Abstracts
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Application of VK-50 Reactor Plant Operation Experience to Address Radiation Safety and Explosion Protection Issues in a Single-Circuit VVER-SCP

A.S. Kursky,
JSC TVEL, 49, Kashirskoe highway, Moscow, 115409

This paper presents research results on hydrogen explosion protection and radiation safety assurance in a vessel-type boiling water reactor with natural circulation of coolant. Both these aspects are ensured by the design features of equipment installed at turbine ejector outlets. Efficiency of the hydrogen catalytic combustion technology demonstrated at the VK-50 reactor facility is shown for both normal operation and accident conditions. In two-circuit LWRs, environmental radioactive releases comply with the technological solutions implemented into the activity suppression unit. Research data of the last 60 years can serve to validate the efficiency and safety of single-circuit VVER-SCP reactors, provided that relevant experiments confirm the similarity of radiolytic processes and interphase radioactivity transfer in wet saturated steam and supercritical pressure (SCP) conditions.

Key Words: single-circuit scheme, coolant boiling, catalytic combustion of radiolytic hydrogen, gaseous fission products.

EDN: RLLZAB

UDC 621.039.5

Reflector Efficiency Analysis for Pulse Fast Neutron Research Reactors

D.G. Chereshkov,
Joint Institute for Nuclear Research, 6, Joliot-Curie st., Dubna, 141980

As key nuclear reactor elements, neutron reflectors surround the core, reduce neutron leakage therefrom and align neutron fluxes. Traditionally, fast neutron reactor reflectors and blankets use high atomic weight nuclei such as ^{208}Pb and ^{238}U. However, pulse periodic research reactors with small cores often use various steels and beryllium, which are also good reflectors for fast and intermediate neutrons. Reflectors can also serve to increase thermal neutron fluxes in experimental channels, and to change neutron lifetimes and pulse durations.

This paper analyzes and compares the efficiency of reflectors made of different materials in pulse fast neutron research reactors. It considers how the reflector material affects the reactor parameters such as in-core multiplication factor, thermal neutron flux density on moderator surfaces, average fission neutron lifetime per generation (which characterizes the fission pulse duration), and fuel isotopic composition during the first year of reactor operation at rated power.

Key Words: neutron reflectors, pulse periodic reactor, fast neutron reactor, neptunium nitride, reflector efficiency.

EDN: ZOVDQY

UDC 621.039.586

MAVR-TA Code Validation by STORM Experiments on Fission Product Transport in Reactor Primary Circuit

E.S. Sarychev
National Research University "Moscow Power Engineering Institute",
14, Krasnokazarmennaya st., Moscow, 111250,
NRC "Kurchatov Institute", 1, Akademika Kurchatova sq., Moscow, 123182,
Yu.B. Shmelkov,
NRC "Kurchatov Institute", 1, Akademika Kurchatova sq., Moscow, 123182

NPP radiation safety assurance even in beyond-design-basis accidents with fuel meltdown (including severe ones) is a necessary condition for nuclear power development. During severe accidents, fission products are released in aerosol, gas and vapors forms, and their retention in the primary circuit determines the potential scale of radioactive release into the containment and the environment. Moreover, at later accident stages, the fission products deposited on the walls may resuspend back into the circuit, hence creating an additional source of radioactivity in the containment. This paper describes respective resuspension model validation in SR09, SR10, and SR11 experiments from the international STORM program that is based on the force balance method and implemented in the MAVR-TA code. Variant calculations performed in model sensitivity analysis framework rely on both constant and model-based resuspension rates. The constant rate relies on expert estimation, which shows good agreement with experimental data, though the resuspension starts before the respective experimental phase. At the same time, model calculations based on the force balance method underestimate the resuspended aerosol fraction. The model sensitivity analysis shows the force balance model to have high and low sensitivity to material properties and friction coefficient, respectively. The resulting modified model eliminates the disadvantages of both the force balance model and the constant-rate calculations.

Key Words: severe accident, aerosols, primary circuit, fission products, experiments, deposition, resuspension.

EDN: TJYPIG

UDC 539.166.2+621.039.51...17

Software Module Development for Monte Carlo Calculation of Gamma Transport

V.I. Belousov, M.V. Ioannisian, M.R. Malkov, A.N. Pisarev, K.F. Raskach,
NRC "Kurchatov Institute", 1, Akademika Kurchatova sq., Moscow, 123182

This paper considers Monte Carlo algorithms implemented in the GAM physical module of the KIR code [1] to solve gamma-ray transfer equations. GAM is intended to simulate γ-transport problems of any complexity. GAM preliminary validation relies on a series of simple test calculations performed in simplified systems, the results of which are compared with those obtained by the precision MCNP code [2].

Key Words: gamma-ray transfer equation, Monte Carlo method, precision calculations, gamma-ray transfer algorithms, KIR program, GAM software module.

EDN: PCPDYV

UDC 621.039.4

Simulating Noble Metal Behavior in MSR Primary Circuits

P.V. Gatsa, V.V. Ignatiev, O.S. Feinberg,
NRC “Kurchatov Institute”, 1, Akademika Kurchatova sq., Moscow, 123182

In molten salt nuclear reactors (MSRs), insoluble fission products (noble metals or NMs) are deposited onto primary circuit internal surfaces. Under specific conditions, this may result in: overheating caused by decay heat released upon fuel salt discharge into the drain tank; tellurium-induced intergranular cracking in high-nickel alloys; and higher intermediate heat exchanger thermal resistance due to deposit buildup therein. Developed specifically to simulate NM transport, the NM-MSR software relies on mass transfer models and similarity methods to calculate average volume concentrations and deposition rates over the primary circuit. This paper provides calculated data on: distributions of deposited NMs including tellurium isotopes for both research and commercial MOSART reactors designed specifically to process transuranic elements extracted from irradiated fuel of PWRs with cavity-type cores; NM deposition parameters, tellurium fluence levels; decay heat distributions for both research and commercial MOSARTs; and intermediate heat exchanger efficiency over the reactor operating cycle.

Key Words: noble metal, high-nickel alloy, Molten Salt Reactor (MSR), Molten Salt Actinide Recycler and Transmuter **(**MOSART), intergranular cracking, decay heat, deposits, primary circuit, tellurium, fuel salt.

EDN: VDTWDF

UDC 539.1.074.55

Instrument for Noble Gas Volume Activity Measurement in Emergency Modes

V.O. Nebolsin,
NRC “Kurchatov Institute”, 1, Akademika Kurchatova sq., Moscow, 123182

Modern NPP safety assurance relies on multiple passive and active systems including the one that monitors volume activities of noble radioactive gases. When present in gas-air media (including releases) monitored at NPPs, noble radioactive gases indicate fuel cladding depressurization. Anticipated operational occurrences and accident developments require reliable online status data on NPP protective barriers to be available for the operator to develop appropriate measures. Post-accident monitoring of noble radioactive gas volume activities is equally important. Indirect methods implemented to monitor volume activities above 10^8 Bq/m^3 do not provide the required measuring accuracy due to significant contribution of gamma-active nuclides to measured results. The current task is to develop a radiometer to enable direct measurements of noble gas volume activities ranging from 10^8 Bq/m^3, which corresponds to normal operating conditions, to 10^{17} Bq/m^3, which is characteristic for severe accidents with core damage.

Key Words: radiometer, noble radioactive gases, NPP emissions, direct measurement method, beta radiation, indirect measurement methods, protection from ionizing radiation, gamma radiation, silicon-based detection unit, emergency response measures.

EDN: ZWKPUU

UDC 621.039.546

Simulating Tritium Release from Fuel to Primary Coolant in VVERs

A.B. Sazonov, V.A. Grachev, O.S. Bystrova,

NRC "Kurchatov Institute", 1, Akademika Kurchatova sq., Moscow, 123182

This paper presents a mathematical model describing the processes of tritium release from under the fuel cladding due to diffusion mechanisms. The simulation results are in good agreement with data on tritium activity in discharges and emissions from BWR NPPs, where tritium mainly sources from fuel nuclei fission. The rate of tritium release into the coolant is found to be determined by specific energy yield, fuel matrix and cladding temperatures, and fuel burnup. Zirconium alloys are shown to effectively accumulate tritium because of both hydrogen exothermic dissolution in zirconium and high activation energy barrier at the gas—ZrO_2 interface. Respective calculations show that VVER-1000 fuel rods clad in E110 alloy annually release about 1.7—1.8% of tritium formed therein into the primary coolant in steady state fuel cycles. Use of chromium-nickel alloys as VVER fuel cladding leads to almost tenfold increase in tritium specific activity both in primary circuit process media and in NPP liquid effluents. The results obtained for both E110 and chromium-nickel alloys are confirmed by published experimental data on hydrogen isotopes' behavior in these materials.

Key Words: VVER, BWR, fuel, tritium, zirconium, cladding, diffusion, permeability, coolant.

EDN: FCYDOK

UDC 621.039.52

Thermohydraulic Simulation of IRT-3M Fuel Assembly with Account of Nonuniform Power Distribution

D.V. Pasko, N.V. Smolnikov, M.N. Anikin, I.I. Lebedev, A.G. Naymushin, I.A. Ushakov, O.Yu. Dolmatov,

Tomsk Polytechnic University, 30, Lenin Avenue, Tomsk, 634050

Traditional methods of simulating thermohydraulic processes in research reactor cores based on one- or two-dimensional approaches (e.g., ASTRA or ATHLET codes) have significant limitations. They do not account for three-dimensional coolant flow effects and complex fuel assembly geometries, which reduces the accuracy of predicting operational temperature modes and hydrodynamic parameters. This paper uses computational fluid dynamics (CFD) methods within the SolidWorks code with the Flow Simulation extension to perform a detailed three-dimensional numerical simulation of heat and mass transfer based on solving Navier-Stokes equations with account of both turbulence and heat exchange to include spatial nonuniformities of flow parameters. The results hereof demonstrate that CFD simulation eliminates the limitations of simplified methods, provides the accuracy required to ensure both optimal core layouts and operational safety cases, and pave the way for CFD methods to be introduced in everyday practices of computational safety assessment for research reactor fuel assembles.

Key Words: IRT-T research reactor, thermohydraulic simulation, CFD methods, SolidWorks, IRT-3M fuel assembly.

EDN: PYHZWH

UDC 621.039.536.2

Thermal Ageing Mechanisms of VVER-1200 Reactor Pressure Vessel Steels

D.A. Maltsev,
NRC “Kurchatov Institute”, 1, Akademika Kurchatova sq., Moscow, 123182,
E.A. Kuleshova,
NRC “Kurchatov Institute”, 1, Akademika Kurchatova sq., Moscow, 123182,
NRNU “MEPhI”, 31, Kashirskoe highway, Moscow, 115409,
S.A. Bubyakin.
NRC “Kurchatov Institute”, 1, Akademika Kurchatova sq., Moscow, 123182

The paper presents experimental phase composition data for VVER-1200 RPV temperature sets of surveillance specimens (including base metal and welds made of 15Kh2NMFA class 1 steel and Sv-09KhGNMTAA-VI welding wire) with temperature exposure times of up to ~65,000 h yielded by: transmission electron microscopy (TEM) and scanning electron microscopy (SEM) of fine steel structure; fractography; Auger electron spectroscopy of grain boundary segregation; and mechanical tests. It is shown that the initial strengthening carbide phases of both base and weld metals undergo no significant modifications in the above temperature exposure range, so the observed changes in mechanical properties can only be due to grain-boundary segregations of impurities. Moreover, both fractography and Auger electron spectroscopy indicate that weld metals have high grain boundary phosphorus segregation levels already in the initial state.

Key Words: reactor pressure vessel, surveillance specimens, ductile-brittle transition temperature, phase composition, yield strength, reversible temper embrittlement.

EDN: XKVEBY

UDC 621.039.531:620.186.1:004.942

Computational and Experimental Evaluation of Grain Boundary Phosphorus Segregation Accumulation Kinetics in the Framework of Solving Materials Science Problems of VVER-440 Water Reactor Operation for over 60 Years

E.A. Kuleshova,
NRC “Kurchatov Institute”, 1, Akademika Kurchatova sq., Moscow, 123182,
NRNU “MEPhI”, 31, Kashirskoe highway, Moscow, 115409,
E.D. Malinovskiy, M.M. Dementyeva,
NRC “Kurchatov Institute”, 1, Akademika Kurchatova sq., Moscow, 123182

This paper presents research on structural parameters and computational-experimental simulation of grain boundary segregation in VVER-440 reactor pressure vessel steels, and demonstrates the effectiveness of recovery annealing in eliminating radiation hardening of both the base and weld metals. Auger electron research shows that grain boundary segregation accumulates in the weld metal throughout the entire service period; however, this does not lead to grain boundary embrittlement during long-term irradiation. Computational-experimental simulation of the segregation kinetics shows a significant decrease in the phosphorus grain boundary segregation accumulation rate in reactor vessels operating for over 60 years. This allows for a preliminary optimistic conclusion regarding the feasibility of VVER-440 service life extension beyond 60 years through recovery annealing.

Key Words: reactor pressure vessel, neutron irradiation, radiation embrittlement, radiation hardening, grain boundary segregation, radiation-induced precipitates, Auger electron spectroscopy, segregation thermodynamics, segregation kinetics, diffusion.

EDN: MUDBCD

UDC 621.039.546

Influence of Annealing Temperature and Time on Hydrides Precipitation and Dissolution in E110 Alloy Cladding

A.V. Rozhkov, R.A. Kurskiy, D.V. Safonov, A.S. Frolov,
NRC “Kurchatov Institute”, 1, Akademika Kurchatova, sq., Moscow, 123182,
O.O. Zabusov,
NRC “Kurchatov Institute”, 1, Akademika Kurchatova sq., Moscow, 123182**,**
NRNU MEPhI, 31, Kashirskoe highway, Moscow, 115409,
A.A. Shishkin, M.M. Grekhov,
JSC “TVEL”, 49, Kashirskoe highway, Moscow, 115409

This paper describes hydride dissolution and precipitation processes in unirradiated E110 alloy samples with 75, 115 and 200 wppm hydrogen content investigated by differential scanning calorimetry (DSC) and transmission electron microscopy (TEM) (in-situ) following different temperature programs. As the maximum annealing temperature T_{max} increases from 400 to 600 ºC, the hydride precipitation temperature T_P decreases, which can be attributed to partial annealing of dislocation structures. Microstructure analysis by TEM shows that hydrides precipitation can occur both homogeneously in sample volume and heterogeneously in grain boundaries, dislocation structures and second phase precipitates. Study of samples during in-situ heating up to 450 °C in the TEM microscope column reveals the presence of tangled dislocations and dislocation loops in hydride dissolution regions. Sixty-minute annealing at 450 °C is shown not to dissolve these dislocation structures completely. Comparison of DSC and TEM results indicates that hydride dissolution temperature T_D determined by TEM corresponds to DSC peak temperature within the error range.

Key Words: zirconium alloys, E110 alloy, fuel cladding, dry storage, zirconium hydrides, hydrogen solubility, transmission electron microscopy, differential scanning calorimetry.

EDN: PIIZUO

UDC 621.039.531, 620.186.1, 53.086

Atom Probe Tomography of Radiation-Induced Structural Evolution in RPV Steels: Data Interpretation Features

S.V. Fedotova,
NRC “Kurchatov Institute”, 1, Akademika Kurchatova sq., Moscow, 123182

This paper demonstrates possible applications of atom probe tomography (APT) to clarify radiation damage mechanisms of RPV steels, and indicates respective data interpretation features. It also analyses specific phase formation features based on atom probe tomography of RPV steels (both currently used and suggested for use in Russian-made VVERs) and provides relevant reference data for PWR steels. Research results confirm both the thermal and radiation-induced precipitation contributions to vary depending on material composition and irradiation conditions: the higher the irradiation temperature and the concentration of precipitating elements within the matrix, the higher the thermal contribution. Furthermore, APT studies confirm the feasibility of using the accelerated irradiation data to predict material behavior in terms of phase formation during extended service lifetime, and demonstrate that precipitates are the key contributors to changing RPV steel properties throughout the entire service lifetime.

Key Words: atom probe tomography, RPV steel, phase formation, fast neutron fluence, fast neutron flux, segregation.

EDN: PUHTKC

UDC 620.193.4+519.216.3

Importance Sampling Method to Optimize the Eddy Current Testing Zone in Steam Generator Heat-Exchange Tubes at VVER NPPs

O.M. Gulina, A.I. Nikeshin, A.G. Kostenko,
JSC OKB GIDROPRESS, 21, Ordzhonikidze str., Podolsk, Moscow region, 142103

Since the preventive maintenance inspection of steam generator heat-exchange tubes (HETs) is selective, some potentially damaged HETs may stay undetected. This paper suggests a special method to select HETs so as to reduce the risk of dangerous defects to be missed. The main objective is to develop a subsequent testing program to minimize the number of missed potentially damaged HETs, which is consistent with the current trend of reducing excessive conservatism in equipment technical condition assessment. In this case, the importance sampling method — i.e., testing the number of HETs in each row that is proportional to the number of defects (deeper than 60% of the HET wall thickness) interpreted in this row for the entire previous period — may be effective.

Key Words: steam generator heat-exchange tubes, eddy current testing, tube location in the assembly, damaged HETs distribution by tubesheet height.

EDN: DWRESO

UDC 629.039.58

About Technologies and Equipment for Coolant Water Deferrization in Single-Circuit VVER-SKD. Part 1. High-Temperature Titanium and Electromagnetic Filters

E.B. Yurchevsky, I.A. Chusov, V.P. Semishkin, O.A. Alyoshina,
JSC OKB GIDROPRESS, 21, Ordzhonikidze st., Podolsk, Moscow region, 142103**,**
V.M. Kuznetsov,
Intellektservice Group of Companies LLC, 2, Ugreshskaya st., Moscow, 115088

This paper discusses problems and proposals related to coolant water deferrization in single-circuit direct-flow VVER-SKD reactors. An important feature of the VVER-SKD direct-flow design, which increases coolant deferrization requirements, is the absence of water-steam interface that limits radionuclide releases with steam and is characteristic for boiling water reactors. The first part of this paper analyzes the use of pre-reactor high-temperature filters to remove iron from the coolant during startup and transient operating modes, presents the operating experience of high-temperature filters as part of serial VVER-1000 reactor plants, and provides process parameters of electromagnetic filters for turbine condensate deferrization at thermal power plants. It also shows the feasibility of full-flow high-temperature coolant water deferrization in VVER-SKD using electromagnetic filters.

Key Words: vessel-type single-circuit water-cooled nuclear power reactor, supercritical parameters, coolant deferrization, high-temperature bulk filters, electromagnetic filters, zinc dosing into water coolant.

EDN: IVOXLG

UDC 621.015.58

Important Areas of VVER Reactor Plant Feasibility Studies

N.V. Shary, V.P. Semishkin, E.A. Frizen,
JSC OKB GIDROPRESS, 21, Ordzhonikidze st., Podolsk, Moscow region, 142103

This paper analyzes important aspects of VVER NPP feasibility studies, such as strength, reliability, and safety. It demonstrates the relationship and mutual influence between these areas. The paper also discusses resource management organization at different NPP lifecycle stages depending on these areas, and determines their place in databases used by 3-level resource management information systems.

Key Words: strength, reliability, safety, degradation of material properties, damage, resource management.

EDN: KFIRPC