amount	Test Amount	Test Temp.	Total No. of Test		
0.4T CT	24x25x10	A(9),B(1),C(1)	11	A(9),B(*),C(*)	RT,250,360	*	A(0),B(*),C(*)	
Standard Tensile	108x25x2.5	A(14)	14	A(9)	RT,250,360	*	A(1)	4 specimens were failed during the desembling
Small Tensile	26x5x0.5	A(20),B(10),C(10)	40	A(9),B(*),C(*)	RT,250,360	*	A(11),B(*),C(*)	
Hardness	10x10x2	A(5),B(6),C(6)	17	A(*),B(*),C(*)	RT	*	A(*),B(*),C(*)	Using the TDT partially
TEM	ø3x0.1	A(10),B(10),C(10)	30	-	-	-	**	3 Boxes
Thermal Diffusivity	10x10x2	A(16),B(12),C(12)	40	A(4),B(4),C(4)	RT~360℃	*	-	
Total Amount of Specimen		152	152	-	-	*	**	

Hardness Test

Before and after the irradiation, the hardness test for Alloy 690 was performed. As shown in Table 7, the micro-vickers hardness values for non-irradiated materials are not affected by the direction of the specimen sampling. The hardness values for non-irradiated materials were measured in the range of 158~184 according to the heat treatments. As the results of hardness test for 1st and 2nd irradiated specimens, the hardness values of 1st irradiated specimens increased about 30% more than those of non-irradiated specimens. The hardness values of 2nd irradiated specimens tested at room temperature are similar to those of 1st irradiated specimens.

Tensile Test

The tensile test for the non-irradiated and irradiated Alloy 690 was conducted. After the 1st irradiation, the yield strengths of specimens tested at room temperature and 250℃ increased average 25% compared with the non-irradiated ones. But the yield strengths of specimens tested at design temperature increased 10%. After the 2nd irradiation, the yield strengths of specimens tested at room temperature increased average 41% compared with the non-irradiated specimens and the yield strengths of specimens tested at design temperature increased 13%. When Alloy 690 material was irradiated at $1x10^{23}$ n/m^2 and $5x10^{24}$ n/m^2 of neutro fluence, the increasing rates of the yield strength with the temperature increase were different, but all yield strengths increased.

The change of tensile strength values with the neutron fluence didn't occur in all test conditions of the non-irradiation, 1st irradiation, and 2nd irradiation and the difference of its value is within 2% of error range. Table 8 shows the measuring results of the yield strength values for Alloy 690 as the steam generator tube materials before and after the irradiation. Fig. 3 shows the decrease trend of yield strength values for Alloy 690 with the temperature conditions before and after the irradiation and Fig. 4 shows the dependence of temperature conditions for the tensile strength values of Alloy 690.

Table 7: Hardness values of non-irradiated and irradiated Alloy 690 according to the heat treatments

Irradiation Status	Specimen No.	Specimen ID	Characteristics (Heat, Direction)	Test Force [g]	Hardness (Mean Value)	Standard Deviation	Remarks
Non-irradiation	1	SA-L1	Heat-A, Longitudinal	100	174.48	2.45	
	2	SB-L1	Heat-B, Longitudinal	100	168.54	3.74	
	3	SB-L2	Heat-B, Longitudinal	100	158.82	3.08	
	4	SC-L1	Heat-C, Longitudinal	100	173.79	3.44	
	5	SC-L2	Heat-C, Longitudinal	100	166.38	2.09	
	6	SA-R1	Heat-A, Radial	100	173.77	3.01	
	7	SB-R1	Heat-B, Radial	100	172.31	4.10	
	8	SB-R2	Heat-B, Radial	100	160.86	4.83	
	9	SC-R1	Heat-C, Radial	100	173.33	2.41	
	10	SC-R2	Heat-C, Radial	100	162.18	3.49	
1st Irradiation	1	A-04	Heat-A, Radial	100	230.60	5.03	
	2	A-05	Heat-A, Radial	100	212.50	2.52	
	3	B-03	Heat-B, Radial	100	225.00	3.95	
	4	B-04	Heat-B, Radial	100	248.50	1.79	
	5	C-03	Heat-C, Radial	100	227.80	1.34	
	6	C-04	Heat-C, Radial	100	242.20	5.30	
2nd Irradiation	1	A-15	Heat-A, Radial	100	213.20	1.44	
	2	A-16	Heat-A, Radial	100	213.8	2.53	
	3	B-14	Heat-B, Radial	100	215.1	1.87	
	4	B-15	Heat-B, Radial	100	235.1	2.70	
	5	C-15	Heat-C, Radial	100	253.4	4.18	
	6	C-16	Heat-C, Radial	100	221.1	5.87	

Table 8: Yield strength values of Alloy 690 before and after the irradiation

Irradiation Status	Test temperature (℃)	Heat	0.2% YS (MPa)	UTS (MPa)
Non -Irradiation	25	A	287	678
	250	A	227	584
	360	A	218	560
1st Irradiation ($1x10^{23}$ n/m^2)	25	A	355	682
	250	A	283	591
	360	A	239	556
2nd Irradiation ($1x10^{24}$ n/m^2)	25	A	405	698
	250	A	311	580
	360	A	247	547

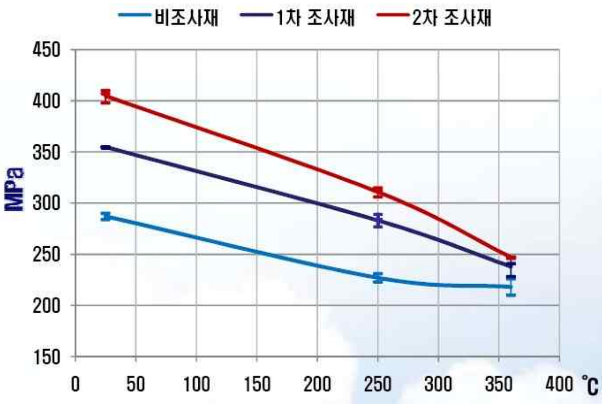


Fig. 3: Yield strength values for Alloy 690 with the temperature conditions before and after the irradiation

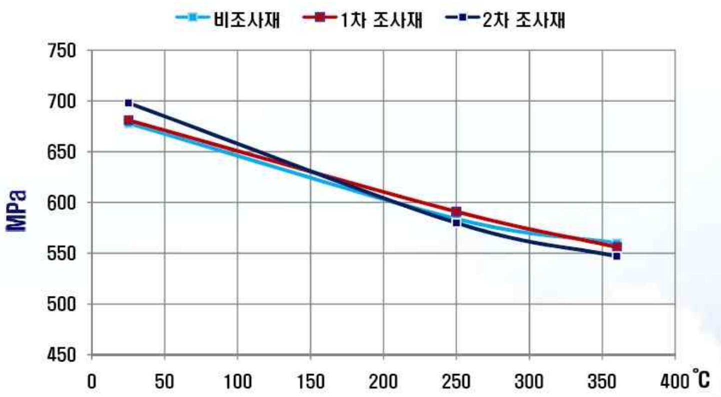


Fig. 4: Dependence of temperature conditions for the tensile strength values of Alloy 690

Fracture Toughness Test

The fracture toughness test for Alloy 690 was performed before and after the irradiation. Because the deviation of the fracture toughness values tested at room temperature was severe, it is hard for their values to compare each others. But the deviation of fracture toughness values for the irradiated specimens tested at room temperature is smaller than that for the non irradiated specimens. Therefore the probability that the fracture toughness values become lower is high. When the temperatures of the 1st irradiated specimens increase, its fracture toughness values decrease compared with those of non-irradiated specimens. Also the fracture toughness values of 2nd irradiated specimens decrease compared with those of non-irradiated specimens and those of 2nd irradiated specimens is similar to those of the 1st irradiated specimens. That is, the fracture toughness values tested at 250℃ decrease after the irradiation. After the 1st irradiation and the 2nd irradiation, their values decrease to 72% and 78% of J_Q values for the non-irradiated specimens, respectively. Table 9 shows the measuring results of the fracture toughness values for Alloy 690 as the steam generator tube materials before and after the irradiation. Fig. 5 shows the decrease trend of yield strength values for Alloy 690 with the temperature conditions before and after the irradiation and Fig. 6 shows the change of fracture toughness [J_Q] with the increase of the neutron fluence for Alloy 690 at 250℃.

Table 9: Fracture toughness values of Alloy 690 before and after the irradiation

Irradiation Status	Test Temperature (℃)	Heat	C_1 (kJ/m²)	C_2	$J_{0.5}$ (kJ/m²)	$J_{1.0}$ (kJ/m²)	J_Q (kJ/m²)
Non -Irradiation	25	A	774.3	0.51	546	775	608
	250	A	730.8	0.51	514	731	609
	360	A	693.4	0.52	486	694	571
1st Irradiation (1×10^{23} n/m²)	25	A	755.8	0.52	529	756	581
	250	A	620.7	0.60	411	621	441
	360	A	528.5	0.72	323	528	305
2nd Irradiation (1×10^{24} n/m²)	25	A	656.4	0.56	451	656	455
	250	A	637.0	0.56	432	637	473
	360	A	505.3	0.72	307	505	286

$J=C_1(\Delta a/k)^{C_2}$ (k=1.0mm)

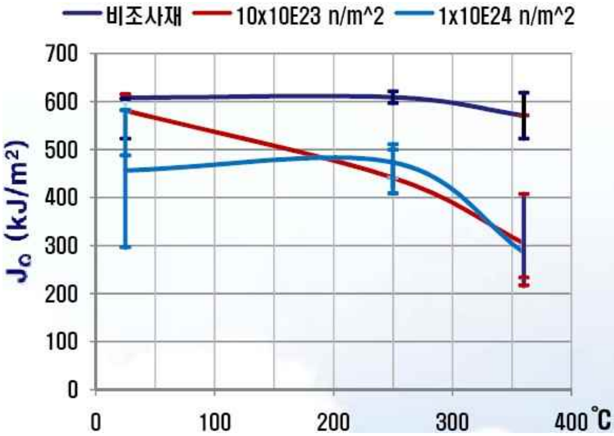


Fig. 5: Fracture toughness values for Alloy 690 with the temperature conditions before and after the irradiation

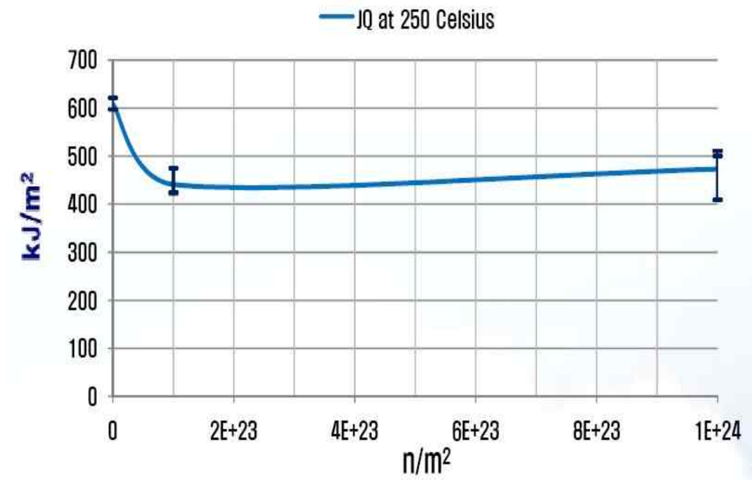


Fig. 6: Dependence of the neutron fluences for the fracture toughness values of Alloy 690

CONCLUSION

The materials selected for reactor pressure vessel assembly and support structures, reactor vessel internals, reactor closure head assembly, steam generator, and reactor coolant pump as structural materials for the SMART primary system were summarized and the reliability of the candidate materials in aging aspect were evaluated.

The changes of mechanical properties for the irradiated Alloy 690 material were observed. The summary of major test results is as follows;

(a) The yield strength values of Alloy 690 tested at 250℃, that is the operating temperature of SMART, increase after the irradiation and after the 1^{st} irradiation, its increasing rate was 25% compared with that of the non-irradiated specimens. Also after the 2^{nd} irradiation, its increasing rate was 37%.

(b) Before and after the irradiation, the difference of tensile strength values is within 2% of error range.

(c) The fracture toughness values tested at 250℃ decrease after the irradiation. After the 1^{st} irradiation and the 2^{nd} irradiation, their values decrease to 72% and 78% of J_Q values for the non-irradiated specimens, respectively.

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