Examination of ChS-68 Steel Used as a BN-600 Reactor Cladding Material

N.V. Glushkova[1], A.V. Kozlov[1], E.A. Kinev[1], V.L. Panchenko[1], I.A. Portnykh[1], S.V. Barsanova[1], V.A. Tsygvintsev[1], Yu.P. Budanov[2], A.V. Tselishchev[2], E.V. Spitsyn[2]

[1] Joint Stock Company "Institute of Nuclear Materials" (INM), Zarechny, Russia

[2] Stock Company "A.A. Bochvar High-technology Research Institute of Inorganic Materials" (VNIINM), Moscow, Russia

E-mail contact of main author: Natalya1972@inbox.ru, xray@irmatom.ru

Abstract. Austenitic stainless steels, used as a material for fast reactor claddings, are more heat-resistant than ferritic-martensitic steels. At the same time austenitic steels are susceptible to corrosion, radiation-induced swelling, creep, embrittlement, and strength reduction. Swelling is the main problem. Optimization of ChS-68 properties – a standard material for BN-600 fast reactor claddings – has been carried out by High-technology Research Institute of Inorganic Materials in cooperation with Beloyarsk Nuclear Power Plant and Institute of Nuclear Materials and provided damage dose increase from 44 to 73 dpa by 2000, and up to 87 by 2015. ChS-68 properties optimization is still in process.

Key Words: austenitic steel, swelling.

1. Introduction

Fast reactors are capable of providing closed fuel cycle that enables effective use of natural raw materials and reduces need in rare ^{235}U. Fuel burn-up is one of the main parameters of effective reactor operation. The higher the burn-up, the higher is economic performance of fast reactors. Nowadays fuel burn-up increase in fast reactors is limited by capabilities of used cladding materials.

Austenitic stainless steel has been used as a standard material for BN-600 fast reactor claddings for many years. High-temperature strength is one of the advantages of austenitic steels over ferritic-martensitic ones. However, austenitic steels affected by neutron irradiation are exposed to radiation-induced swelling, embrittlement and strength reduction, and their corrosion damages are registered. In this respect at high burn-up levels swelling is the main problem limiting operation of the material.

Limit swelling value is 15 %. In case of greater value fuel elements increase in diameter, contact with each other, thereby reducing clearance, where coolant (liquid sodium) flows.

From 1980 to 1987 fuel burn-up in BN-600 reactor reached 7.2% fissions per initial metal atom (FIMA) at damage dose less than 50 displacements per atom (dpa) [1]. EI-847 austenitic steel was used as a cladding material [2].

Based on EI-847 steel new austenitic steels have been developed: EP-172 (boron-modified) and ChS-68, doped with boron, silicon and titanium [3]. ChS-68 is used as a standard material for BN-600 fast reactor claddings. Chemical composition of ChS-68 is given in *Tab*. Swelling constraining elements, introduced into new cladding steels, as well as sequential modernizations of BN-600 reactor core made it possible by 2000 to attain burn-up level of 9.2 % FIMA and damage dose level of 73 dpa per cladding [4].

TAB. Chemical composition (wt %) of the fuel element cladding made of ChS-68 steel

Cr	Ni	Mo	Mn	Si	C	B
15.5-17.0	14.0-15.5	1.9-2.5	1.2-2.0	0.3-0.6	0.05-0.08	0.002-0.005

At Institute of Nuclear Materials numerous material-testing examinations have been carried out to determine corrosion characteristics, physical and mechanical properties and swelling of claddings operated in BN-600 reactor. Irradiated in BN-600 claddings of standard and test fuel assemblies, as well as materials test assembly specimens (ChS-68 and other austenitic steels, similar in composition) have been examined at INM hot cells.

2. Swelling

In case of cladding manufacturing technology used till 2002 many fuel elements reached 15% swelling at maximum dose of ~70 dpa. However, swelling values for claddings irradiated in similar conditions differ at least by 2 [5, 6]. Such significant differences show a latent reserve of materials and need to improve cladding manufacturing technology.

ChS-68 properties optimization has been carried out by High-technology Research Institute of Inorganic Materials using the results of post irradiation examination made by Beloyarsk NPP and ROSATOM materials testing enterprises, including INM. During optimization steel chemical composition has been varied within specifications, at different cold work levels, according to different metal manufacturing and tube production technologies [3, 6, 7, 8].

The Machine-Building Plant has improved cladding manufacturing technology in 2006-2008. This provided increase of resistance to swelling and greater cladding uniformity.

Swelling, determined by hydrostatic weighing, depends on both damage dose and irradiation temperature. As it can be seen in *Fig. 1* irradiation temperature and damage dose vary along the reactor core.

To compare results of post irradiation examinations in a proper way the data is divided into groups correlating to irradiation temperature intervals (10 °C step size), in which temperature effect is considered to be the same. Dose dependence of cladding swelling in different temperature intervals is given in *Fig. 2-4*. Filled marker points indicate data for claddings manufactured before 2008 (the first phase of manufacturing technology improvement). Unfilled marker points indicate data for claddings manufactured after the technology had been improved (the second phase of manufacturing technology improvement). Solid lines are obtained with empirical model [9] based on the first manufacturing technology.

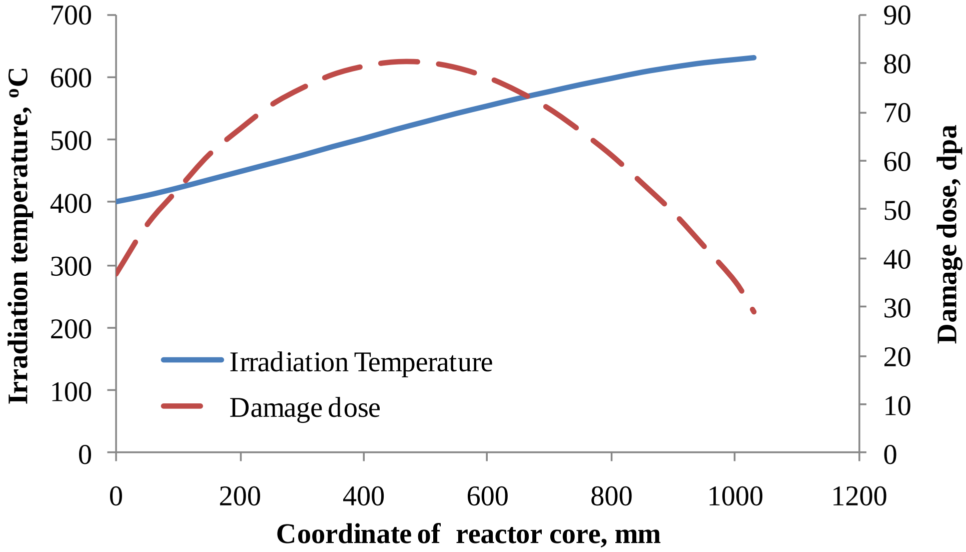


FIG. 1. Temperature and damage dose variation in reactor core.

The highest swelling rate is registered in 420-429 °C temperature interval (*see Fig. 2*). However, damage dose in this cladding area is beyond 55 dpa, therefore maximum swelling is less than 4 %. With irradiation temperature increase to 460 °C swelling dose rate decreases, whereas, due to greater dose values in this area, swelling values are greater as well (*see Fig. 2*).

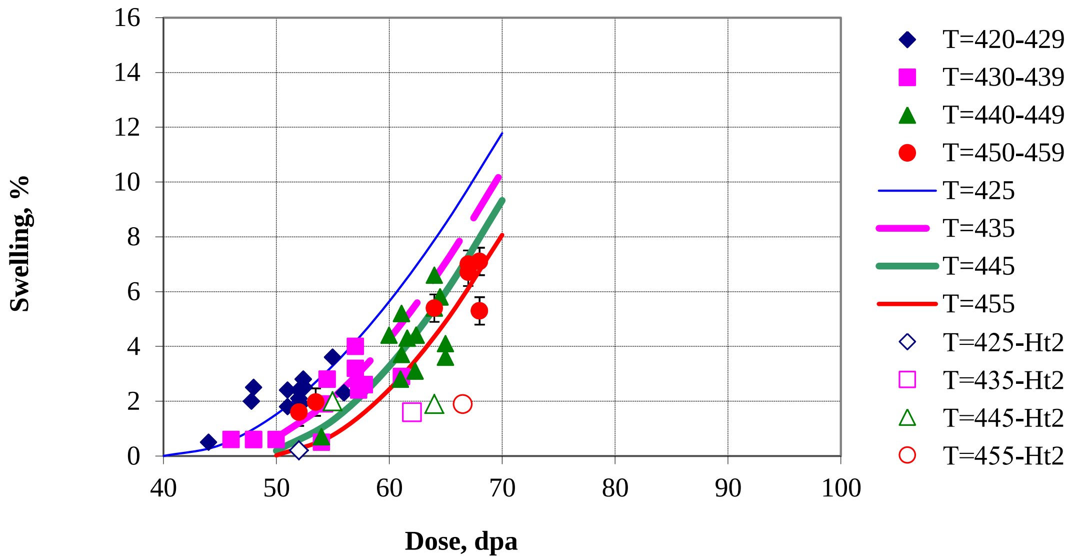


FIG. 2. Dose dependence of swelling variation in 420-460 ℃ temperature interval: points indicate experimental results, lines indicate empirical dependence.

There is a big spread of data in cladding areas, irradiated at temperatures of 460–500 °C and up to 80 dpa (*see Fig. 3*), therefore swelling increases sharply.

Irradiation to 80-85 dpa at temperatures of 510-560 °C caused less swelling than in areas with lower temperatures and smaller doses (*see Fig. 4*).

Significantly less swelling is registered in all temperature intervals for claddings manufactured according to the improved technology (see Fig. 2, 3, 4, unfilled marker points). These claddings are more resistant to swelling and have low dose swelling rate.

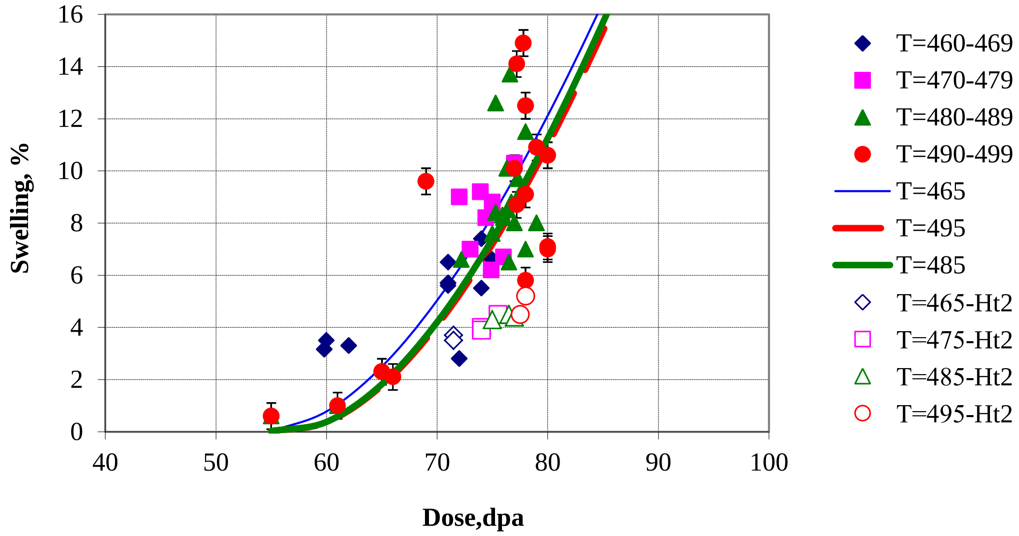


FIG. 3. Dose dependence of swelling variation in 460-500 °C temperature interval: points indicate experimental results, lines indicate empirical dependence.

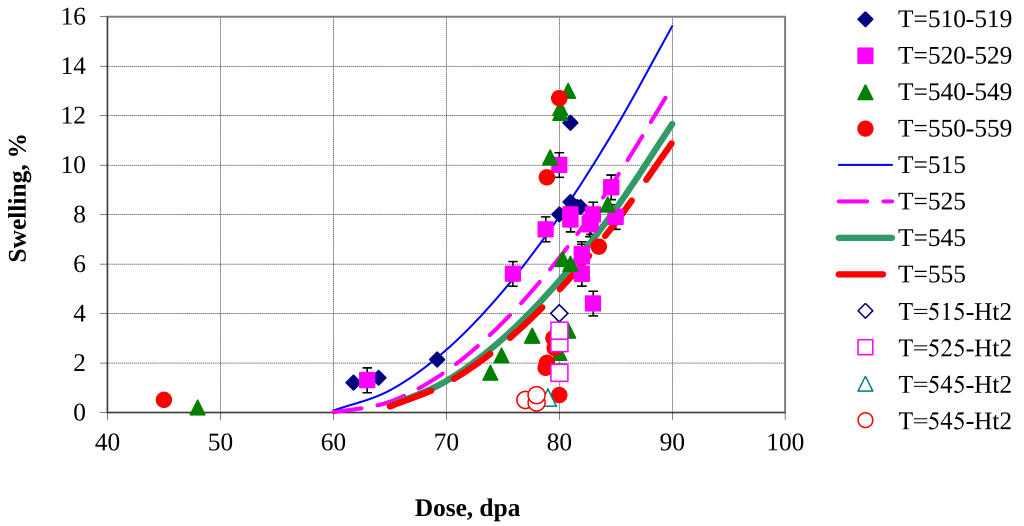


FIG. 4. Dose dependence of swelling variation in 510-560 °C temperature interval: points indicate experimental results, lines indicate empirical dependence.

3. Mechanical Properties

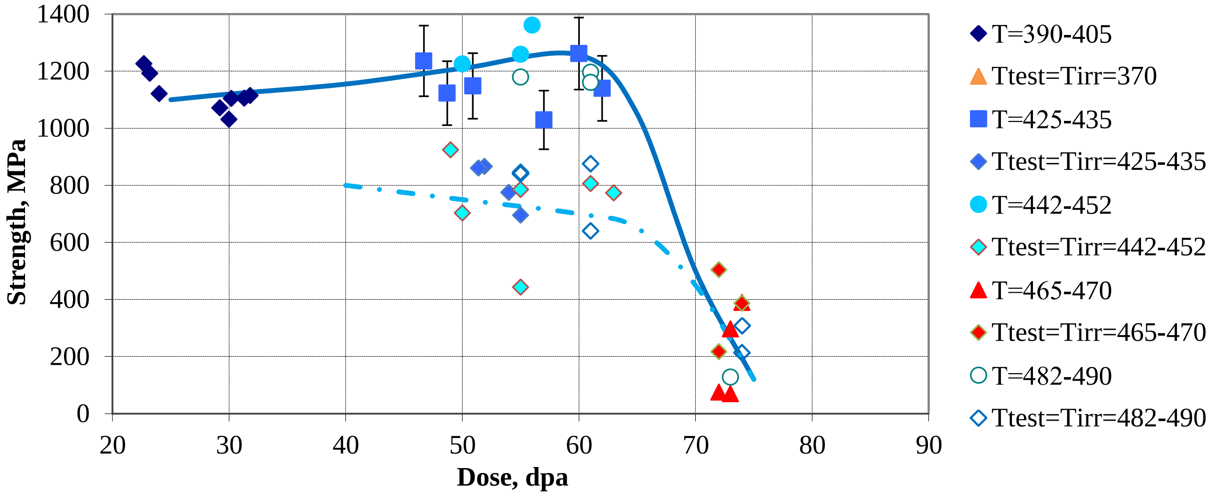
Mechanical characteristics are measured with two techniques: uniaxial tension of ring specimens and testing of tubular specimens with internal aggregate pressure [10, 11].

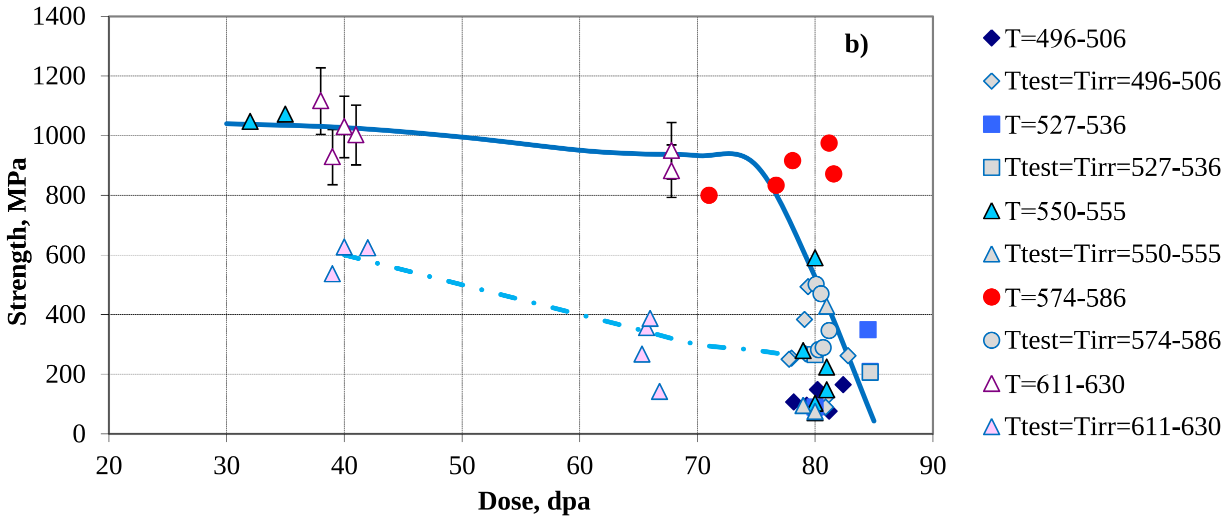
Dose dependence of tensile strength variation for ring and tubular specimens at test temperature of 20 °C is given in *Fig. 5*. The given data describes characteristics for the material of claddings manufactured according to the first technology.

Ring specimens irradiated at temperatures of 390–490 °C show high strength properties up to 64 dpa (*see Fig. 5a*). Dose increase in this temperature interval causes sharp (3-6 times) strength reduction.

At irradiation temperatures of 500–630 °C material from areas irradiated up to ~80 dpa is characterized with high strength. A big spread of strength values 50-1000 MPa is registered at doses of 78-92 dpa.

Uniaxial tension of ring specimens is a conservative technique, which does not fully reflect mechanical properties of the material. Therefore at INM a testing technique for tubular specimens, made of irradiated claddings, with internal pressure of plastic aggregate has been designed and certified. Strength characteristics of the irradiated ChS-68 steel, obtained at room temperature and at irradiation temperature of specimen testing (*see Fig. 5c*), remain high within the whole operation temperature range. It can be seen (*see Fig. 5)* that strength characteristics, obtained on ring specimens at room temperature are higher than at the irradiation temperature up to doses of 70 dpa; on the tubular samples – from 55 to 65 dpa.





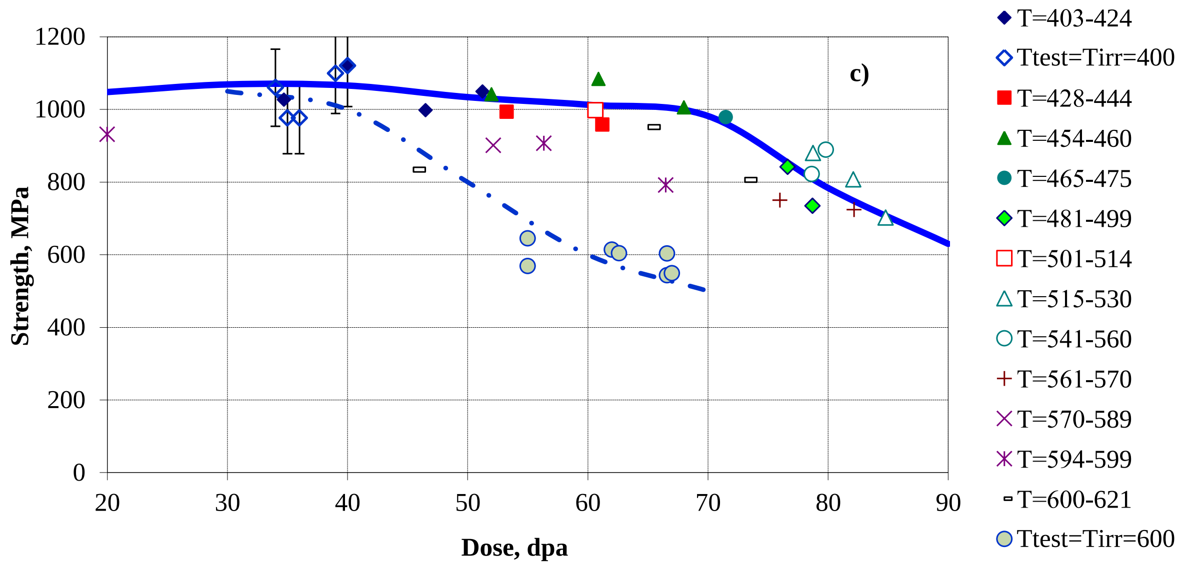


FIG. 5. Dose dependence of tensile strength variation: a, b) ring specimens, c) tubular specimens. Solid line – test temperature is 20 °C; dotted line – test temperature is irradiation temperature. Points indicate experimental results, line indicates smoothed dependence.

The same is true for elongation (*see Fig. 6*). In 390–490 °C temperature interval plasticity of ring specimens is on average twice reduced with dose increase from 20 to 60 dpa (*see Fig. 6a*). At doses over 70 dpa plasticity value is 0%.

With irradiation temperature increase elongation increases 3-4%, ring specimens, irradiated at doses over 78 dpa, lose plasticity features (*see Fig. 6b*).

Tubular specimen tests show elongation significantly exceeding ring specimen characteristics within the whole irradiation temperature and dose range (*see Fig. 6c*). Tubular specimens show some plasticity reserve at damage doses over 78 dpa (*see Fig. 6c*).

Swelling dependence of tensile strength and elongation is based on the results of testing with internal pressure of tough aggregate (*see Fig. 7*). Apparently claddings made of ChS-68 austenitic steel have high strength and enough residual plasticity.

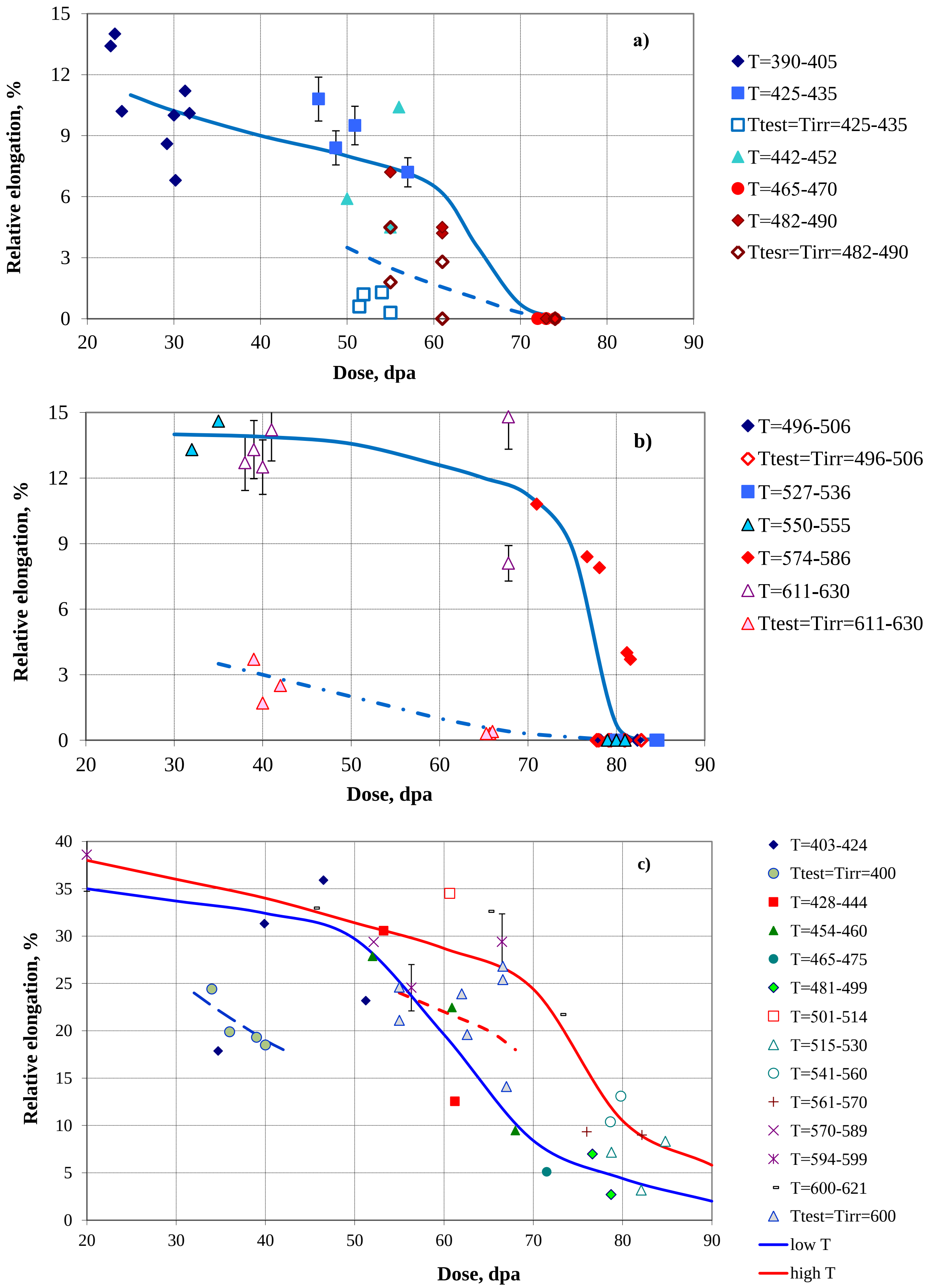


FIG. 6. Dose dependence of elongation variation: a), b) ring specimens, c) tubular specimens. Solid line – test temperature is 20 °C; dotted line – test temperature is irradiation temperature. Points indicate experimental results, line indicates smoothed dependence.

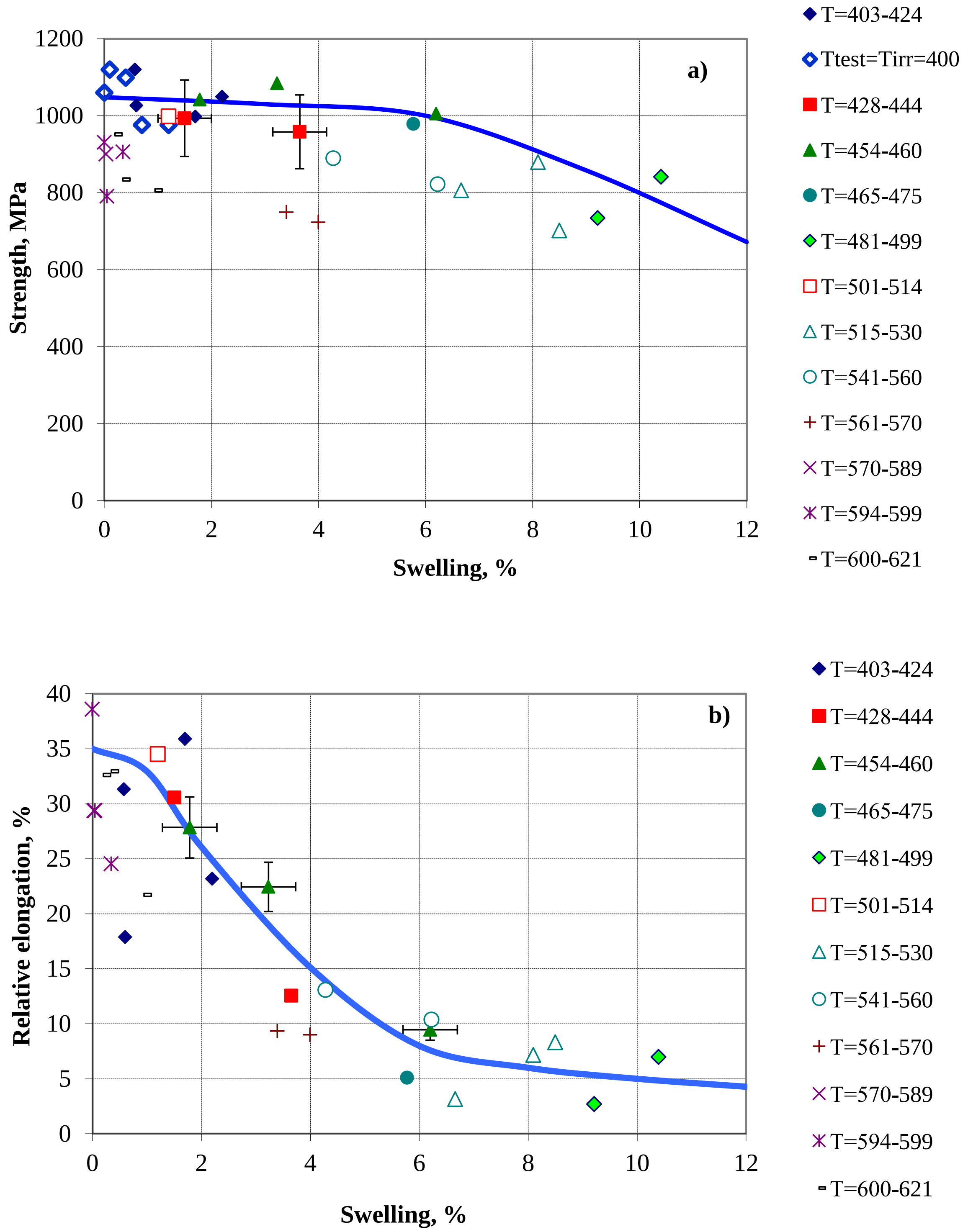


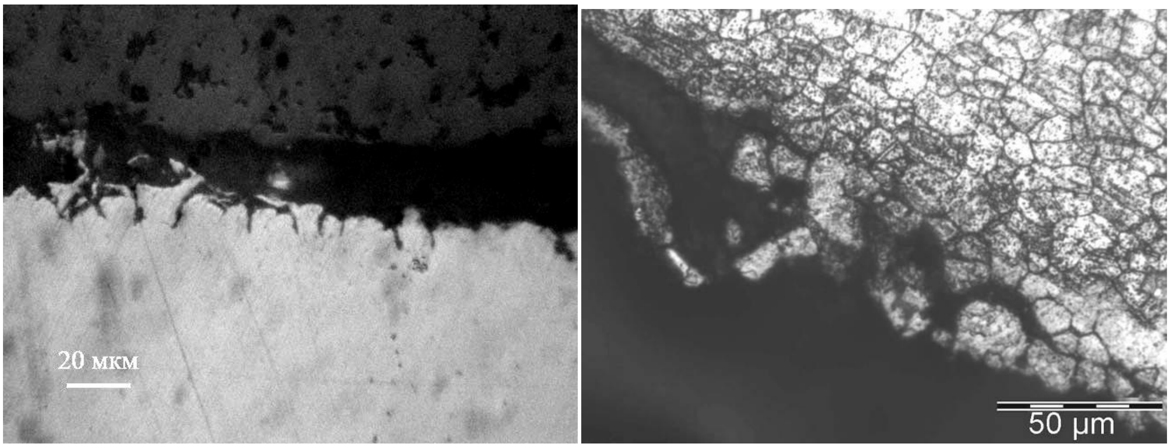
FIG. 7. Swelling dependence of tensile strength and elongation variations. Test temperature is 20 °C.

4. Corrosion damages

Corrosion damages of claddings induce localized thinning of the cladding and may cause generation of stress raisers, which due to brittle cladding is depressurization hazard. ChS-68 is corrosion resistant austenitic steel. Corrosion damages registered during active BN-600 reactor operation are not critical.

As a rule minimum intergranular damages of claddings, observed in maximum swelling area from fuel composition side, are beyond 10 μm (*see Fig. 8a*).

As irradiation temperature increases (in the core top area), so does the corrosion depth. Internal surface of claddings in the core top area, where irradiation temperature is above 590 °C, is exposed to maximum damages up to 50 μm (*see Fig. 8b*).



a) b)

FIG. 8. Corrosion of cladding internal surface: a) irradiation temperature is 490 °C, b) irradiation temperature is 610 °C.

Corrosion from the side of liquid-metal coolant is typical for high-temperature areas. However, depth of external damages does not exceed 10 μm (*see Fig. 9*).

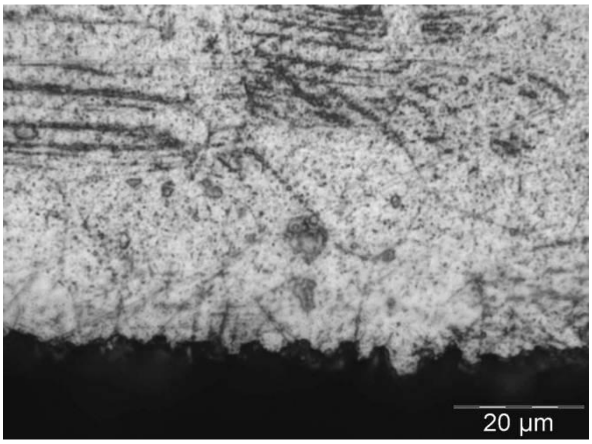


FIG. 9. Corrosion of cladding external surface. Irradiation temperature is 610°C.

5. Conclusion

Dependence of the main material characteristics on BN-600 operation conditions is based on long-term examinations carried out at INM. Examination results have been used to improve cladding manufacturing technology. The results of swelling, strength and plasticity characteristics, and surface corrosion investigation show that claddings made of advanced ChS-68 steel are capable to provide safety life to damage dose of 92...95 dpa.

6. References

[1] CHUEV, V.V., ROSLYAKOV, V.V., MALTSEV, V.V. Specific aspects of structural materials in neutron spectrum of high-power fast reacor. Izvestia visshikh uchebnikh zavedeniy, Yadernaya energetika, No. 1, 2005, 113-126 (in Russian).

[2] NOSOV, Yu.V. Solution of scientific and technical issue of fast reactor operation - from BN-600 to BN-800. The Ninth International Scientific and Technical Conference "Safety, Efficiency and Economics of Nuclear Power Industry", Proceedings http://mntk.rosenergoatom.ru/archiv/2014/plenary-reports-2014.

[3] AGEEV, V.S., BUDANOV, Yu.P., YOLTUKHOVSKI, A.G. et al. Structural materials for cores of Russian fast reactors. State and prospects. Izvestia visshikh uchebnikh zavedeniy, Yadernaya energetika, No. 2, 2009, 210-218 (in Russian).

[4] V. V. CHUEV, A. N. OGORODOV, V.V. MALTSEV. Basic Results of Forced Tests of Core Structural Materials in Fast BN-600 Reactor. The Fifth International Ural Seminar. Abstracts. Snezhinsk, 2003, p 68.

[5] PORTNYKH, I.A., GLUSHKOVA, N.V., PANCHENKO, V.L., KINEV, E.A., KOZLOV, A.V. Difference in radiation-induced swelling of ChS-68 claddings occurring in similar neutron irradiation conditions. Problems of Atomic Science and Technology, No. 1 (68-69), 2007, 377-388 (in Russian).

[6] MOSIN, A.M., EVSEEV, M.V., PORTNYKH, I.A., SHCHERBAKOV, E.N. et al. Changes in physical and mechanical properties of EK-164 and ChS-68 claddings after operation in BN-600 reactor for four cycles. Izvestia visshikh uchebnikh zavedeniy, Yadernaya energetika, No. 1, 2001, 224-229 (in Russian).

[7] MITROFANOVA, N.M., TSELISHCHEV, A.V., AGEEV, V.S., BUDANOV Yu.P. et al. Structural materials for for BN-600 fast reactor claddings and shrouds. Izvestia visshikh uchebnikh zavedeniy, Yadernaya energetika, No. 1, 2011, 211-223 (in Russian).

[8] AGEEV V.S., TSELISHCHEV A.V., SHKABURA I.A., BUDANOV YU.P., LEONTYEVA-SMIRNOVA M.V., MITROFANOVA N.M., IOLTUHOVSKIY A.G. Structural Materials for Russian Fast Reactors Cores. Status and Prospects. International Conference on Fast Reactors and Related Fuel Cycles – Challenges and Opportunities, Kyoto, Japan, December 7-11, 2009. http://www-pub.iaea.org/MTCD/Meetings/PDFplus/2009/cn176/cn176_Presentations/parallel_session_4.2/04-09.Ageev.pdf

[9] KOZLOV, A.V., KINEV, E.A., BRYSHKOVA, S.V., PORTNYKH, I.A. The swelling dependence of cold worked 16Cr-15Ni-2Mo-1Mn steel on neutron irradiation temperature, fluence and damage rate during its use as a cladding material in BN-600 reactor. // Effect of Radiation on Materials: 20th International Symposium, ASTM STP 1405, S.T. Rosinsky, M.L. Grossback, T.R. Allen, and A.S. Kumar, Eds., American Society for Testing and Materials, West Conshohocken, PA, 2001, 457- 468.

[10] KOZLOV, A.V., AVERIN, S.A., BRYUSHKOVA, S.V., KALASHNIK, G.V. A Problem to Determine Short Term Mechanical Properties Changees of Ferrite-Martensite and Austenitic Steels as Material of Fuel Assembly of Fast Reactor under High Doze Neutron Irradiation // Proceeding of a Technical Committee meeting held in Obninsk, Russian Federation, 16-19 June 1997. IAEA-TECDOC-1039, August, 1998, 211-222.

[11] KOZLOV, A.V., BRYUSHKOVA, S.V., KOVALEV, I.N., KUZNETSOV, Yu.G. Facility for tubular specimen testing // Utility model certificate No. 26129 in the National register of utility models in the Russian Federation, Moscow, 10 November, 2000 (in Russian).