中国核电工程有限公司
China Nuclear Power Engineering Co.,LTD.
中核集团 CNNC

The investigation of material selection and performance for Gas-cooled micro-reactor pressure vessel(RPV)

JIAO Shaoyang

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CNNC China National Nuclear Corporation

Content Outline

I. Background

II. Material selection analysis□

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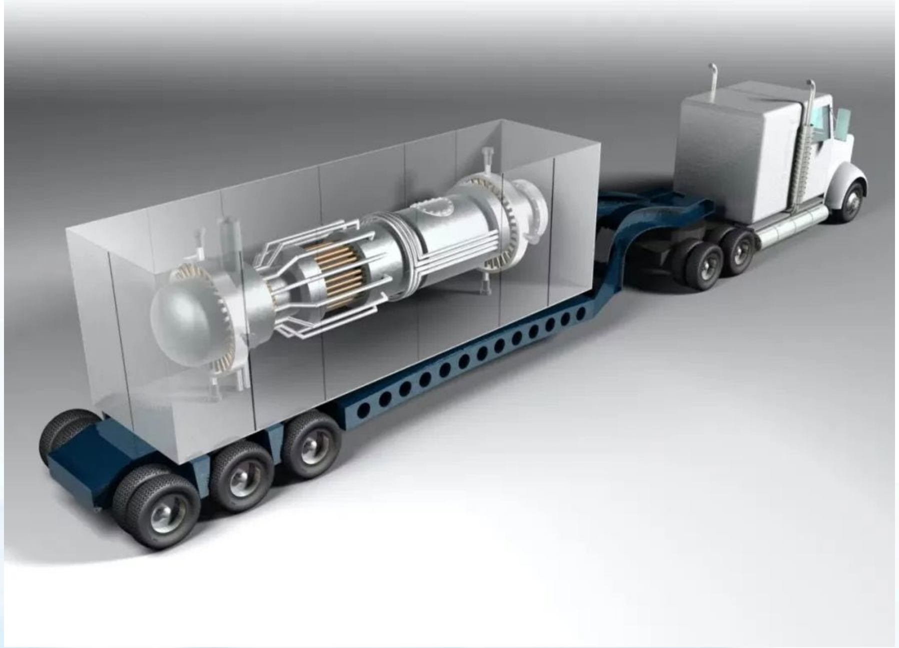
IV. Conclusion□

I. BACKGROUND-the advantage of GCMR

- High inherent safety
- miniaturization
- Low nuclear safety requirements
- High power generation rate
- Multiple application scenarios



The design input of RPV

- **Design temperature**: 550℃□
- **Design pressure**: 2MPa□
- **The accumulated fast neutron fluence at EOF**□ $\leq 1\times 10^{20} n/cm^2$;
- **Service environment**: helium;
- **Design code**: ASME III-D5□

RPV material selection principle

- **Specification compliance requirement**: use permitted material;
- **Service environment requirement:** corrosion/ irridiation, etc□
- **Maturity and accessibility**: best to have engineering experience;
- **Fabrication requirement:** convenient to process;
- **Light weight**: facilitate transporation□

II. Material selection analysis

------Material grade of different type RPV

Reactor type	Design temperature□ ℃□	Design pressure □ MPa□	Fast neutron fluence □ E□ 1Mev□	Material grade
AP1000	343	17	~9× 10^{19}□ 60a□	SA-508 Gr.3 Cl.1
M310	343	17	~7× 10^{19}□ 40a□	SA-508 Gr.3 Cl.1
HTR	350	8	2.42× 10^{18}□ 60a□	SA-508 Gr.3 Cl.1
PBMR-DPP □ South Africa□	280	9	-	SA-508 Gr.3 Cl.1
GT-MHR	495	7	-	9Cr-1Mo-V□ concept design□
HTTR (High temperature test reactor)	440	4.7	~1× 10^{17}□ 20a□	2.25Cr-1Mo
Sodium-cooled fast reactor	440	0.06	2.02× 10^{17}□ 40a□	316H
Thorium-based molten salt reactor	650	0.25		UNS N10003
GCMR	550	2	1× 10^{20}	□ □

The permissible material grade for RPV in ASME-III-D5

Table HBB-I-14.1(a)
Permissible Base Materials for Structures Other Than Bolting

Base Material	Spec. No.	Product Form	Types, Grades, or Classes
Types 304 SS and 316 SS [Note (1)], [Note (2)], [Note (3)]	SA-182	Fittings & Forgings	F 304, F 304H, F 316, F 316H
	SA-213	Smls. Tube	TP 304, TP 304H, TP 316, TP 316H
	SA-240	Plate	304, 316, 304H, 316H
	SA-249	Welded Tube	TP 304, TP 304H, TP 316, TP 316H
	SA-312	Welded & Smls. Pipe	TP 304, TP 304H, TP 316, TP 316H
	SA-358	Welded Pipe	304, 316, 304H, 316H
	SA-376	Smls. Pipe	TP 304, TP 304H, TP 316, TP 316H
	SA-403	Fittings	WP 304, WP 304H, WP 316, WP 316H, WP 304W, WP 304HW, WP 316W, WP 316HW
	SA-479	Bar	304, 304H, 316, 316H
	SA-965	Forgings	F 304, F 304H, F 316, F 316H
	SA-430	Forged & Bored Pipe	FP 304, FP 304H, FP 316, FP 316H
Ni-Fe-Cr (Alloy 800H) [Note (4)]	SB-163	Smls. Tubes	UNS N08810
	SB-407	Smls. Pipe & Tube	UNS N08810
	SB-408	Rod & Bar	UNS N08810
	SB-409	Plate, Sheet, & Strip	UNS N08810
	SB-564	Forgings	UNS N08810
$2\frac{1}{4}$Cr-1Mo [Note (5)]	SA-182	Forgings	F 22, Class 1
	SA-213	Smls. Tube	T 22
	SA-234	Piping Fittings	WP 22, WP 22W [Note (6)]
	SA-335	Forg. Pipe	P 22
	SA-336	Fittings, Forgings	F 22a
	SA-369	Forg. Pipe	FP 22
	SA-387	Plate	Gr 22, Class 1
	SA-691	Welded Pipe	Pipe $2\frac{1}{4}$ CR (SA-387, Gr. 22, Cl. 1)
9Cr-1Mo-V	SA-182	Forgings	F91
	SA-213	Smls. Tube	T91
	SA-335	Smls. Pipe	P91
	SA-387	Plate	91

Maximum allowable stress intensity for design condition calculations

SI Units

For Metal Temperature Not Exceeding, °C	304 SS	316 SS	Ni-Fe-Cr (Solution Annealed) UNS N08810	$2^1/_4$Cr-1Mo	9Cr-1Mo-V
375	...	...	...	123	184
400	...	...	...	123	178
425	105	110	105	116	172
450	102	108	104	116	165
475	101	108	103	99	154
500	99	107	101	81	133
525	86	101	99	64	117
550	74	88	89	48	102
575	69	77	74	35	81
600	65	76	68	26 [Note (1)]	62
625	51	62	62	...	46
650	42	51	51	...	29
675	34	39	41	...	...
700	27	30	34	...	...
725	21	23	28	...	...
750	17	18	23 [Note (2)]	...	...
775	14	13	...	...	...
800	11 [Note (3)]	11 [Note (4)]	...	...	...

NOTES:
(1) This is the value of S_o for $2^1/_4$Cr-1Mo at 593°C.
(2) At 760°C the value of S_o for UNS N08810 is 21 MPa.
(3) At 816°C the value of S_o for 304 SS is 9.7 MPa.
(4) At 816°C the value of S_o for 316 SS is 9.0 MPa.

2.1 Weight effect

	2.25Cr-1Mo	9Cr-1Mo-V	304H	316H	800H
Allowable stress/S0（550℃）	48	102	74	88	89
Thickness of cylindrical shell（mm）	83.7	38.8	53.7	45.0	44.5
Thickness of spherical shell（mm）	41.9	19.4	26.8	22.5	22.2
weight（ton）	**42.3**	19.1	26.7	**22.3**	22.0

Note: calculate the minimum wall thickness as specified in ASME-III-NB3324.

NB-3324 Tentative Pressure Thickness

The following equations are given as an aid to the designer for determining a tentative thickness for use in the design. They are not to be construed as equations for acceptable thicknesses. However, except in local regions (NB-3221.2), the wall thickness of a vessel shall never be less than that obtained from the equations in NB-3324.1 and NB-3324.2, in which:

P = Design Pressure
R = inside radius of shell or head
R_o = outside radius of shell or head
S_m = design stress intensity values (Section II, Part D, Subpart 1, Tables 2A and 2B)
t = thickness of shell or head

NB-3324.1 Cylindrical Shells.

$$t = \frac{PR}{S_m - 0.5P} \quad \text{or} \quad t = \frac{PR_o}{S_m + 0.5P}$$

NB-3324.2 Spherical Shells.

$$t = \frac{PR}{2S_m - P} \quad \text{or} \quad t = \frac{PR_o}{2S_m}$$

91 is the lightest, 22 material is heaviest, 304/316/800H close□

2.2 Irradated degradation effect

Designation: E 185 – 02

INTERNATIONAL

Standard Practice for
Design of Surveillance Programs
Nuclear Power Reactor Vessels[1]

This standard is issued under the fixed designation E 185; the number
original adoption or, in the case of revision, the year of last revision. A r
superscript epsilon (ε) indicates an editorial change since the last revisio

1. Scope

1.1 This practice covers procedures for designing a surveillance program for monitoring the radiation-induced changes in the mechanical properties of ferritic materials in the beltline of light-water moderated nuclear power reactor vessels. This practice includes the minimum requirements for the design of a surveillance program, selection of vessel material to be included, and a schedule for evaluation of materials.

1.2 This practice was developed for all light-water moderated nuclear power reactor vessels for which the predicted maximum fast neutron fluence ($E > 1$ MeV) at the end of the design lifetime (EOL) exceeds 1×10^{17} n/cm^2 (1×10^{21} n/m^2) at the inside surface of the reactor vessel.

For ferritic material ,when the fast neutron fluence exceeds $1*10^{17}$n/cm^2, the reactor vessel need to design a surveillance program, due to the fast neutron irradiation damage;

where as for the austenitic stainless steel, what's the effect and critical value of fast neutron irradiation?

Effect of neutron irradiation on the mechanical properties of 304

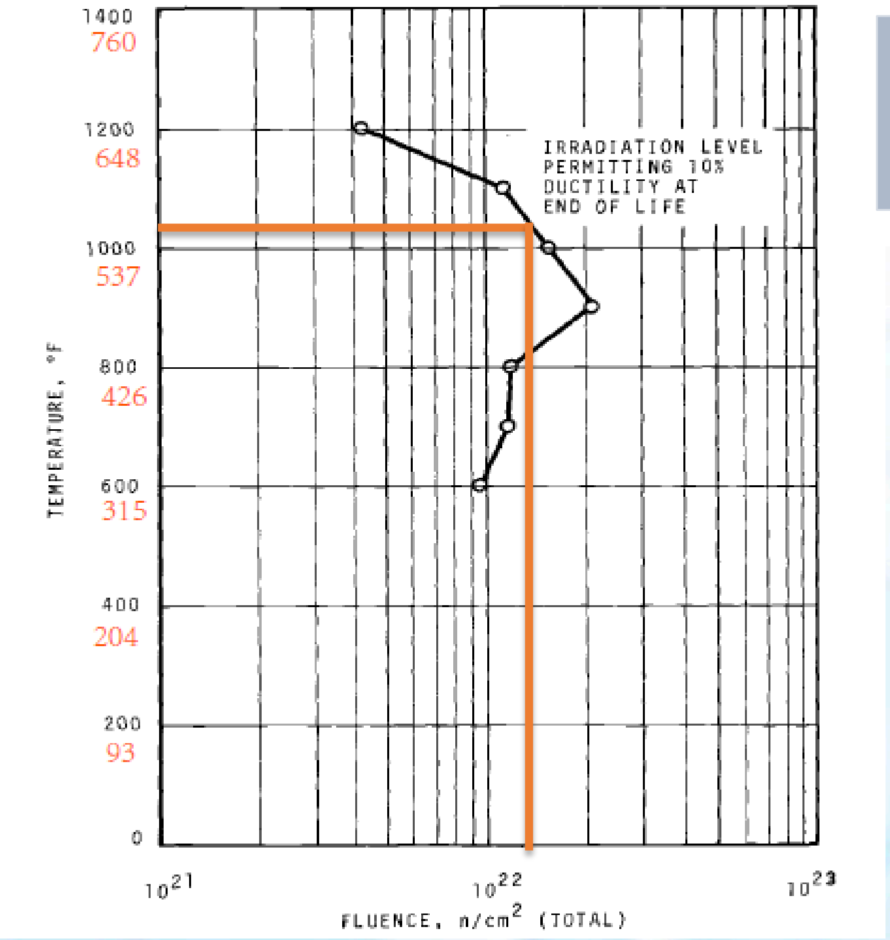
Property	Test temperature	Irradiation conditions	Irradiated specimen	Unirradiated specimen	%Change	Reference □ specimen number)
Offset yield strength□ $R_{p0.2}$	1100°F □ 593°C□	1.4× 10^{22}n/cm^2 E□ 0.1MeV 1004°F □ 540°C□	36ksi	15ksi	+140	10
Ultimate tensile strength□ R_m			46ksi	47ksi	-2	10
Total elongation			8%	55%	-85	Unirradiated ,10; Unirradiated, 11;
Rupture time for 30-ksi stress		1.4× 10^{22}n/cm^2	72h	110h	-35	12
Minimum creep rate for 30ksi stress		1.6× 10^{22}n/cm^2 E□ 0.1MeV 1100°F □ 593°C□	1× 10^{-4}h^{-1}	4.8× 10^{-4}h^{-1}	-72	13

R. A. Moen,[1] J. C. Tobin,[1] and K. C. Thomas[2]

Neutron Fluence Limit Determinations for Some Fast Flux Test Facility Components*

properties would fall within the overall safety envelope. At the present time BNWL and ARD have defined an arbitrary fluence limit for the core barrel, core support structure, and reactor vessel based on a 10 percent residual ductility at end of life.

Fluence limit for 304 stainless steel at different temperature



Fluence limit for type 304 stainless steel

the cirtial fast neutron fluence at 550°C is approximately $1.5\times\ 10^{22}$n/cm^2.

The effect of neutron irraidation on material selection

(1)According to the material service experience in PWR and Reactor material science , it's generally believed that austenitic stainless steel has obvious irradiation effect after 10^{21} n/cm^2 irradiation fluence, and the ferritic steel is about 10^{18}n/cm^2.

 2 The effect of fast neutron irradiation on ferritic material (2.25Cr-1Mo, 9Cr-1Mo-V)need to be considered, the effect of fast neutron irradiation on austenitic stainless steel material on (304/316) can be ignored, the effect of fast neutron irradiation on 800H need to be studied.

2.3 Fabrication effect

Material	Fabrication difficulty	Welding difficulty
2.25Cr-1Mo	easy	easy
9Cr-1Mo-V	Diffcult for big forging	diffcult
304	easy	easy
316	easy	easy
800H	easy	easy

2.4 Service experience

- 2.25Cr-1Mo used as RPV in HTTR□
- 9Cr-1Mo-V, no service experience□
- 316H, used as RPV in Sodium-cooled fast reactor;
- 304, no service experience;
- 800H, no service experience.

2.5 Advantages for 316H for RPV of GCMR-316H

- Good high-temperature property, and mature service experience as RPV;
- The neutron irradiation damage can be ignorable;
- It’s light with less weight and helpful for vehicle transportation;
- It’s relatively simple to manufacture and weld.

III. Verification of material performance as required in ASME code

- Creep rupture verification test
- Thermal aging effect
- Irradiation simulation test with ion injection
- Corrosion performance in the helium with impurities

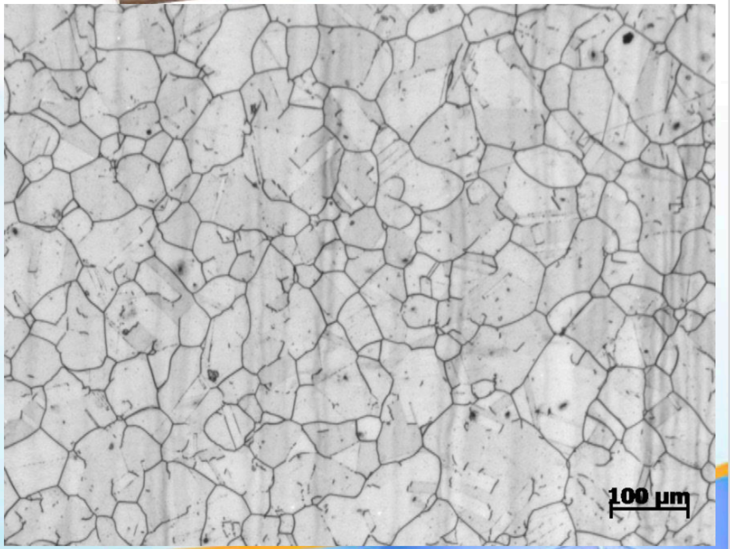
HBB-2160 DETERIORATION OF MATERIAL IN SERVICE

(a) Consideration of deterioration of material caused by service is generally outside the scope of this Subsection. It is the responsibility of the Owner to select material suitable for the conditions stated in the Design Specifications (NCA-3250), with specific attention being given to the effects of service conditions upon the properties of the material.

Test material composition(wt%)-316H

C	Si	Mn	P	S	Cr
0.04-0.06	≤0.60	1.0-2.0	≤0.030	≤0.020	17.0-18.0
Ni	Mo	Al	Sb	B	Co
11.0-12.5	2.5-3.0	≤0.05	≤0.02	≤0.003	0.02
Pb	Se	Sn	V	Zn	N
≤0.003	≤0.015	≤0.015	≤0.05	≤0.01	0.04-0.07

As reruired in HBB-U-I, the melt method shall be AOD OR AOD/ESR.



3.1 Creep property in ASME-III-D5

Table HBB-I-14.6B
Expected Minimum Stress-to-Rupture Values, 1,000 psi (MPa), Type 316 SS

U.S. Customary Units

Temp., °F	1 hr	10 hr	30 hr	10^2 hr	3×10^2 hr	10^3 hr	3×10^3 hr	10^4 hr	3×10^4 hr	10^5 hr	3×10^5 hr
800	64.5	64.5	64.5	64.5	64.5	64.5	64.5	64.5	64.5	64.5	64.5
850	63.3	63.3	63.3	63.3	63.3	63.3	63.3	63.3	60	56	52
900	62.2	62.2	62.2	62.2	62.1	62	58	54.1	48	42.6	38
950	60	60	60	60	56	51.6	46.5	42.6	37.5	32.4	28.3
1,000	58.5	58.5	55	51.7	47	42.1	37.5	33.6	28.8	24.6	21
1,050	56	52.9	47.5	43.4	38.2	34.4	30.2	26.4	22.3	18.8	16
1,100	53.5	45.1	40	36.4	32.2	28.1	24.2	20.8	17.3	14.3	11.7
1,150	46.5	38.4	34	30.5	26.6	23.0	19.5	16.4	13.4	10.9	8.8
1,200	40	32.7	29	25.6	22	18.8	15.6	12.9	10.3	8.3	6.7
1,250	35	27.8	24.3	21.4	18.1	15.4	12.7	10.2	8.1	6.3	4.9
1,300	30	23.7	20.8	18.0	15	12.5	10.0	8.0	6.2	4.8	3.7
1,350	26	20.0	17.5	15.0	12.7	10.4	8.2	6.4	4.9	3.6	2.7
1,400	22.5	17.1	14.8	12.4	10.2	8.4	6.6	5.0	3.8	2.8	2.1
1,450	19.5	14.6	12.6	10.5	8.6	6.8	5.2	3.9	2.9	2.1	1.5
1,500	17	12.5	10.6	8.8	7.2	5.6	4.2	3.1	2.3	1.6	1.2

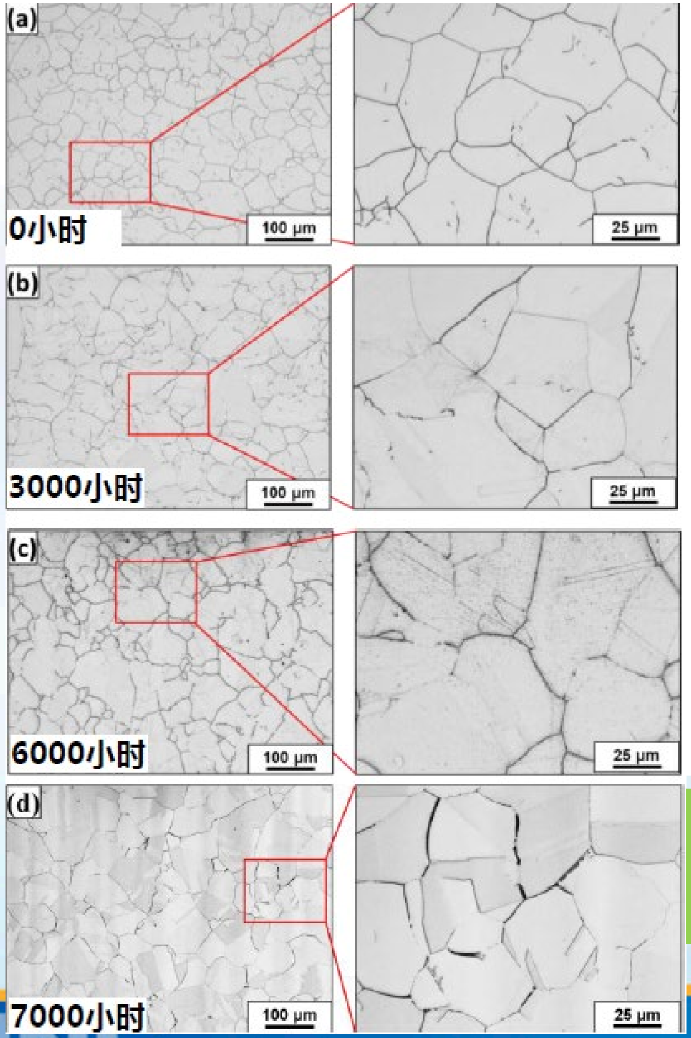
SI Units

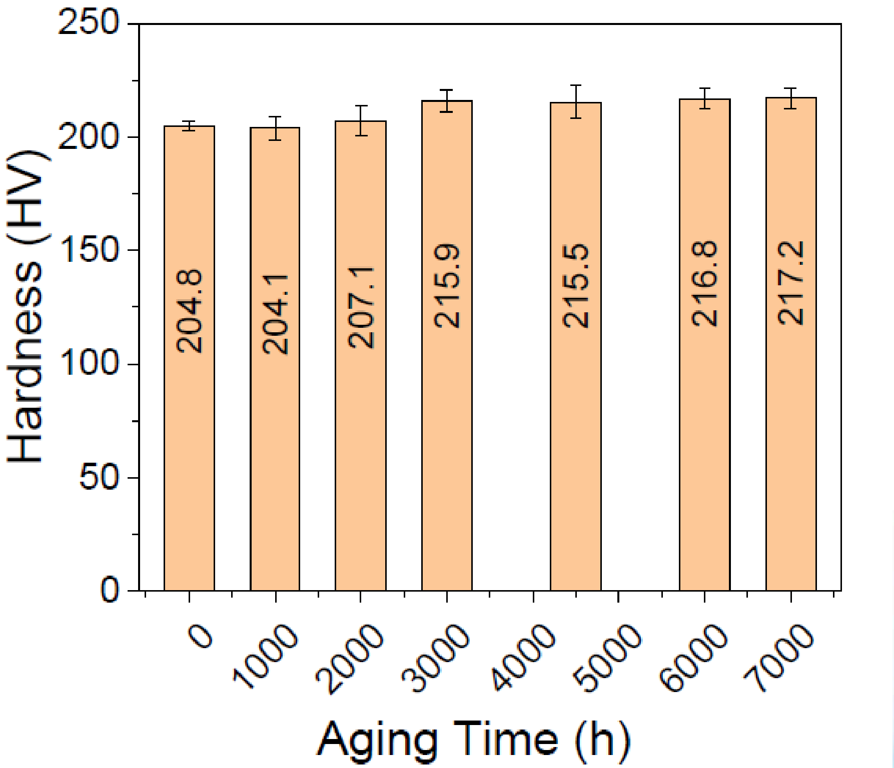
Temp., °C	1 h	10 h	30 h	10^2 h	3×10^2 h	10^3 h	3×10^3 h	10^4 h	3×10^4 h	10^5 h	3×10^5 h
425	445	445	445	445	445	445	445	445	445	445	445
450	437	437	437	437	437	437	437	437	419	395	372
475	431	431	431	431	430	429	409	389	352	317	286
500	419	419	419	419	401	381	349	322	285	248	219
525	406	406	388	371	340	307	275	248	226	183	158
550	393	381	350	323	289	268	230	203	173	147	125
575	380	347	311	283	249	223	194	169	142	120	100
600	357	300	266	241	212	185	159	136	112	94	79
625	315	259	229	205	179	155	130	110	89	72	59
650	275	224	199	176	151	129	107	88	70	57	46
675	244	194	170	150	127	108	89	71	57	44	35
700	212	167	147	128	106	89	72	57	45	34	27
725	186	144	127	108	92	76	60	47	36	27	21
750	163	125	109	91	76	63	50	38	29	21	16
775	144	109	94	78	64	52	41	30	23	16	12
800	124	92	79	65	54	42	32	24	18	12	9

Stress rupture test results

number	Temperature □ ℃□	Test stress in the code□ MPa□	Stress to rupture time □ h□	Test stress in the code □ MPa□	Stress to rupture time □ h□	Test stress in the code □ MPa□	Stress to rupture time □ h□
1	550	268 □ 1000h□	1717	230□ 3000h□	3419	203 (10000h)	□ 8000
			1577		3358		□ 8000
			1629		3505		□ 8000
2	575	223 (1000h)	1300	194 (3000h)	4042	169 (10000h)	□ 8000
			1242		3794		□ 8000
			1148		4071		□ 8000
3	600	185 (1000h)	1505	159 □ 3000h)	4251	136 (10000h)	□ 8000
			1417		4010		□ 8000
			1573		4328		□ 8000

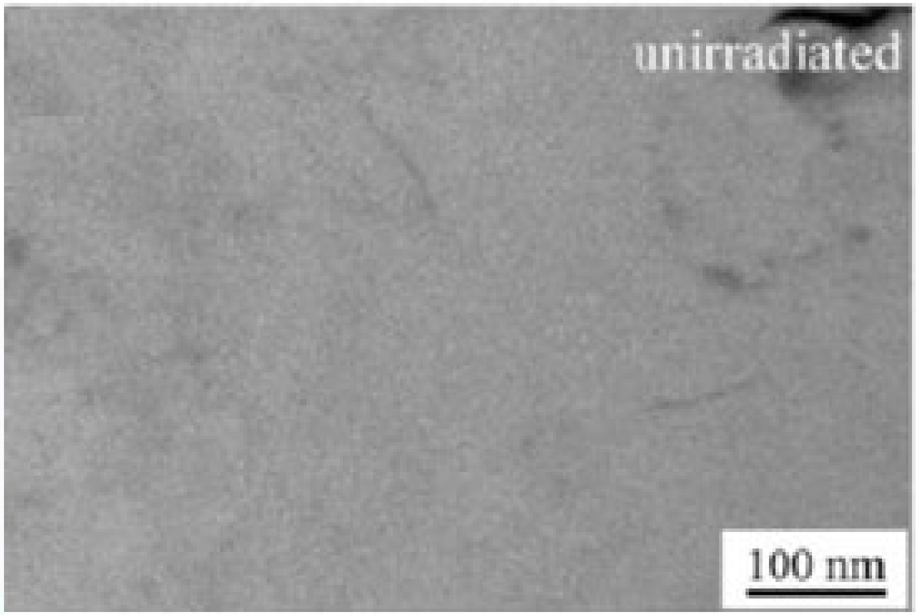
3.2 Thermal aging test

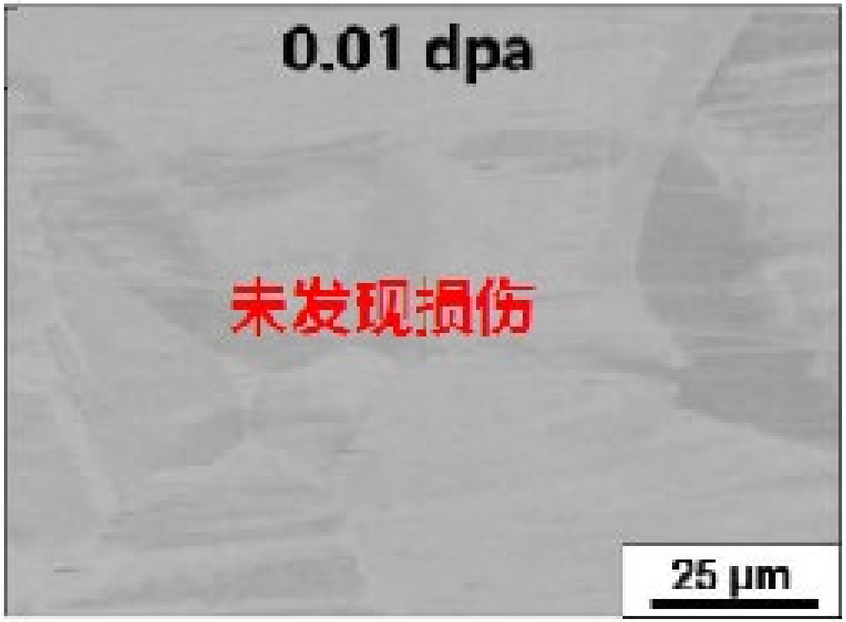




After thermal aging at 550°C/7000h, the hardness of the sample did not change much, and the carbide precipitated along the grain boundary.

3.3 Irradiation damage simulation



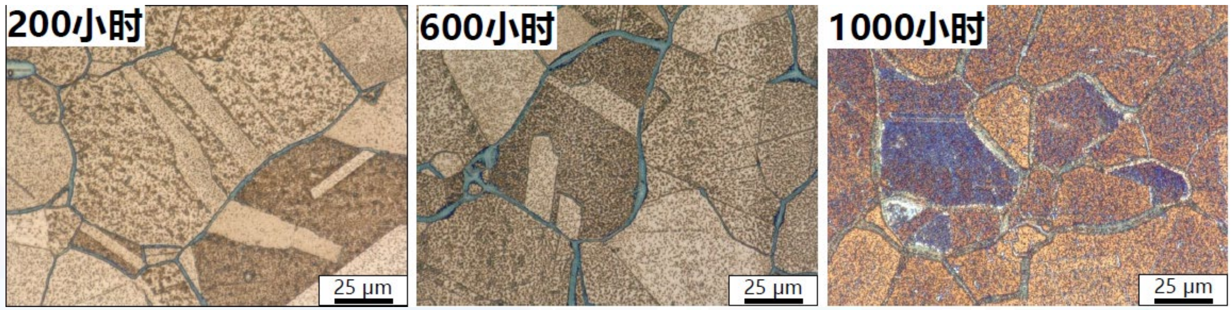


The neutron fluence was converted to dpa, which is about 0.01dpa; Fe^{n+} was used to simulate neutron inrradiation at a depth of 1-1.5μm□ and there is none damage phenomenon.

3.4 The corrosion test in helium with impurities

Impurity	Volume concentraton (ppm,10^{-6})	10xVolume concentration (vol%,10^{-2})	100xVolume concentration (vol%,10^{-2})	1000xVolume concentration (vol%,10^{-2})
CO_2	6	0.006	0.06	0.6
H_2	30	0.03	0.3	3.0
CO	30	0.03	0.3	3.0
CH_4	5	0.05	0.5	0.5
N_2	2	0.02	0.2	0.2
O_2	2	0.02	0.2	0.2
He	Bal	Bal	Bal	Bal

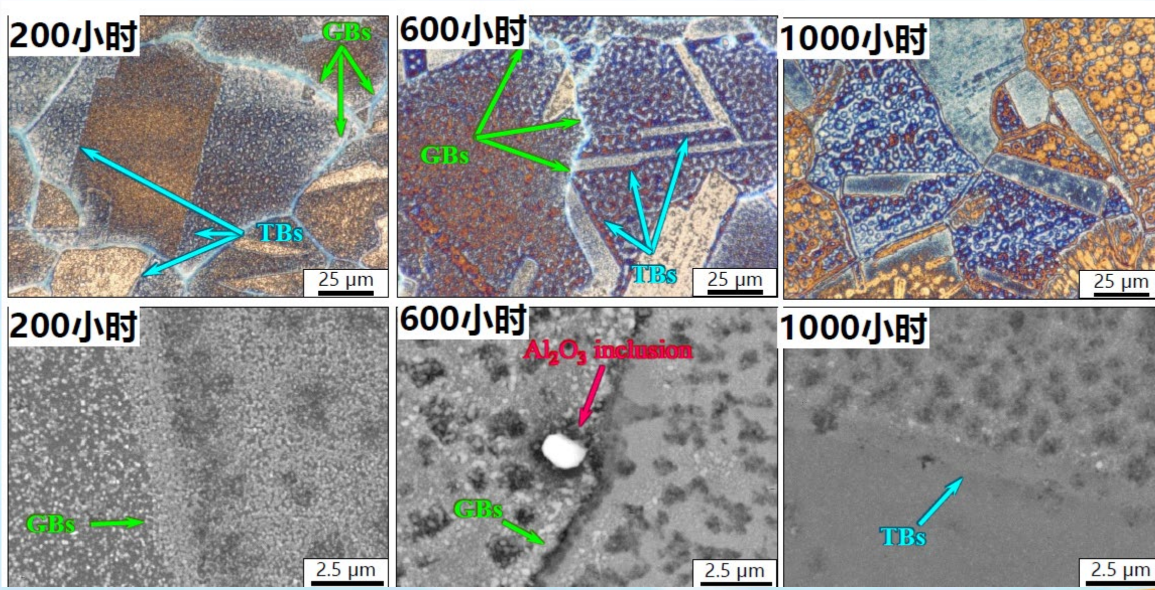
Corrosion test results



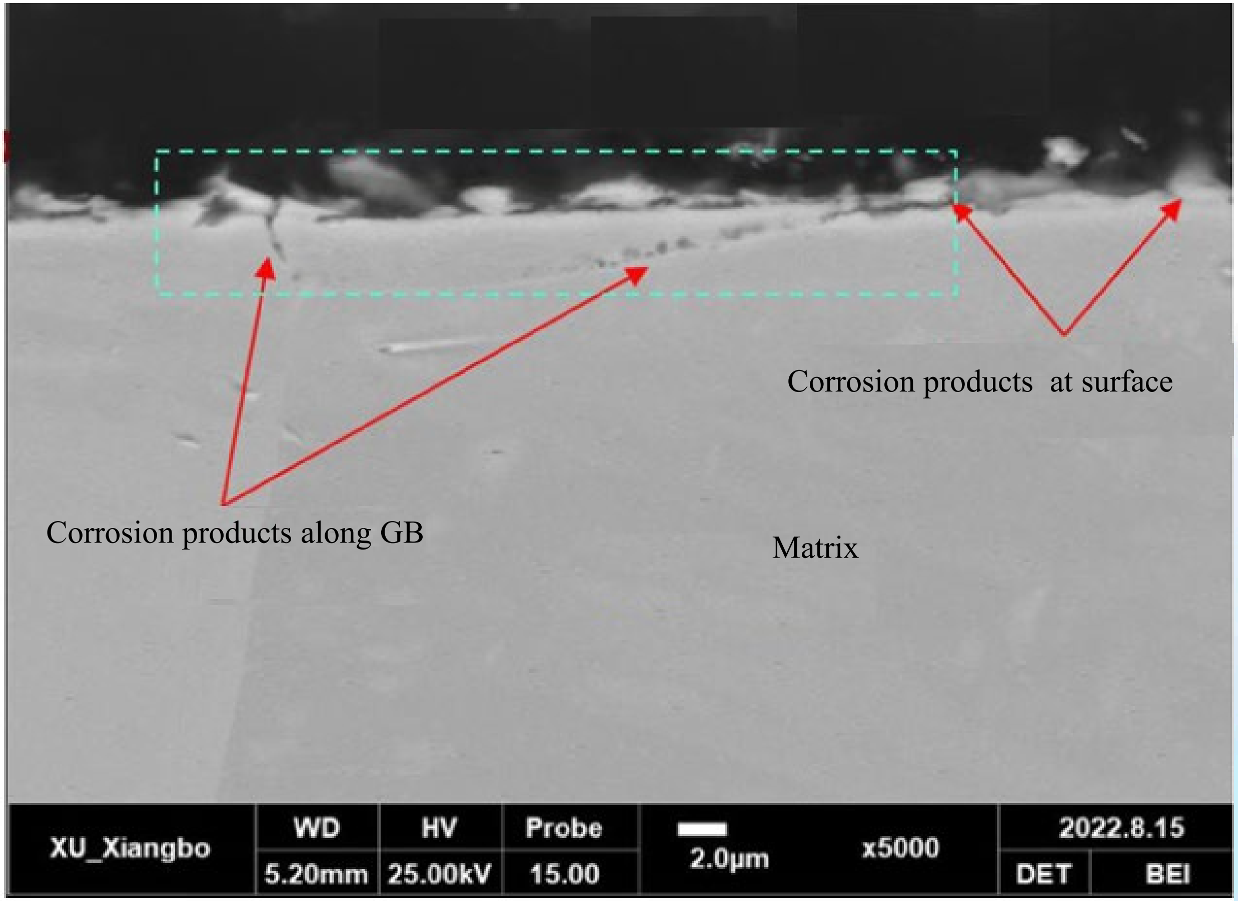
The corrosion phenomenon at 10 times volume concentration at 550 °C

optical

SEM



The corrosion phenomenon at 1000 times volume concentration at 550 °C



SEM cross-section corrosion test results after 1000h at 550℃ with 1000 times volume concentration

4. Conclusion

- The stress-rupture of 316H with specific melting method was higher than the values stipulated in the code;
- The carbide precipitated along grain boundary after 7000h at 550°C, whereas no brittle phase precipitated, thus the thermal aging effect is not sensitive;
- The irradiation damage at the required fast neutron fluence (0.01dpa) can be negligible;
- The corrosion effect in the helium with impurities in certain range can be negligible;
- It can be determined that 316H can be used as GCMR-RPV material.

Thanks for your attention!