DETERMINATION OF THE LOWER TEMPERATURE LIMIT OF VOID SWELLING OF STAINLESS STEELS AT RELATIVELY LOW DISPLACEMENT RATES - S.I. Porollo, Yu.V. Konobeev, A.M. Dvoriashin, and V.M. Krigan (Institute of Physics and Power Engineering, Russia), F. A Garner (Pacific Northwest National Laboratory)*

OBJECTIVE

The purpose of this effort is to determine the lower temperature limit of void swelling in austentic steels.

SUMMARY

An issue of current interest to PWRs is the possibility that void swelling of austenitic near-core internal components may exert some deleterious effect on component functionality, particularly during extended operation to 60 years. A similar concern has also been raised for water-cooled fusion devices. One question of particular interest is the range of temperature over which void swelling can occur, since the internal components experience temperatures from ~290 to perhaps as high as 390°C in some limited locations. This question was addressed using a flow restrictor component from the low-flux breeder zone of the BN-350 fast reactor in Kazakhstan. This component was constructed of annealed 12X18H10T, an alloy similar to AISI 321 which is used in Russian reactors for applications where AISI 304L would be used in comparable Western and Japanese reactors. Extensive sectioning to produce 114 separate specimens, followed by examination of the radiation-induced microstructure showed that void swelling in the range of temperatures and dpa rates of PWR interest occurs down to ~305°C. At 330°C the swelling reached ~1% at 20 dpa. Comparison of these data with other published results from Russian LWR reactors at <10 dpa confirms that the lowest temperature that stainless steels can begin swelling appears to be ~300°C. Since fusion and PWR spectra generate similar levels of hydrogen and helium, it is expected that these conclusions are also applicable to fusion devices operating at comparable dpa rates.

INTRODUCTION

Voids as a new type of radiation defect were discovered in 1967 to form in austenitic stainless steel fuel pin cladding after irradiation in the Dounreay Fast Reactor [1]. Since that time thousands of scientific papers have been written on this subject, with most data derived from fast reactors, some high-flux thermal reactors and experiments utilizing charged particle irradiations at very high damage rates. An extensive summary of swelling data for stainless steels is presented elsewhere [2].

For a long time it was assumed that neutron-induced void formation was characteristic only of fast reactor or test reactor irradiation with very high atomic displacement rates and elevated irradiation temperatures of 370 to 650°C. Recently, however, new insight and data became available showing that void formation may occur in the internal structural components of Western PWR and Russian VVER internals, which operate at lower temperatures (280 to 430°C) and significantly lower neutron fluxes [3-12]. In spite of these low fluxes these components can accumulate damage exposures during typical reactor life times (30-40 years) as high as 50-100 dpa. Similar conditions can be found in some fusion device designs.

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In addition to dimensional changes, the swelling in structural materials of PWR and VVER internals (in particular, austenitic stainless steels such as AISI 304 in PWRs and X18H10T in VVERs) may lead to the generation of mechanical stresses caused by the non-uniformity of swelling over each component, and also to severe void-induced embrittlement for swelling levels above approximately 10% [2, 13].

The upper temperature limit of void formation is related to the lower supersaturations of vacancies at high temperatures. The lower temperature limit of void formation is thought to arise from low vacancy mobility and/or to the inability of vacancies to aggregate and nucleate voids. It is the lower temperature limit that is of interest to PWRs and VVERs. Is it possible that void swelling can occur over all or a large portion of the PWR, and VVER fusion operating temperature range?

Available data at high damage exposure on swelling of AISI 304 stainless steel in fast reactors are in general limited to temperatures above 380°C, reflecting the relatively high inlet coolant temperature of Western fast reactors. It is therefore very desirable to collect high fluence data at lower PWR-relevant temperatures and hopefully at lower PWR-relevant damage rates. Since the overwhelming majority of data on AISI 304 and 304L were generated in EBR-II with an inlet temperature of 370°C, no additional PWR-relevant data can be obtained without destructively examining actual PWR components. However, it is possible to examine X18H10T components removed from both fast reactors and VVER reactors from the former states of the Soviet Union, choosing components that operated at PWR-relevant conditions. Since the overall swelling behaviors of all stainless steels are in general similar (2), it is thought possible to use data on this Russian steel to infer the lower temperature limit of swelling of AISI 304L in PWRs of Western or Japanese design.

The present paper presents the final results of a microstructural study designed to determine the swelling of type12X18H9T austenitic stainless steel after long-term irradiation at PWR-relevant damage rates in the BN-350 fast reactor in Kazakhstan at temperatures ranging from 280°C to 334°C to doses from 4 to 56 dpa. A previous paper presented the initial results of this experiment [12]. A comparison will also be made with results from other Russian reactors at comparable irradiation conditions.

The path chosen in this experiment was to assess the lowest temperature at which swelling could occur over a range of dpa rates and temperatures relevant to PWR internals and some fusion conditions. Transmission electron microscopy of many small specimens was used to determine the swelling, even at very low levels, avoiding resolution problems associated with bulk measurement techniques such as immersion density.

1. *Material and irradiation conditions.*

To study the swelling of type 12X18H9T steel after low-temperature neutron irradiation, one of the out-of-core components of BN-350 was chosen. This component served as a flow restrictor to the sodium coolant in the breeder zone. The component was a hexagonal tube of 96 mm flat-to flat size, having a central cylindrical hole 65 mm in diameter. The total length of the flow restrictor component equals the 3.435 m length of reference driver subassemblies of the BN-350 fast reactor.

This flow restrictor component was made from annealed type 12X18H9T austenitic stainless steel of the following nominal specification (wt. %): C≤ 0.12; Cr, 17-19; Ni, 8-9.5; Mn ≤2; Si ≤0.8; Ti, 0.5-0.7. The nearest Western equivalent to this steel is AISI 321 stainless steel. The composition of the examined component was confirmed using an "Kamebax" x-ray

micro-analyzer in scanning mode on an irradiated specimen and found to have the following composition (wt.%): Cr, 18.5; Ni, 9.5; Mn, 1.5; Si, 0.6-0.8; Ti, 0.6-0.7. Thus, the measured composition falls within the specification.

The component was exposed without removal or rotation for 46,536 h (November, 1972 till June, 1984) in the breeder zone of the BN-350 reactor at a radial distance of 94.5 cm from the core axis. The maximum neutron fluence at the core midplane was equal to 3.3×10^{23} n/cm^2 (E>0) or 1.6×10^{23} n/cm^2 (E>0.1 MeV). This fluence in the softer neutron spectra of the out-of-core region corresponds to a neutron damage dose of 56 dpa. Hence, the maximum dose rate was 1.56×10^{-7} dpa/s, comparable to the damage rate experienced by the re-entrant corners of many PWR baffle plates at mid-core levels. The lowest dose regions of the restrictor explored in this experiment reached ~4 dpa at 1.12×10^{-8} dpa/s, a damage rate comparable to that experienced by many regions of PWR former plates and core barrel.

The distribution of irradiation temperatures for both the inboard and outboard flats of the flow restrictor are shown in Table 1. The irradiation temperatures in the component were calculated at the average 60% nominal reactor capacity for the period from 1973 to 1984. The inlet sodium temperature in the core was 280°C. All calculated temperatures are judged to be accurate within ±3°C.

There is a somewhat complicated distribution of neutron fluence, dose, and irradiation temperature throughout the component. Due to different cooling conditions the maximum difference of temperature between internal and outer component surfaces reaches 30°C, the neutron fluence and dose between two arbitrary points in a cross section of the component might differ by roughly a factor of two.

2. Specimen preparation

In the hot cell of the BN-350 facility the flow restrictor component was cut into 5 pieces, each of 300 mm length. Sections #1, 3 and 5 (see Fig. 2) corresponding to the bottom, middle and top parts of the component were transported to the IPPE laboratory in Obninsk, Russia. Fig. 1 illustrates the range of dose-temperature irradiation conditions for each component piece. The areas of the map corresponding to absent pieces are shaded. As shown in Fig. 1, the accessible area of dpa values and irradiation temperatures has a complicated form, and is bounded by doses of 4 and 56 dpa, and irradiation temperatures of 282 and 335°C. The dpa-temperature areas corresponding to the three different pieces of the component overlap partially, allowing one to get some information which might have been obtained from absent pieces #2 and #4.

Table 1: The distribution of irradiation temperature (°C) in different cross sections of the flow restrictor component

Distance from midplane, mm	Outer surface, inboard flat	Internal surface, inboard flat	Outer surface, outboard flat	Internal surface, outboard flat
-750	282	282	281	281
-450	290	287	288	286
-150	306	296	303	293
0	317	302	313	299
+150	324	304	321	301
+450	334	307	333	306
+750	337	307	336	306

To obtain the most complete information on swelling the cutting scheme of the TEM-specimens shown in Fig. 2 was chosen. First, the section was sectioned to produce several slices of ~0.6 cm thickness. To prepare specimens for the electron microscopy investigation (disks of 3 mm in diameter and of 0.3-0.4 mm in thickness) several samples were cut from each edge of the component (7 sets of two specimens) and from the middle of each face (5 sets of two specimens). The scheme of cutting of microscopy specimens shown in Fig. 2 was utilized for the three cross sections located with their centers at distances of 0, -150 and +590 mm from the core midplane. For the cross section with its center located at -670 mm, TEM specimens were cut from only three faces. Such a cutting scheme allowed us to fill rather uniformly the area of the dpa–irradiation temperature map with experimental points (shown as crosses in Fig. 1). A total of 114 specimens were examined by microscopy.

3. Microstructure and swelling of the 12X18H9T stainless steel

Data on irradiation conditions and swelling for the cross sections located at distances of -670, -150 and 0 mm from the core midplane are shown in Table 2. Voids were observed in sections -150, 0 and +590 (examples in Fig. 4), but no voids were detected in sections -670 and -450 mm. Fig. 3 shows the dose-irradiation temperature map of void formation observed in the 12X18H9T stainless steel flow restrictor.

Voids, dislocation loops, dislocation segments, two types of precipitates, and deformation stacking faults characterize the microstructure of 12X18H9T steel in the specimens investigated. The dislocation structure of the irradiated steel consists of both faulted and perfect dislocation loops, and also linear dislocations (Fig. 5). The total length of dislocations per unit volume depends slightly on the irradiation temperature.

In the irradiated steel two types of precipitates were observed. The first type was cubical with mean diameter of 59 nm and a rather low concentration of 5×10^{14} cm^{-3}. The volume fraction of these precipitates equals 0.72%. From electron diffraction patterns it was determined that these are TiC precipitates, which probably formed in the steel during crystallization, but new precipitates appear to have formed during irradiation. It should be noted that this type of precipitate was observed in all examined sections of the flow restrictor component regardless of temperature or damage dose. The other type of precipitate is very small, finely dispersed precipitates in the grain interior (Fig. 6). An analysis has shown that these precipitates are G-phase. These did not exist prior to irradiation and are well known to be a radiation-induced phase, especially in Ti-modified stainless steels [2]. As shown in Table 3, the diameter of G-phase particles increases in general and the concentration decreases with increasing irradiation temperature.

Along with the defects mentioned above, long stacking faults were often observed in the irradiated steel (Fig. 7), probably due to deformation of TEM specimen foils during preparation.

4. Results and discussion

In Figs. 8-10 the temperature dependence is shown of the mean void diameter $\langle dv \rangle$, void concentration Nv, and swelling. Despite the range of dpa and dpa rates in this dose/temperature range it is seen that the mean void diameter increases monotonically with increase of irradiation temperature, with an activation energy of about 1 eV. In contrast

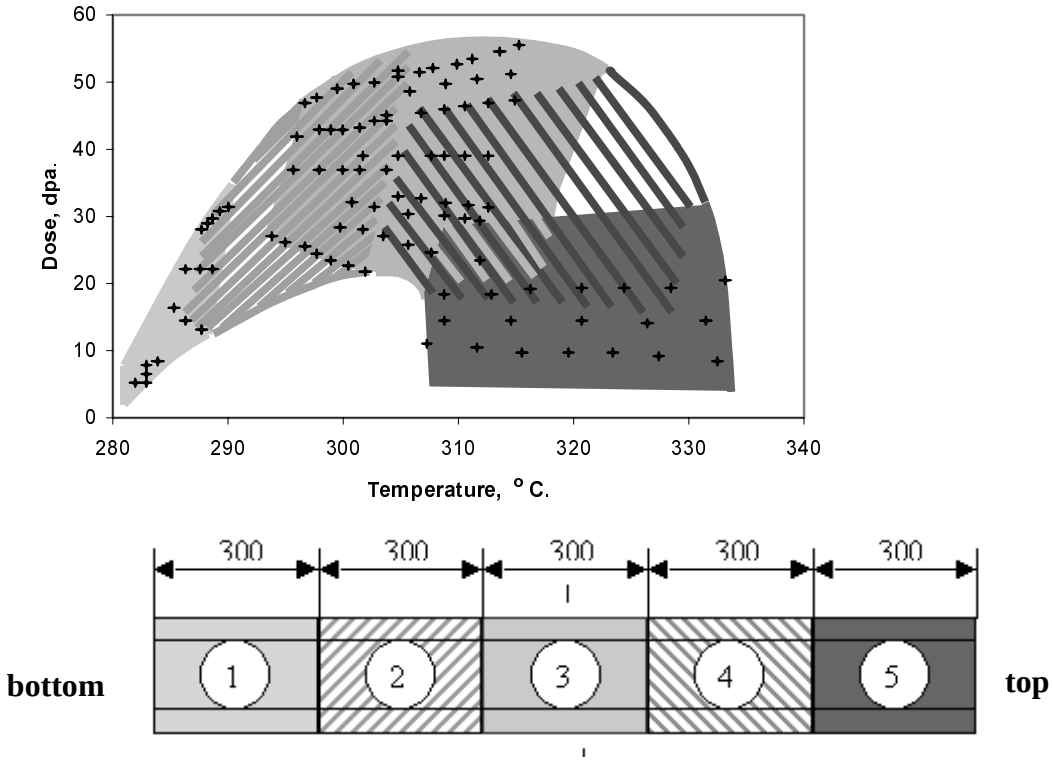

Figure 1. Dose-temperature map of irradiation conditions and the scheme of initial cutting of the flow restrictor component. Dimensions are in mm. The middle of section 3 corresponds to the core midplane. Crosses represent the conditions at which microscopy was performed.

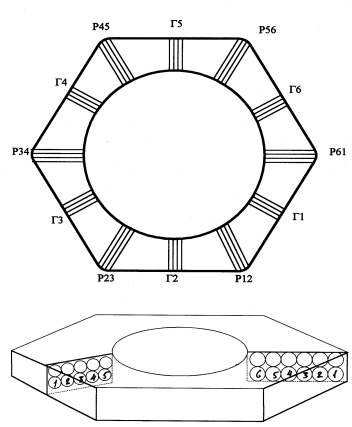

Figure 2. Cutting scheme of microscopy specimens for component cross sections with centers located at distances of 0, -150 and +590 mm from the core midplane. P and Г are derived from the Russian terms for corner and flat, respectively.

Table 2: Microstructural data

Sample number	Dose, dpa	Temperature, °C	Mean void diameter, nm	Void concentration, 10^{14} cm^{-3}	Void volume fraction, %
-150_P34.1	21.6	302	4.4	Voids in single grains	
-150_P56.3	43.4	302	4.9	0.82	0.0008
-150_P56.3	43.6	302	4.9	0.8	0.001
-150_P61.3	50.3	302	5.1	1.2	0.001
-150_P56.2	43.8	303	4.9	1.9	0.002
-150_P56.2	44	303	4.9	1.9	0.002
-150_P56.1	42.2	304	4.4	1.4	0.00144
-150_P61.2	51.2	304	5.0	1.4	0.02
-150_P56.1	44.5	305	4.4	1.4	0.0014
-150_P61.1	52.1	305	4.8	1.7	0.02
0_P34.4	25.7	306	7.9	1.5	0.006
0_Г4.3	30.5	306	5.2	0.5	0.001
0_P23.4	32.5	307	6.4	0.78	0.002
0_P56.5	45.5	307	4.7	1.96	0.002
0_P61.5	51.7	307	4.7	1	0.001
590_P34.7	10.0	308	9.4	8.4	0.06
+590_P34.7	10.3	308	9.4	8.4	0.06
0_P34.3	24.8	308	9.2	3.1	0.02
0_Г2.3	39.2	308	5.4	0.84	0.001
0_Г5.3	39.2	308	5.3	0.52	0.001
0_P61.4/5	52.1	308	4.9	0.9	0.001
590_Г5.5	13.7	309	9.5	6	0.04
590_Г2.5	13.7	309	8.0	6	0.03
+590_Г2.5	14.1	309	8.3	6	0.03
+590_Г5.5	14.1	309	9.5	6.3	0.04
590_61.7	17.5	309	9.0	8	0.05
+590_P61.7	18	309	9.0	7.9	0.05
0_Г4.2	29.9	309	6.0	0.91	0.002
0_P23.3	32.0	309	8.1	1.2	0.006
0_Г2.2/5	39.2	309	5.4	0.78	0.001
0_P56.4	45.9	309	6.2	4.7	0.01
0_Г1.3	49.8	309	6.0	1.08	0.002
0_P61.4	52.6	310	5.0	1.4	0.002
0_Г4.1/5	29.5	311	7.0	2.1	0.007
0_P23.2	31.6	311	8.5	2.9	0.013
0_Г2.2	39.2	311	6.6	1.7	0.004
0_Г5.2	39.2	311	6.6	1.8	0.005
0_P56.3	46.4	311	6.2	4.75	0.01
+590_P34.6	9.9	312	9.0	21	0.12
0_P34.1	23.0	312	9.0	7.1	0.044
0_Г3.1	29.2	312	8.7	0.28	0.001
0_Г4.1	29.2	312	8.0	2.6	0.012
0_Г1.2	50.6	312	6.6	2.7	0.006
0_P61.3	53.6	312	5.9	4.5	0.009
+590_P61.6	18.3	313	10.1	22	0.19

0_Р23.1	31.1	313	9.1	2.1	0.013
0_Г2.1	39.2	313	8.7	0.85	0.004
0_Г5.1	39.2	313	7.2	8.6	0.024
0_Р56.2	46.8	313	6.9	13.9	0.037
0_Р61.2	54.5	314	6.7	10	0.026
+590_Г2.4	14.1	315	9.0	15	0.09
+590_Г5.4	14.1	315	10.4	18	0.16
0_Р56.1	47.3	315	7.0	15.8	0.049
0_Г6.1	51.4	315	7.1	7.9	0.021
+590_Р34.5	9.6	316	11.0	38	0.4
0_Р61.1	55.4	316	7.1	9.2	0.024
+590_Р61.5	18.6	317	11.4	20	0.24
+590_Р34.4	9.3	320	13.1	37	0.65
+590_Г2.3	14.1	321	10.4	19	0.18
+590_Г5.3	14.1	321	11.8	22	0.3
560	16.8	321	12.4	12	0.17
+590_Р61.4	19	321	12.5	27.5	0.44
+590_Р34.3	8.9	324	14.0	27	0.6
+590_Р61.3	19.3	325	14.2	22	0.5
+590_Г2.2	14.1	327	11.1	29	0.32
+590_Г5.2	14.1	327	12.8	21	0.35
+590_Р34.2	8.6	328	18.0	25	0.95
590_Р34.2	8.4	329	18.0	25	0.96
+590_Р61.2	19.6	329	14.0	21	0.45
+590_Р34-Г4	8.4	332	16.9	22	0.69
+590_Г5.1	14.1	332	15.0	24	0.67
+590_Г2.1	14.1	332	13.8	22	0.47
590	14.5	332	17.3	6.4	0.28
590_Р34.1	8.1	333	18.0	18	0.77
+590_Р34.1	8.3	333	17.0	20	0.70
590_Г2.1	13.7	333	14.0	22	0.50
590_Г5.1	13.7	333	15.0	24	0.70
590_Р61-Г1.1.4	18.8	334	16.7	18	0.65
590_Р61-Г1.1.3	19.0	334	14.4	22	0.50
590_61.1	19.4	334	17.0	14	0.50
+590_Р61.1	19.9	334	17.0	14	0.49

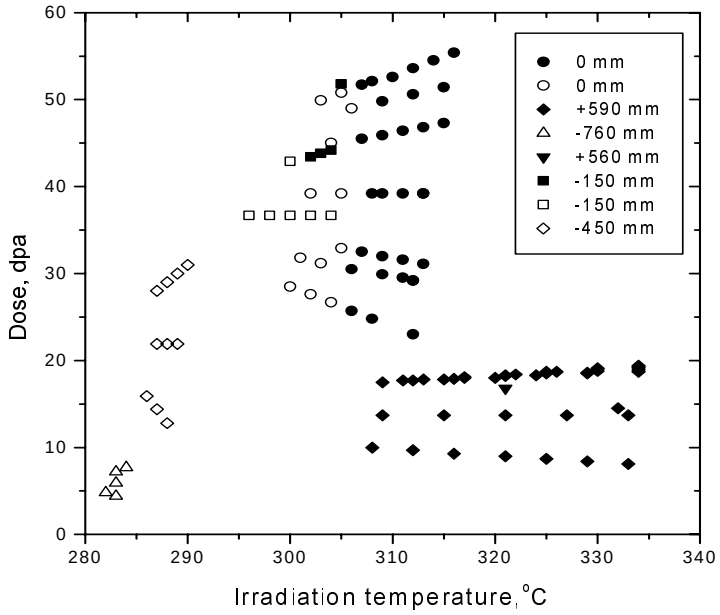

Figure 3. The dose-irradiation temperature map of void formation in the 12X18H9T steel irradiated in BN-350. Solid symbols denote that voids were observed. Open symbols denote the absence of observable voids.

to the mean void diameter, the void concentration appears to behave non-monotonically with increasing temperature. This behavior is thought to reflect primarily the shape of the dose-temperature space shown in Fig. 1, however, rather than any parametric sensitivity of swelling. The range of swelling shown in Figs. 3 and 10 is clearly bounded on the lower temperature side at about ~305°C over the full range of dpa rates examined. Below this temperature no voids were observed, even at ~50 dpa.

At a temperature of ~330°C and doses ranging from 8 to 20 dpa the swelling level ranges from ~0.3 to 1.0 %, depending on both the dpa and dpa rate. At the highest temperature examined, ~334°C, the swelling ranged from 0.5 to 0.8% at ~19 dpa. Most significantly, swelling below ~305°C did not occur even for doses on the order of 50 dpa, indicated that some thermal barrier exists to void nucleation that can not be overcome by continued irradiation, at least to dpa levels that reach a very large fraction of the exposure expected in operating PWRs and VVERs. On the other hand, the region of the PWR baffle/former assembly operating above 305°C comprises a very large fraction of the total.

Neustroev and co-workers have recently published a series of swelling data on this same steel from several LWR reactors, with all data derived at comparably low dpa rates [14]. Fig. 11 shows a comparison of these data with those of the current study. In general, the two

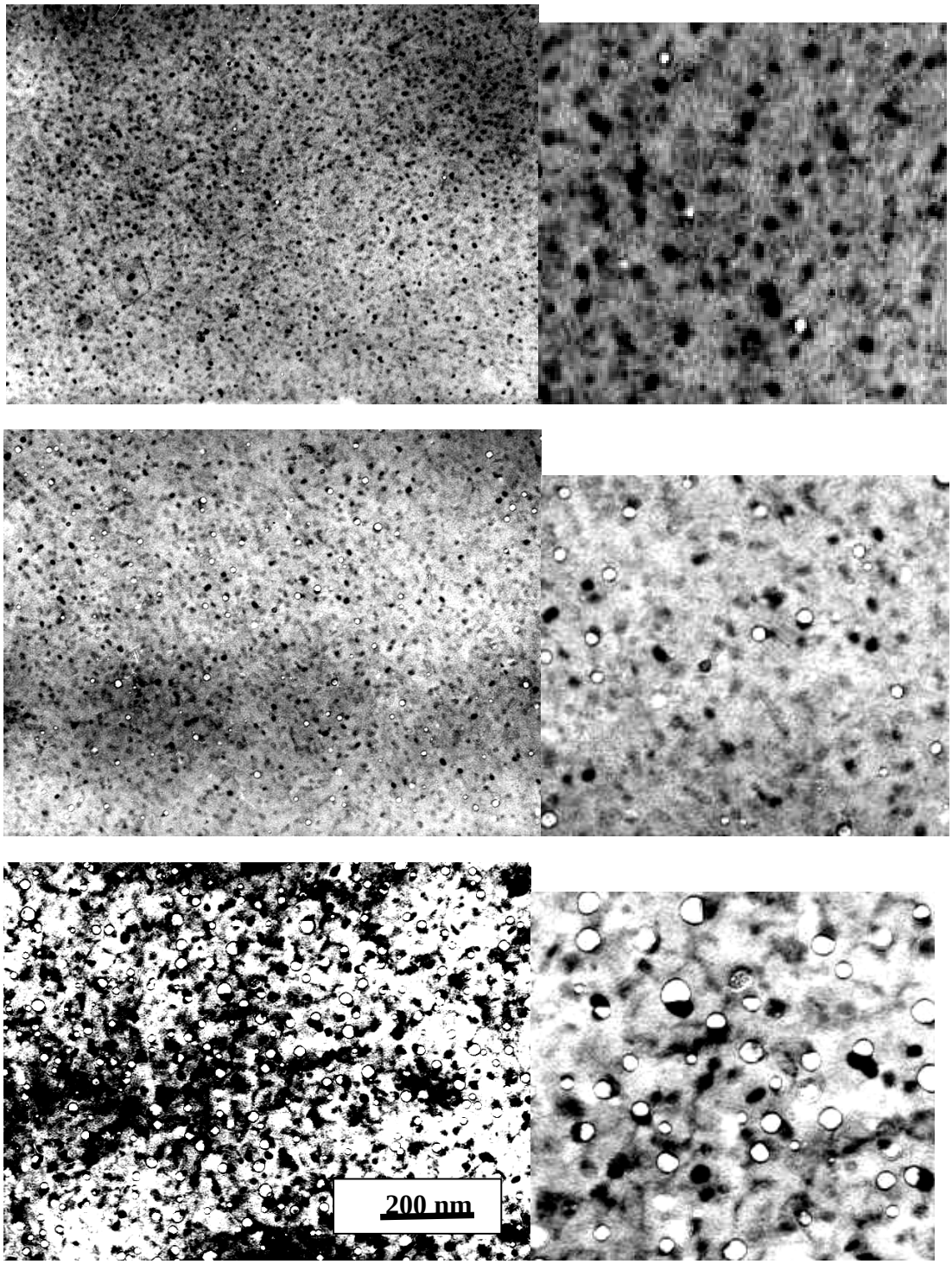

Figure 4. Voids observed in the 12X18H9T steel after neutron irradiation: 1) barely observable at 302°C to 50.3 dpa, 2) more visible at 308°C to only 10.3 dpa, and 3) at 324°C to 8.9 dpa. TiC precipitates are also visible in all conditions. Enlarged views of the central sections are placed on the right side to show finer detail of typical cavities.

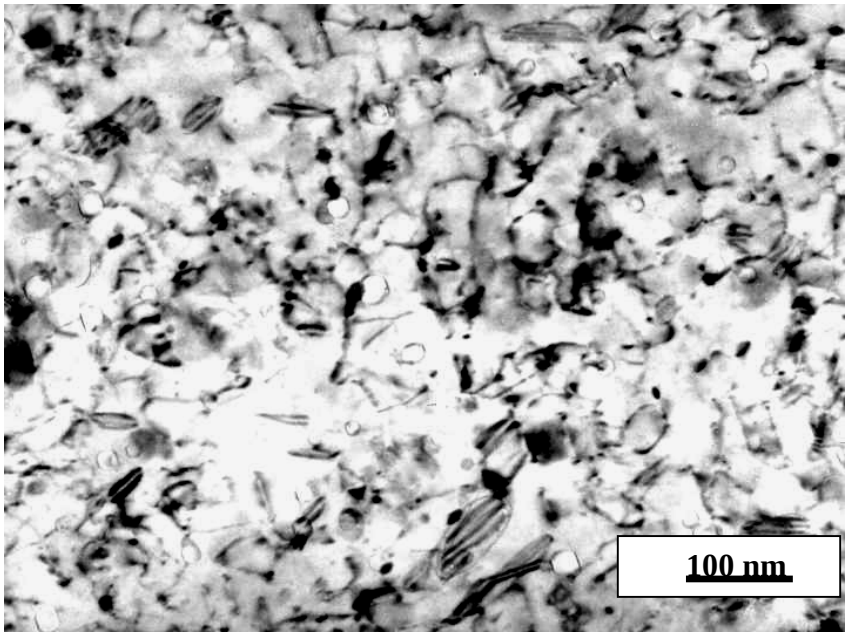

Figure 5. Dislocation structure of 12X18H9T steel after neutron irradiation at 329°C to 19.6 dpa (cross section at +590 mm). Some voids are also visible.

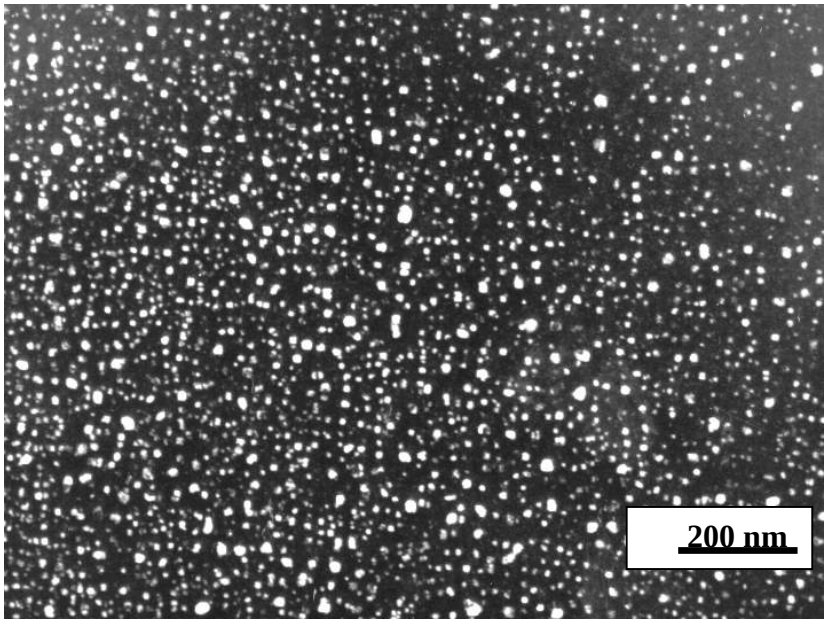

Figure 6. Dark-field image of G-phase precipitates in the 12X18H9T steel after neutron irradiation: at 302°C to 27.6 dpa (cross section at 0 mm).

Table 3. Microstructural data on G-phase

Temperature, °C	Dose, dpa	Mean diameter, nm	Concentration, 10^{16} cm^{-3}	Volume fraction, %
289	30.5	5.5	3.0	0.5
299	23.4	8.0	2.0	0.9
302	27.6	5.0	11.0	0.3
308	24.8	8.0	1.0	0.4
311	10.3	11.0	1.0	1.0
313	10.3	12.0	0.7	0.7
319	9.4	11.0	0.3	0.3
327	9.5	15.0	0.6	0.6

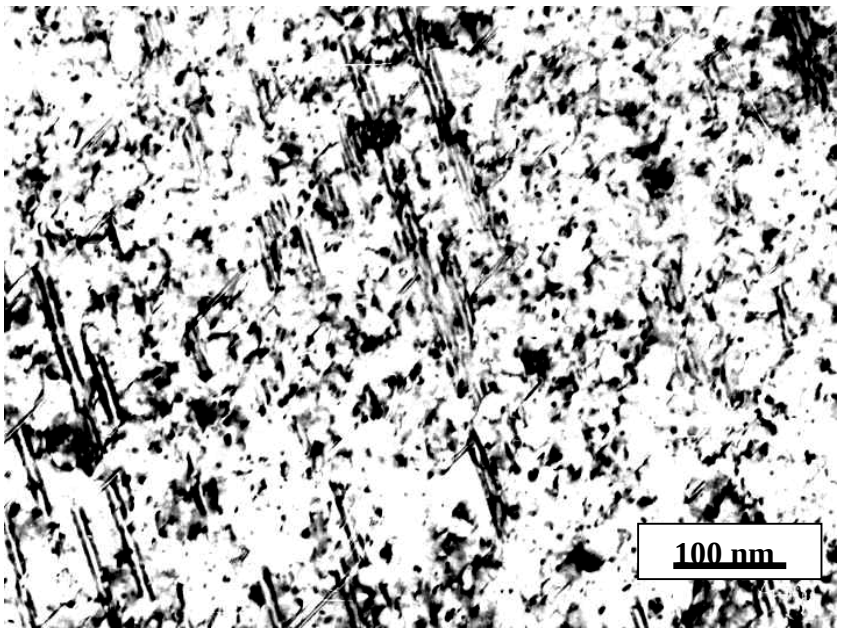

Figure 7. Bright-field image of stacking faults in the 12X18H9T steel after neutron irradiation at 314°C to 54.5 dpa (cross section at 0 mm).

data sets are very consistent, especially in extending the ~305°C swelling/no swelling boundary to lower dpa levels. Once again it appears that 305°C is the lower limit of void formation for this steel at doses of 10 dpa or greater, and that voids can form at doses perhaps as low as 7 dpa at temperatures above the 305°C boundary.

Garner and co-workers have also published swelling data on various Russian stainless steels from both fast reactors and water-moderated reactors operating at PWR-relevant dpa rates [3,6-12]. Many of these data also support a low temperature boundary in the vicinity just above 300°C.

Neustroev also has published data on the X18H9T steel after irradiation at 1.3x 10^{-7} dpa/s in the reflector region of the BOR-60 fast reactor [8]. At ~345°C, 1.4% swelling was reached at

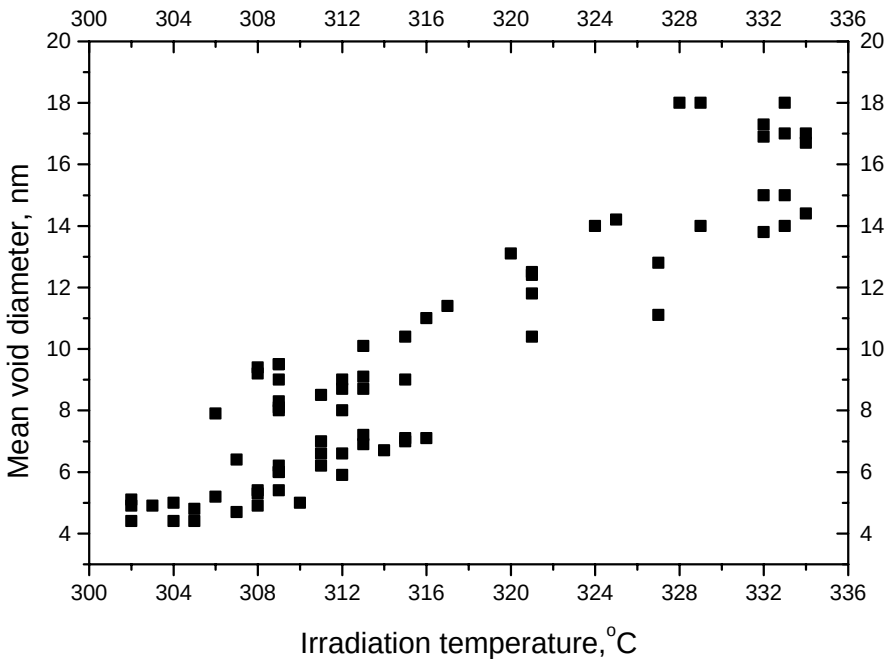

Figure 8. Temperature dependence of the mean void diameter.

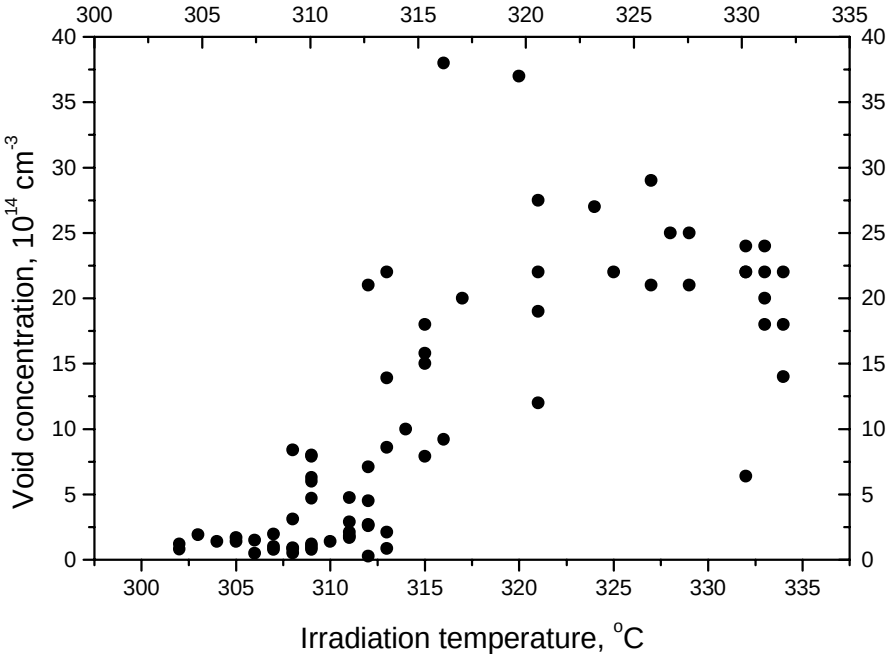

Figure 9. The temperature dependence of the void concentration.

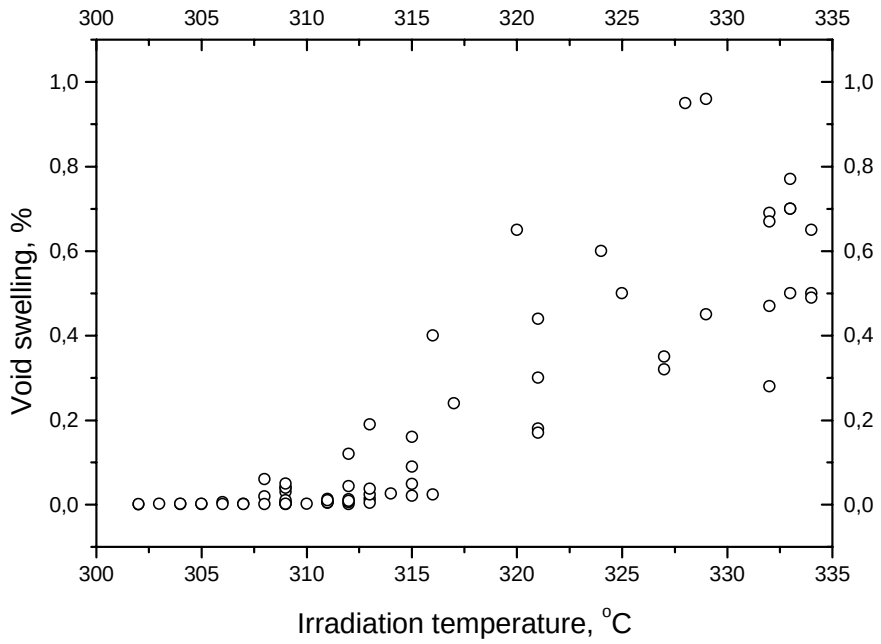

Figure 10. The temperature dependence of void swelling.

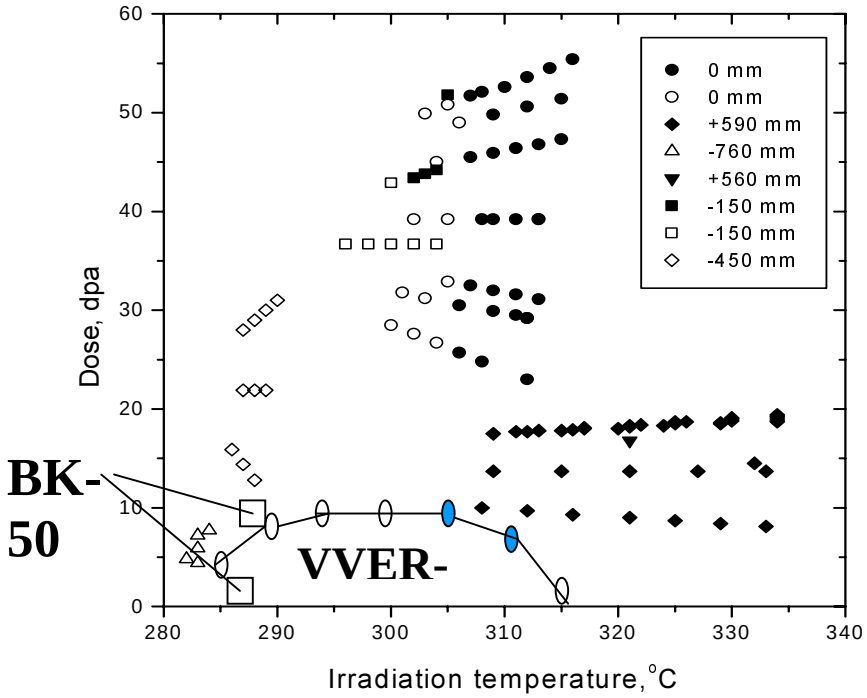

Figure 11. Comparison of results by Neustroev et al. (14) and those of this study. BK-50 is a BWR located in Dimitrovgrad Russia, and the VVWR-1000 is a Russian PWR in Rovenskaya, Ukraine.

only 38 dpa, with a significantly smaller incubation dose than observed in comparable irradiations conducted at higher dpa rates. As shown by Edwards et al., voids are already forming in some PWR baffle bolts made with cold-worked 316 stainless steel irradiated to only 8-12 dpa at temperatures in the range 330-345°C [15]. This steel is normally considered to be rather swelling resistant, especially when compared to annealed 304 stainless steel.

Conclusions

On the basis of the present microstructural study it is possible to make the following conclusions.

1. Voids in type 12X18H9T austenitic steel irradiated in the BN-350 fast reactor at dose rates ranging from 1.1×10^{-8} to 1.6×10^{-7} dpa/s are observed in the temperature range of 305-334°C at doses ranging from 8 to 56 dpa. At irradiation temperatures below ~305°C voids resolvable in an electron microscope were not found.
2. The ~305°C lower boundary of swelling appears to be relatively insensitive to dpa rate in the range examined, and also insensitive to the differences in neutron spectra found between fast reactors and water-moderated reactors.
3. The ~305°C boundary also appears to be consistent with observations on other Russian steels, providing some confidence in predicting that such a boundary may also apply to the AISI 304 and 316 steels employed in Western and Japanese PWRs.
4. It is anticipated that the above conclusions are also applicable to fusion spectra irradiation of 300 series stainless steels.

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REFERENCES

[1] C. Cawthorne and J. E. Fulton, Nature 216 (1967) 515-517.

[2] F. A. Garner, Chapter 6: "Irradiation Performance of Cladding and Structural Steels in Liquid Metal Reactors", Vol. 10A of Materials Science and Technology: A Comprehensive Treatment, VCH Publishers, 1994, pp. 419-543.

[3] F. A. Garner and M. B. Toloczko, "Irradiation Creep and Void Swelling of Austenitic Stainless Steels at Low Displacement Rates in Light Water Energy Systems", J. of Nucl. Mater., 251 (1997) 252-261.

[4] F. A. Garner, L. R. Greenwood and D. L. Harrod, "Potential High Fluence Response of Pressure Vessel Internals Constructed from Austenitic Stainless Steels", Proc. Sixth Intern. Symp. on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, San Diego, CA. August 1-5, 1993, pp. 783-790.

[5] F. A. Garner, "Materials Issues Involving Austenitic Pressure Vessel Internals Arising From Void Swelling and Irradiation Creep", Trans. Am. Nucl. Soc., 71 (1994) 190.

[6] F. A. Garner, M. B. Toloczko, S. I. Porollo, A. N. Vorobjev, A. M. Dvoriashin and Yu. V. Konobeev, “Potential Irradiation Creep and Void Swelling of Stainless Steels in PWR Internals”, Proceedings Eighth International Symposium On Environmental Degradation of Materials in Nuclear Power Systems-Water-Reactors, Amelia Island, Florida, Aug 10-14, 1997, pp. 839-845.

[7] S. I. Porollo, A. N. Vorobjev, Yu. V. Konobeev, A. M. Dvoriashin, V. M. Krigan, N. I. Budylkin, E. G. Mironova and F. A. Garner, “Void-Induced Embrittlement of Austenitic Stainless Steel Irradiated to 73-82 dpa at 335-360°C”, J. of Nucl. Mater., 258-263 (1998), pp. 1613-1617.

[8] V. S. Neustroev, V. K. Shamardin, Z. E. Ostrovsky, A. M. Pecherin, and F. A. Garner, "Temperature-Shift of Void Swelling Observed at PWR-Relevant Temperatures in Annealed Fe-18Cr-10Ni-Ti Stainless Steel Irradiated in the Reflector Region of BOR-60", Effects of Radiation on Materials: 19th International Symposium, ASTM STP 1366, M. L. Hamilton, A. S. Kumar, S. T. Rosinski and M. L. Grossbeck, Eds., American Society for Testing and Materials, 2000, pp. 792-800.

[9] F. A. Garner, S. I. Porollo, A. N. Vorobjev, Yu. V. Konobeev, A. M. Dvoriashin, V. M. Krigan, N. I. Budylkin, and E. G. Mironova, "Void-Induced Swelling and Embrittlement in the Russian Equivalent of AISI 316 Stainless Steel at PWR-Relevant End-of-Life Conditions", International Symposium on "Contribution of Materials Investigation to the Resolution of Problems Encountered in Pressurized Water Reactors, 14-18 Sept. 1998, Fontevraud, France, pp. 249-260.

[10] S. I. Porollo, A. M. Dvoriashin, A. N. Vorobjev, V. M. Krigan, Yu. V. Konobeev, F. A. Garner, N. I. Budylkin and E. G. Mironova, Neutron-Induced Swelling and Embrittlement Behavior of Two Russian Stainless Steels at PWR-Relevant Temperatures and 65-85 dpa”, 9th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, 1999, pp. 1061-1068

[11] G. M. Bond, B. H. Sencer, F. A. Garner, M. L. Hamilton, T. R. Allen and D. L. Porter, “Void Swelling of Annealed 304 Stainless Steel at ~370-385°C and PWR-Relevant Displacement Rates”, 9th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, 1999, pp. 1045-1050.

[12] F. A. Garner, S. I. Porollo, A. N. Vorobjev, Yu. V. Konobeev and A. M. Dvoriashin, “Void Swelling at Low Displacement Rates in Annealed X18H10T Stainless Steel at 30-56 dpa and 280-332°C”, 9th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, 1999,pp. 1051-1059.

[13] M. L. Hamilton, F. H. Huang, W. J. S. Yang and F. A. Garner, "Mechanical Properties and Fracture Behavior of 20% Cold-Worked 316 Stainless Steel Irradiated to Very High Exposures", Effects of Radiation on Materials: Thirteenth International Symposium (Part II) Influence of Radiation on Material Properties,

ASTM STP 956, F. A. Garner, N. Igata and C. H. Henager, Jr., Eds., ASTM Philadelphia, PA, 1987,pp. 245-270.

[14] V. S. Neustroev V. N. Golovanov, Z. Ye. Ostrovsky, A. M. Pecherin, and V. K. Shamardin, “Radiation Effects in 18Cr-10Ni-Ti Steel Irradiated in Different Reactors under Conditions Close to VKU VVER Reactors Operating Conditions”, Proceedings of the Fourth International Ural Seminar on Radiation Damage Physics of Metals and Alloys, Feb. 25 – Mar. 3, 2001. Snezhinsk, Russia, in press.

[15] D. J. Edwards, F. A. Garner, B. A. Oliver and S. M. Bruemmer, “Microstructural Evaluation of a Cold-Worked 316SS Baffle Bolt Irradiated in a Commercial PWR”, Proc. of Tenth Intern. Conf. on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, Aug. 5-9, 2001, in press.