SMALL MODULAR REACTOR
NUCLEAR SYSTEMS

SOME NEW R&D FOCUS IN STRUCTURE MATERIALS LICENSING FOR THE SVBR-100 REACTOR FACILITIES

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SVBR-100 PROJECT: MATERIALS SELECTION

Basic principles for primary circuit materials selection

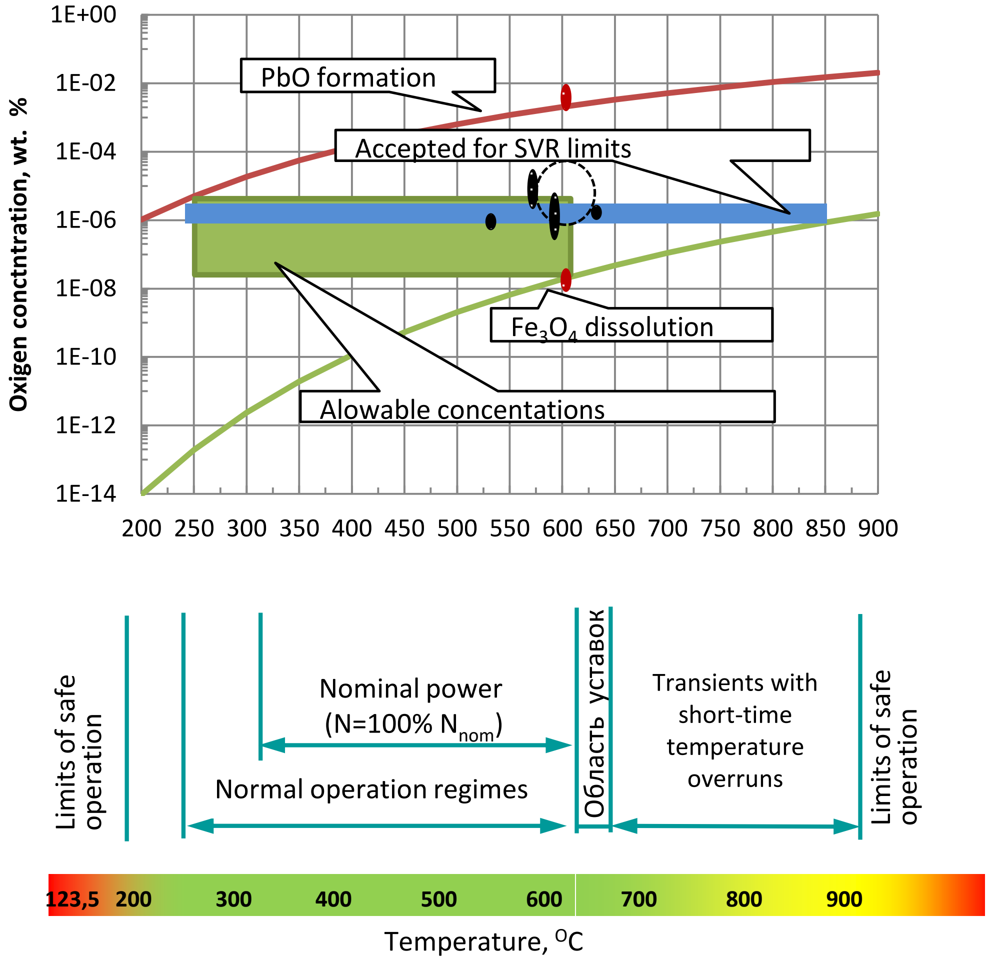
- The project uses lead-bismuth coolant and mastered technology of dissolved oxygen control in the coolant to ensure corrosion resistance of materials
- Tthis is generation IV technology that can demonstrate, among other, how the inherent safety characteristic of fast neutron heavy liquid metal cooled (HLMC) reactors can be converted into cost savings
- Tthis technology is based on industrially available materials and technologies that have clear limits, including temperature limits, thus, design elements, materials, technological schemes, parameters and modes of operation have been optimized to exclude redundant systems and equipment, the volume of construction and finally – capital costs for plant construction

SVBR-100 primary circuit materials

- EP-823 (16Х12МВСФБР) steel of ferritic-martensitic class
- EP-302(Х15Н9С3В) steel of austenitic class
- 08Х18Н10Т steel of austenitic class.

These materials meet the necessary requirements in terms of their characteristics, have been mastered by industry and tested in real operating conditions

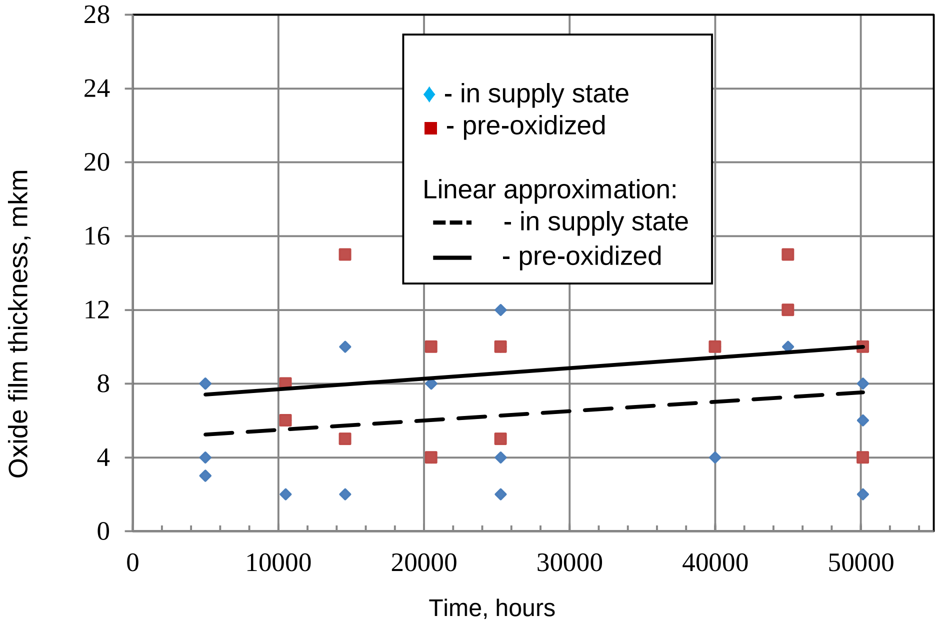
LONG-TERM & SHORT-TERM CORROSION RESISTANCE OF FUEL ROD CLADDING MATERIAL



- 6000 - 29530 hours
- 500-600 hours
- 50 000 hours

- During the development of the project, a detailed analysis of the parameters of the reactor plant in all operating modes, including standby, start-up, and operation at different power levels, planned and emergency shutdown was performed.
- The key designs limits are determined taking into account the transients and uncertainties.
- Research area for material testing is significantly larger and includes extreme temperatures and oxygen concentrations

KEY RESULTS OF LONG-TERM RESOURCE TESTING OF FUEL ROD CLADDINGS (I)



Bench corrosion testing of fuel rod models in the flow of coolant lead-bismuth was carried out on two identical loops SM-1 and CU-2M in the SSC RF-IPPE at the design values of the dissolved oxygen concentration and maximum circulation velocity of the alloy

- There is no effect of exposure in a lead-bismuth coolant on the mechanical properties of steel
- The mechanical properties of the shells vary within the limits typical for pipes made of ferritic-martensitic steel subjected to normal thermal aging at 600 ° C

- The total time for 12 campaigns of resource corrosion tests at temperature of 600 ° C was 50 135 hours, i.e. corresponded to the core lifetime provided for the SVBR-100 reactor
- As a result of corrosion tests, the claddings of fuel rod models (both in the delivery state and pre-oxidized according to the technological process previously adopted for fuel rods with lead-bismuth coolant) were not subjected to any corrosion-erosive damage.
- No liquid metal corrosion, no signs of development of crevice corrosion at the places of sample spacing in perforated grids of working areas, nor the progress of corrosion-erosion damages of spacing ribs of tubular samples
- The applied artificial defects with the complete removal of the "protective oxide film" after 25 269 hours effectively "self-healed" without the additional corrosion-erosive damages

KEY RESULTS OF LONG-TERM RESOURCE TESTING OF FUEL ROD CLADDINGS (II)

The results of metallographic studies have shown that steel retains high structural stability under prolonged exposure to high temperatures. The influence of the coolant on the change of mechanical properties of the metal is not noted.

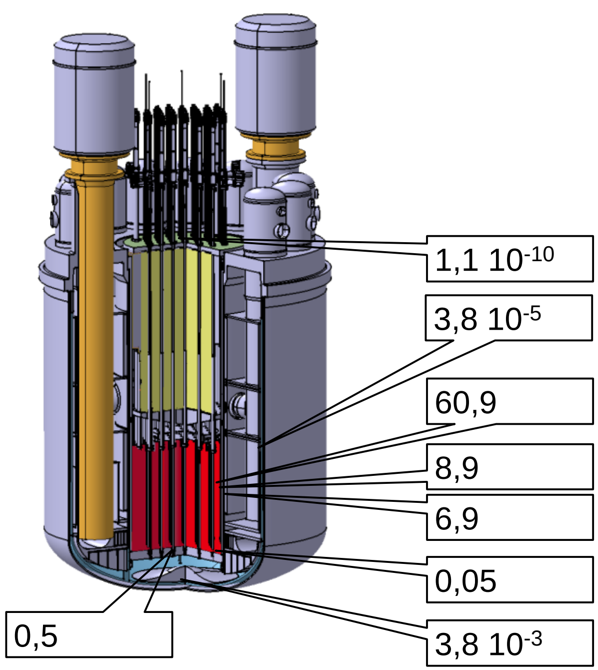
A)

B)

Appearance of samples and microstructure of material after tests in Pb-Bi:
A) 50135 hours at 600 ^{O}C;
B) 2895 hours at 620-630 ^{O}C

OPERATION CONDITIONS OF SVBR-100 REACTOR MATERIALS

The values of typical damage dose (in dpa) on the SVBR-100 reactor elements (for reactor lifetime)

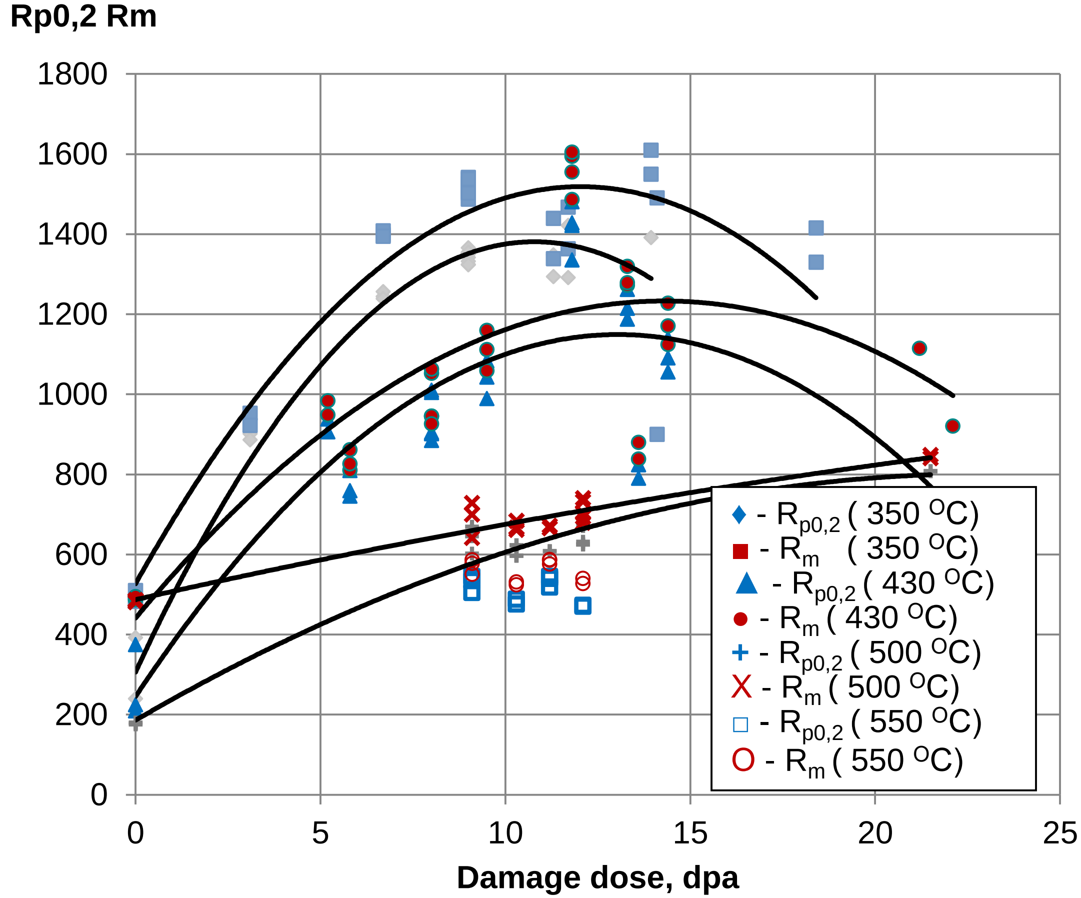


Title and dimension	Parameter value at corresponding power level				
	100%	70%	50%	30%	0%
Thermal power, MWt	280	196	140	84	0
Coolant temperature, °C: - at core inlet; - at core outlet; - at steam generator inlet - at steam generator outlet (top/bottom)	339 484 478 333/335	321 465 459 313/316	308 453 447 299/304	298 443 437 285/292	200-250
Maximum (with «overheating factors») temperature of fuel rod cladding, °C	620	602	589	579	250-300

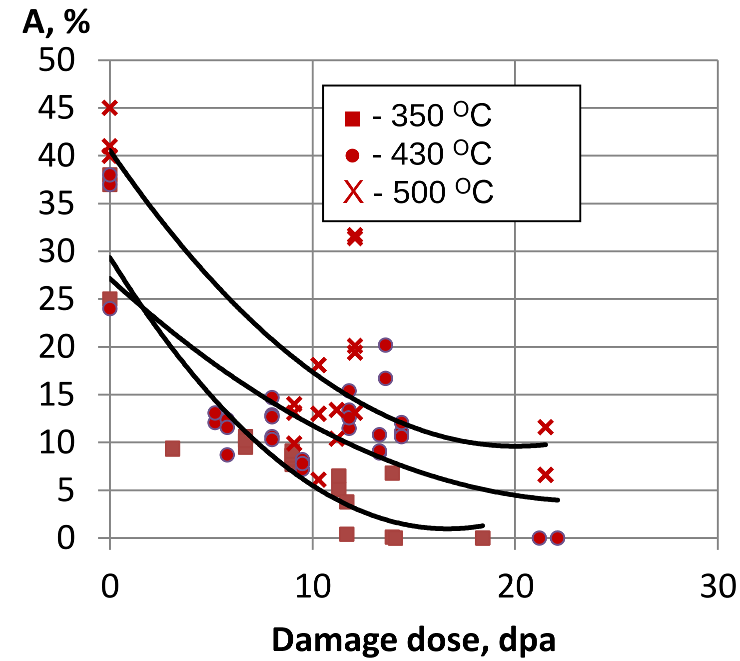
Reactor testing program for SVBR materials were performed taking into account predicted operating conditions for its elements (for reactor lifetime)

DEPENDENCE OF EP-302 STEEL PROPERTIES AND WELDED JOINTS ON THE TEST TEMPERATURE AND THE DAMAGING DOSE

Dependence of the tensile strength and yield strength of EP-302 steel and welded joints on the test temperature and the damaging dose



Dependence of elongation of EP-302 steel and it welded joints on temperature and damaging dose



As expected, general dependence of sufficiently rapid hardening of the steel and embrittlement, up to zero values of relative elongation, were found and evaluated

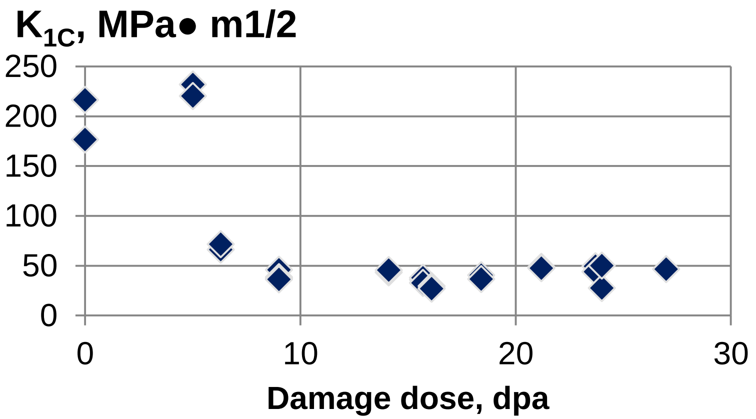
KEY RESULTS OF EP-302 STEEL IRRADIATION TESTS

Comparison of the obtained data with similar data for the well-studied 08X18H10T steel showed that:

- radiation resistance of steel EP-302 are slightly worse than those of steel 08H18H10T, and the mechanisms and dependencies of the changes of mechanical properties are somewhat different from similar mechanisms for steel 08X18H10T
- the values of radiation swelling for steel EP-302 are located in the field of variation the same data for steel 08X18H10T

The criteria for the acceptability of austenitic steel properties were selected similar to those developed for VVER and BN reactors.

- value of relative elongation A ≥ 1% and reduction of area ≥ 50%;
- fracture toughness K_{1C} ≥ 50 MPa•$m^{1/2}$.



Under such criteria, the damaging dose to the materials of reactor internals should be limited to the value of 6÷8 dpa.
To meet the accepted criterion, the design of the reactor was modified with changes in both the design of the reactor and the use of materials for its elements.

FUTURE R&D PROGRAM (IN CONCLUSION)

In substantiation of service properties and licensing of structural materials for SVBR-100 scheduled:

- Study of long-term strength, plasticity and creep
- Study of cyclic strength and kinetics of crack growth during cyclic tests of steel and its welded joints in presence of coolant
- Radiation studies of steel in a narrower area corresponding to normal operation to increase statistics
- Annealing conditions of steel and its characteristics after annealing mode

The future R & D program will focuses on the following pilot activities:

- irradiation tests of fuel rods in BOR-60 and BN-600 reactor plants
- resource tests of models of steam generators (on the first and second circuits)
- primary pumps resource tests
- creation of an experimental facilities for the study of physical and chemical processes in the conditions of destruction of the core elements

The main focus of future R & D for the qualification of materials for SVBR-100 will be:

- In-depth study of extreme states of materials
- Damage mechanisms for specific structures and processes under the influence of damaging factors
- Study of the chemical interaction of structural materials, fuel and coolant at extreme temperatures corresponding postulated severe accidents

The most important and critical for reactor concept experiments gave successful results

Thank you for attention !