Examination of Fast Reactor Materials and Structural Elements at JSC “INM” Premises

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Abstract. The directions of structural element examinations at JSC "INM" after operation in BN-600 reactor are given. The examinations aimed to predict structural materials properties and identify the causes of variation. The examples of the obtained results, applied to predict the behaviour of in-vessel facilities and based on developed theoretical concepts on the mechanism of radiation-induced changes in materials under irradiation in the reactor, are given.

Key words: fast reactors, fuel elements and shroud tubes, reactor vessel and in-vessel facilities, service life extension.

1. Introduction

At the Institute of Nuclear Materials post irradiation examination has been carried out since BN-600 reactor commissioning to justify safety and reliability of different core elements during routine operation and to search for new ways of extending their service life. The examination is carried out in close cooperation with the design bureau (Afrikantov OKBM), nuclear operator (Beloyarsk NPP), materials testing enterprise (VNIINM), fuel assembly manufacturer (MSZ) and other enterprises. It helps to use post irradiation examination results promptly to modify reactor structural elements and units, and improve economic efficiency as well. The main aspects of the examination are as follows:

- examination of fuel elements and shroud tubes of standard, trial and test fuel assemblies;
- examination of reactor control and safety units (control rods including absorber elements and shroud tubes);
- examination of materials science assembly specimens irradiated in BN-600 reactor;
- examinations to justify possible service life extension from 30 to 45, and then to 60 years.

INM carries out unique examination because it is not limited with statement of fact of changes in structural elements and their material properties. The examination is aimed to predict their further behaviour and identify the causes of the changes. It is not sufficient to carry out separate post irradiation examinations, there should be a systematic result set based upon theoretical concepts on the process mechanisms. Descriptive modeling of structural evolution processes and corresponding variations in physical and mechanical properties is conducted. It is also necessary to improve existing examination techniques and develop new ones to obtain characteristics used to predict residual and limited life for core elements and the reactor as a whole.

The paper aims to show the main results of BN-600 structural element examination at INM, demonstrate their application, and make a review on developed theoretical concepts and development of techniques used for the examination of fast reactor materials.

The key aspects concern extension of fuel assembly and BN-600 reactor service life.

2. Justification of Service Life Extension for Fast Reactor Fuel Assembles

At the initial stage of BN-600 reactor operation maximum fuel burn-up was 5-7 % FIMA. Post irradiation examination of fuel elements shows that the key factors impeding fuel burn-up to increase are radiation-induced swelling of cladding, its embrittlement and strength reduction, as well as corrosion damage of cladding from the fuel side. A set of steps to increase maximum fuel burn-up has been developed on the basis of the obtained results of post irradiation examinations. The design bureau (Afrikantov OKBM) has conducted several BN-600 core modernizations designed to decrease linear load and reduce neutron flux on cladding material. The responsible materials testing enterprise (VNIINM) sought for the ways to increase radiation resistance of cladding by modernization of the used materials and its replacement by more radiation resistant ones. In this regard it was necessary to obtain the dependence of changes in materials properties on operating characteristics: neutron flux, temperature, fuel element environment, service life. The enterprises equipped with facilities providing examination of materials with high induced activity were looking for a solution (IPPE, RIAR, INM). Beloyarsk NPP and INM are in close vicinity, which facilitates operated fuel assembly transportation to the examination site. Therefore the main results concerning the effect of irradiation in BN-600 reactor on fuel and structural materials of fuel assemblies have been obtained at INM.

To justify safety of new materials as cladding and hexagonal shroud tubes for fuel assemblies in BN-600 reactor irradiation of different steel specimens and alloys is conducted in BN-600 materials science assemblies [1]. After irradiation to high damage doses they are sent to INM for examination. The effect of materials composition and cladding manufacturing conditions on their radiation resistance is experimentally defined.

Examinations at INM show that the concept of high swelling resistance of austenitic steels, rich in nickel, under irradiation to high doses at temperatures in the range between 400 and 600 °C is incorrect [2]. Swelling values for 32 and 35 % Ni steel specimens irradiated in materials science assembly of the central BN-600 core channel are 1.5 times higher than those for ChS-68 steel specimens irradiated under the same conditions.

As another result of the experiment optimal values for cold work at the last limit of cladding manufacturing are determined. It is done during cladding manufacturing from austenitic steels to increase resistance to radiation-induced swelling. The determined optimal value is estimated to be ~20 %. Previously it was supposed that a cold work value is in direct proportion to resistance to radiation-induced swelling. Swelling measurements of cladding specimens of the same materials science assembly after 10, 20 and 30 % cold work show the lowest swelling of 20%-cold-worked specimens. 30%-cold-worked specimens show higher swelling values than 10%-cold-worked ones [1].

Using the results of systematic studies conducted at INM for more than 30 years a sequential increase of standard fuel burn-up to 9...11.2...13 % FIMA and corresponding damage dose increase to 70...82...87 dpa took place. EI-844, EI-847, EP-172, and ChS-68 steels (*see Table I*) have been examined as a cladding material. Right now a changeover to a new cladding material (EK-164 steel) is in process. Trial operation of the material is carried out to its life limit of 100 dpa. Rosenergoatom Concern has developed and implemented a program on increasing the fuel assembly service life using mixed uranium-plutonium dioxide fuel and fast reactor claddings, made of EK-164 steel to fuel burn-up of 14-15 % FIMA and damage doses over 110 dpa. To increase burn-up to 18-20 % FIMA it is planned to use new ferritic-martensitic steels, which are under development now, including oxide dispersion strengthened steels and other candidate materials as well.

Along with selection of an optimal cladding material the material for hexagonal shroud tubes of fuel assemblies is tested. Tests of several materials; one of them at initial operating stages was a high-nickel EP-150 steel, are carried out. It is found that shroud tubes made of this steel are heavily susceptible to swelling and embrittlement. Ferritic-martensitic steels resistant to radiation-induced swelling appear to be the most suitable material for BN reactor shroud tubes. Nowadays shroud tubes made of EP-450 steel, which does not limit fuel assembly service life, are used in BN-600 and BN-800 reactors [3, 4].

Table I. Basic composition of the examined alloys (Fe-based) [5, 6]

Alloy	Element, wt. %								
	C	Cr	Ni	Mo	Mn	Si	Ti	P	B
EI-844	0.08	15.2	11.4	2.1	1.2	0.78	0.32	0.02	-
EI-847	0.05	15.8	15.2	2.8	1.1	-	-	-	0.001
EP-172	0.07	15.2	14.7	2.7	0.3	0.12	-	0.01	0.005
ChS-68	0.07	16.5	15.0	2.3	1.7	0.36	0.36	-	0.003
EK-164	0.07	15.9	19.4	2.3	1.6	0.41	0.30	0.02	0.004
EP-150	0.08	16.2	36.0	2.2	1.4	0.62	0.81	>0.02	0.002
EP-450	0.13	13.3	0.2	1.5	0.3	0.01	-	>0.02	0.004

3. Justification of Service Life Extension for BN-600 Reactor

In 2010 it was 30 years since BN-600 reactor has been launched marking its designed service life. Since that economic performance has been improved significantly. To extend reactor service life it was necessary to justify the survival of unchangeable in-vessel elements, including reactor vessel.

Several pieces irradiated to different damage doses for 22 years of operation were cut out of the *Surveillance package* – a thick-walled Cr18-Ni9 tube made of the same material as the reactor vessel. Specimens for all examinations to certify the state of unchangeable structural elements and make decision on reactor life extension are made from the cut pieces. According to the results of post irradiation examinations carried out at FSUE CRISM "Prometey" and INM [7] BN-600 reactor life has been extended to 45 years.

Similar examination of the *Surveillance package* elements, operated in the reactor for 35 years, has started. The results of this and other planned examinations of unchangeable elements, as well as the results of special simulation experiments on neutron irradiation in materials science assemblies will be used to justify BN-600reactor life extension to 60 years.

4. Development of Theoretical Concepts and Experimental Base for Examination of Fast Reactor Materials and Structural Elements at INM

The examinations carried out at INM are designed to predict service life of used materials, as well as to find new ways to extend this service life. This requires modeling of processes in materials under irradiation suitable for application, as well as procedural base improvement.

For the past 30 years a technology for obtaining experimental results has been developed. This technology is used to find out patterns of changes in structure, physical and mechanical properties, as well as cladding corrosion state characteristics. Their dependence on operating characteristics (neutron irradiation temperature and dose, rate of atom displacement generation for cladding environment) has been identified. Cladding specimens are cut out from the areas with different damage doses at similar temperatures. As maximum dose and core top temperature values are different for claddings of different fuel elements, we obtained the data to find out dose-temperature dependencies of property changes. Moreover, the claddings differ in terms of dose, defect generation rate, temperature in the core top. The results of non-destructive testing are used to define the specimen areas, ensuring that the areas with preestimated swelling values, maximum local corrosion damage, and cladding-fuel interaction peculiarities are chosen. Examination of each fuel element is not carried out separately but as a part of the set of experimental results.

New technologies have been developed. It is found that conventional tensile tests on ring cladding specimens give too conservative estimation of mechanical properties. During tensile tests of ring specimens zero total elongation is registered for specimens with swelling of 4 % and more with tensile strength decrease to 100 MPa and less. Life extension of the cladding with such properties by regulatory authorities involves serious difficulties. At the same time these claddings keep their operating capacity in further operation (to swelling of 15 % and more).

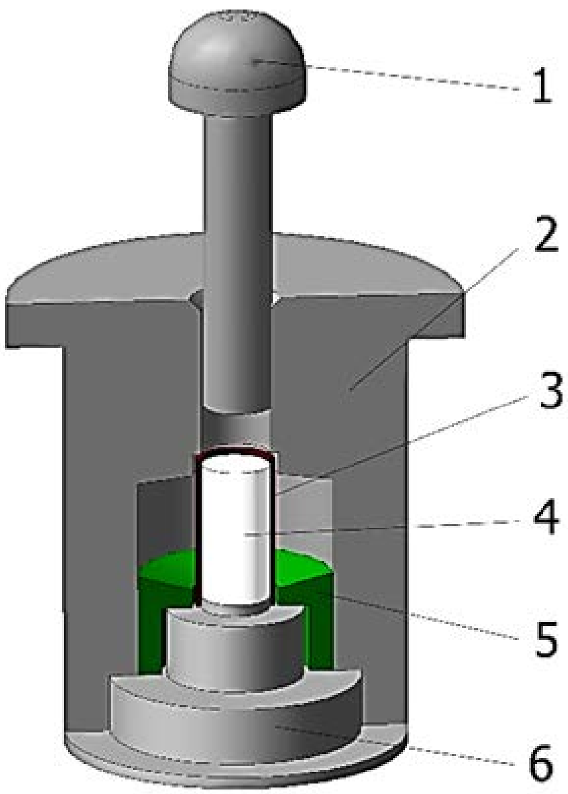
A technology for mechanical testing of tubular specimens with internal pressure of tough plastic aggregate has been developed at INM [8]. Cladding loading circuit for these tests is given in *Fig. 1*. In these test stress strain state better correlates with conditions of the effect on cladding under operation. Initial data processing (aggregate pressure loading – grip movement chart) helps to estimate stress in the cladding at the beginning of its plastic deformation and damage and pressure on the cladding from aggregate environment, as well as all characteristics at the beginning of the damage. Obtained values of mechanical properties can more adequately reflect cladding properties under operation and are used to justify life extension of fuel elements. *Fig. 2* shows relative elongations, obtained during tensile tests of ring specimens and internal pressure tests of tubular specimens.

Successful increase of cladding resistance to radiation-induced swelling makes cladding embrittlement, connected not only with swelling, more serious issue. At high testing temperatures (~600 °C, within operating temperature range) embrittlement, caused by other structural changes, appears. With due regard to the formation of intergranular corrosion cracks in the cladding it is necessary to determine limit state of fast reactor claddings. Therefore it is necessary to determine rupture strength characteristics of notched specimens. A technique of eccentric rupture test of notched ring specimens is under development at INM [9]. Test circuit is given in *Fig. 3*. Determination of rupture strength characteristics is complicated with examination performance, as well as with test diagram interpretation. Standard crack resistance characteristics cannot be used here because thin-walled small-size specimens do not meet Griffith's criterion. Therefore introduction of other characteristics such as crack tip critical stress and strain is required. These characteristics should be applied for the description of full-scale cladding rupture conditions. It is not a trivial task, and FSUE CRISM "Prometey" specialists in physics of strength take part in its solution.

Examination of radiation-induced structural changes and their effect on physical and mechanical properties play an important role at INM. Many researchers noted the effect of radiation-induced swelling on electrical resistivity. Its effect on yield characteristics is less investigated. At INM a lot of results in this field have been obtained, and simple quantitative models have been developed. The models enable estimation of the swelling effect on electrical resistivity and Young's modulus [10]. According to examination experience, with swelling over 3-4% it is a dominant factor for physical properties to vary. It ensures a rather accurate estimation of swelling value with non-destructive testing and use of the results for determination of yield characteristics, when choosing cladding areas for destructive tests.

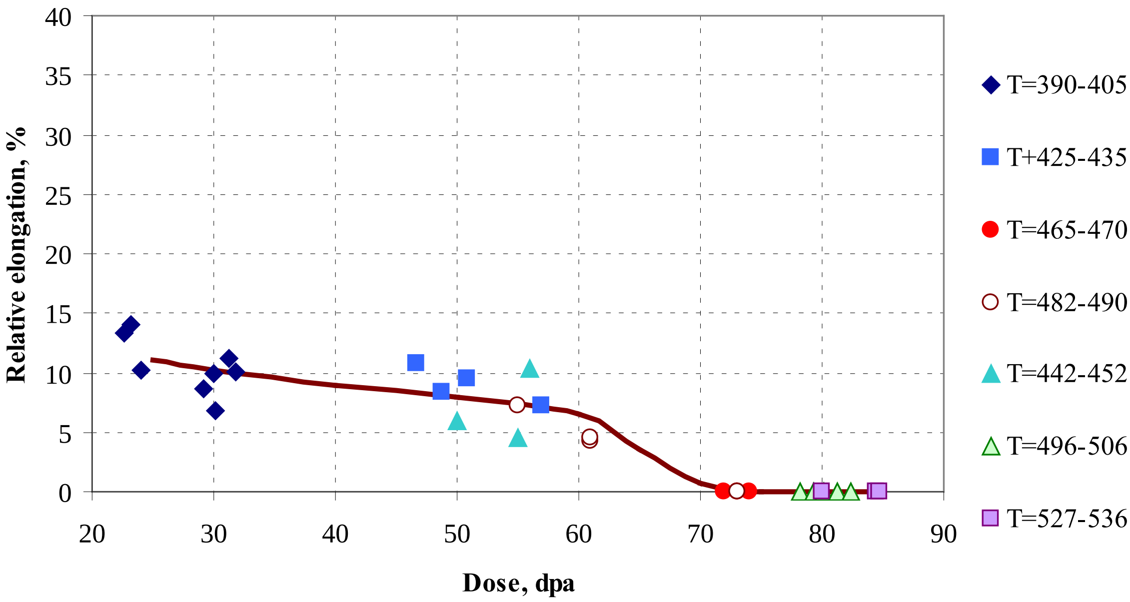
Quantitative characteristics of porosity are important for examination of structural changes. To obtain them transmission and scanning electron microscopy techniques have been developed. Using this techniques proper void distribution by size can be attained, and uniformity of spatial distribution can be investigated [11].

The obtained results show a dominant mechanism of void generation in austenitic steels under neutron irradiation [12]. Incubation swelling stage is modeled [13]. Void evolution modeling helps to identify the causes and conditions of the steady-state swelling stage with constant swelling rate [14]. Within developed models swelling rate value is calculated for ChS-68 and EK-164 steels, using their structural and physical characteristics (grain diameter, dislocation density, energy of vacancy and interstitial migration and formation etc.) [15].

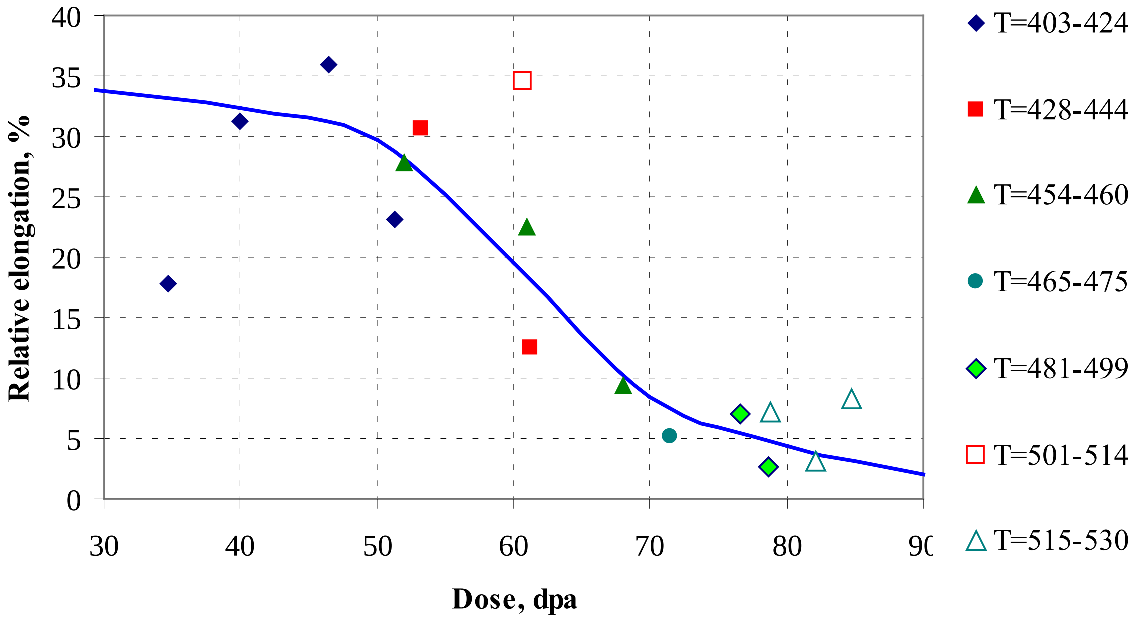


1 – rod, 2 – shroud, 3 – specimen, 4 – aggregate, 5 – washer, 6 – base

FIG. 1. Cladding loading conditions for tubular specimen testing with internal pressure of tough plastic aggregate.



a



b

FIG. 2. Relative elongations obtained for EK-164 cladding specimens irradiated at temperatures in the range between 400 and 500 ℃ and tested at 20 ℃: a – for tensile testing of ring specimens, b – for internal pressure testing of tubular specimens.

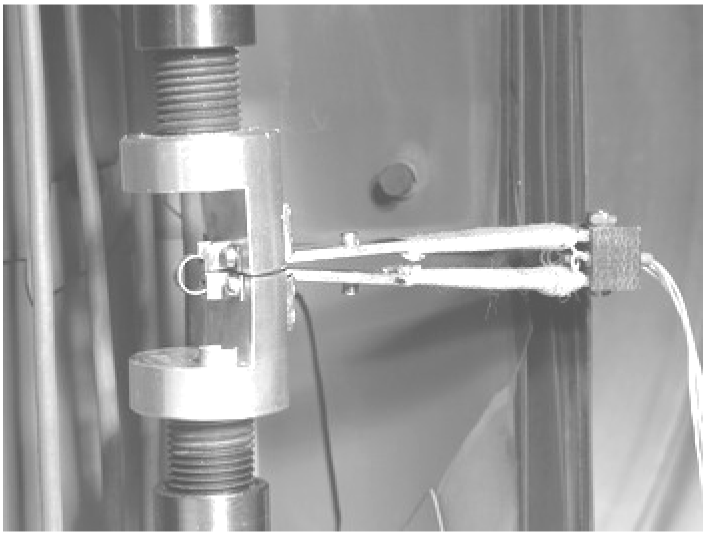


FIG 3. Circuit of rupture strength testing of notched ring specimens (eccentric tension).

5. Conclusion

A set of results for BN-600 reactor cladding examination at INM is used for a sequential increase of standard fuel burn-up to 9...11.2...13 % FIMA and corresponding damage dose increase to 70...82...87 dpa.

The results of the examination of fuel assemblies with EK-164 fuel elements are used to justify fuel assembly trial operation up to 100 dpa. The following examinations show the possibility to increase fuel burn-up and damage dose for EK-164 claddings to 14-15 % FIMA and more than 110 dpa, respectively.

An examination technology to define patterns of changes in structure, physical and mechanical properties, as well as cladding corrosion state characteristics, depending on operation characteristics (neutron irradiation temperature and dose, rate of atom displacement generation for cladding environment), has been developed at INM. Cladding areas to be examined are chosen in accordance with cladding swelling value estimated with non-destructive testing, identification of areas with maximum corrosion damage, and cladding-fuel interaction characteristics.

Techniques of mechanical properties determination during tubular specimen test with internal pressure of tough plastic aggregate and determination of rupture strength characteristics under eccentric rupture of notched ring specimens made of irradiated claddings have been developed.

Transmission and scanning electron microscopy techniques have been developed to obtain proper void distribution by size and investigate uniformity of spatial distribution.

Using the results of examination of microstructure and physical and mechanical properties a dominant mechanism of void generation in austenitic steels under neutron irradiation has been identified, as well as causes and conditions of the steady-state swelling stage. Also the rate of these processes for claddings made of ChS-68 and EK-164 austenitic steels has been estimated.

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