NUCET NUCLEAR ENERGY AND TECHNOLOGY

Nuclear Energy and Technology 9(1): 33–42
DOI 10.3897/nucet.@.100755

Research Article

'Cliff edge effects' in safety justification and operation of NPP units*

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Academic editor: Yury Kazansky ♦ **Received** 28 December 2021 ♦ **Accepted** 5 December 2022 ♦ **Published** 8 March 2023

Citation: Makhin VM, Podshibyakin AK (2023) 'Cliff edge effects' in safety justification and operation of NPP units. Nuclear Energy and Technology 9(1): 33–42. https://doi.org/10.3897/nucet.@.100755

Abstract

The authors consider phenomena that have signs of 'cliff edge effects' according to the definitions of the IAEA and NP-001-15: (1) degradation of the protective barrier (fuel rod claddings in surface boiling mode with the deposition of impurities and borates on their surface and heating of the claddings) and (2) departure from nucleate boiling (DNB) on the fuel rod claddings. Despite the fact that the first phenomenon was previously unknown, the safety of the power unit is ensured by the decisions adopted in the project.

The DNB was studied and measures were taken in the project to prevent it under normal operating conditions and anticipated operational occurrences. The protection against the DNB is also obviously ensured by reducing the reactor power due to the control systems and reactor scram. These phenomena do not reach the state of 'cliff edge effects' (according to the terminology of the IAEA and federal NPs of the Russian Federation) and are prevented at the initial stages. For a small-size reactor using dispersive fuel, it is possible to provide self-protection against the DNB, namely, due to partial washout of the fuel with the insertion of negative reactivity, followed by a decrease in power and termination of the crisis.

Keywords

Cliff edge effects, safety

Introduction

The IAEA materials and the Russian Federal Standards and Regulations (FNP RF) introduce the concept of a cliff edge effect (STI/PUB/1715 2016, NP-001-15 2015, IAEA-TECDOC-1791 2016). High-quality fulfillment of the IAEA requirements and the Russian Federation's Federal Standards and Regulations in design, fabrication and operation makes it possible to define more accurately both the concept of a 'cliff edge effect' in nuclear power as applied to particular systems, as well as the conditions for this to occur, and justification of the efficiency of the measures aimed to avoid it. Thus, a 'cliff edge effect' is considered provided other IAEA recommendations for design, fabrication and operation of power reactors are fulfilled. A sufficient safety margin shall be provided for technological measures to be taken and cliff edge effects avoided (STI/PUB/1715 2016, IAEA-TECDOC-1791 2016). It is evident that a more specific definition of a cliff edge effect and the conditions for this to occur set forth in (STI/PUB/1715 2016, IAEA-TECDOC-1791 2016) need to be justified, primarily with regard for the available

* Russian text published: Izvestiya vuzov. Yadernaya Energetika (ISSN 0204-3327), 2022, n. 1, pp. 90–106.

experimental data and the possibility for the new concept to be used to develop particular designs.

The paper presents information on the selected and explored phenomena, based on which a situation with the said 'cliff edge effect' could take place, and demonstrates technical solutions and measures to make sure that cliff edge effects are avoided.

The set of the incidents exemplified and the data considered in the paper is not exhaustive but allows, as the authors believe, assessing the sufficiency and timeliness of the measures taken in design of plants in the form of design margins, and in implementation of a full-scale defense-in-depth approach (Appendix 2 (NP-001-15 2015)).

Concept of a 'cliff edge effect'

The IAEA materials define a cliff edge effect as follows: "A 'cliff edge effect', in a nuclear power plant, is an instance of severely abnormal plant behavior caused by an abrupt transition from one plant status to another following a small deviation in a plant parameter, and thus a sudden large variation in plant conditions in response to a small variation in an input" (STI/PUB/1715 2016). Section 4.11. Design (par. b) (STI/PUB/1715 2016) notes that the design shall be conservative and the construction shall be of high quality "so as to provide assurance that failures and deviations from normal operation are minimized, that accidents are prevented as far as is practicable and that a small deviation in a plant parameter does not lead to a cliff edge effect". Attention is therefore focused on a) the definition of a cliff edge effect and its connection with and the probability for it to occur with high quality of the power unit design, fabrication and operation; b) peculiarities of the conditions for it to occur, including on "an abrupt transition from one plant status to another"; and c) on its consequences in the form of a "sudden large variation in plant conditions in response to a small variation in an input".

Attention needs to be focused on the quality of design, fabrication and operation expected to provide for such condition that "…a small deviation in a plant parameter does not lead to a cliff edge effect", that is, there is a functional link between the quality of the power unit design, fabrication and operation and the conditions for a cliff edge effect to occur.

The following is noteworthy: "a small variation in an input" causes "an abrupt transition from one plant status to another following a small deviation in a plant parameter". According to (STI/PUB/1715 2016), the design takes into account such plant states as normal operation and anticipated operational occurrences (operational states including operation with deviations) – group 1, and design-basis accidents and beyond-design basis conditions (accident conditions) – group 2. Therefore, a cliff edge effect is viewed as a phenomenon with the transition from "operational states" (normal operation or operational occurrences) to "accident conditions" as a result of a variation in the "input" and "following a small deviation in a plant parameter" (STI/PUB/1715 2016).

A Russian regulatory document, General Provisions for Ensuring Safety of Nuclear Power Plants (NP-001-15), (NP-001-15 2015) provides the following definition which is close to that by the IAEA: "A cliff edge effect is a major abrupt deterioration in the nuclear power plant (NPP) safety caused by minor variations in parameters". Attention is focused on the results – a "major abrupt deterioration in safety" caused by "minor variations in parameters". Since nuclear and radiation safety of an NPP is the "property of the NPP to ensure reliable protection of the population, personnel and the environment against the radiation effect inadmissible under the federal nuclear standards and regulations", then, as the final result, a cliff edge effect is expected to manifest itself in response to the partial or full loss of the above NPP protective function which is ensured through defense-in-depth. NP-001-15 provides particular examples to support a more accurate definition of a 'cliff edge effect' as a "major abrupt deterioration in safety" for the purpose of implementing defense-in-depth (Appendix 2 (NP-001-15 20152)). An example is provided when the "intensity of external flooding exceeding slightly the magnitude to be taken into account in the nuclear plant design leads to both safety systems and the beyond-design-basis management facilities failing due to flooding…". It is recognized that "…a severe accident due to the loss of heat removal becomes inevitable".

A situation is also considered for the VVER power units of generation I (prior to upgrades), in which no NPP auxiliary power systems suggested full independence of the system channels (including the safety system channels). "A cliff edge effect consisted in this case in that the failure of one component, the plant's dc board, could lead to cascade deterioration, up to a severe accident, in the nuclear power plant status due to a common-cause failure of systems and components" involved in different defense-in-depth levels (Appendix 2 (NP-001-15 2015)). Further, the independence of the channels was achieved and the considered cascade deterioration in the NPP status was excluded.

In accordance with NP-001-15, the "NPP safety shall be ensured through the consistent implementation of defense-in-depth based on using a system of physical barriers to the spread of ionizing radiation and radioactive materials into the environment, and a system of engineering and organizational measures to protect the barriers and keep these efficient, as well as to protect personnel, the public and the environment". The system of engineering and organizational measures shall form five defense-in-depth levels, the first of which (the conditions for the NPP deployment and prevention of operational occurrences) includes the development of the NPP design based on a conservative approach with a well-developed property of the reactor inherent safety and measures aiming to exclude the cliff edge effect as implemented in the considered example for plants of generation I.

Thus, NP-001-15 states that the NPP design shall include measures for avoiding a cliff edge effect. The criterion for such measures to be taken is justification of the required defense-in-depth (DID) efficiency at all levels while implementing the strategy for the preemptive prevention of unfavorable events at the first or second levels.

Par. 1.2.9 of NP-001-15 requires that deterministic and probabilistic safety analyses be provided for all operational states and all locations of nuclear materials, radioactive substances and radioactive waste in which there is a potential for an operational occurrence to take place.

In accordance with the NP-006-16 requirements for providing safety analyses, Chapter 15 reads: “It shall be justified for all operational states that the safety criteria are complied with during operational occurrences” and “The minimum (maximum) values shall be defined for the parameters that characterize the margins to the safety criteria”.

We shall remind that, in accordance with NP-001-15, safety criteria are the NPP parameters and/or characteristics in accordance with which the NPP safety is justified and which are set by regulatory documents or in the NPP design. At each DID level, the protection of the barriers is defined by the safety criteria set for each DID level not being exceeded (violated), including the safety criteria that limit radiation effects.

NP-001-15 also specifies the safety targets:

- the total probability of severe accidents, equal to $1 \cdot 10^{-5}$, not exceeded for each NPP unit in an interval of one year;
- the total probability of a major accidental release, equal to $1 \cdot 10^{-7}$, not exceeded for each NPP unit in an interval of one year;
- the total probability of severe accidents, equal to $1 \cdot 10^{-5}$, not exceeded for the NPP onsite nuclear fuel storages (other than being a part of the NPP units) in an interval of one year.

Therefore, implementing the targets specified in NP-001-15 makes it possible to assess the potential for the respective cliff edge effects to manifest themselves, which, subject to high quality of design, manifest themselves as a result of the transition from “operational states” to “accident conditions” and in a “major abrupt deterioration in safety” at all defense-in-depth levels. The higher is the quality of design, fabrication and operation, the smaller is the probability for a cliff edge effect to occur. And justifying the quality of design suggests justifying the consideration of all phenomena and processes that affect the state of the physical barriers as recommended in the IAEA document (IAEA-TECDOC-1575 2008). The probability of implementation can be estimated as the probability of a severe accident at the power unit. Therefore, the existing system for organizing high quality of design, fabrication and operation leads a small probability for cliff edge effects to occur, and this is what the IAEA and NP-001-15 definition focuses on.

To demonstrate the practicability of the said principles and requirements, we shall discuss examples of effects that could be treated, under certain conditions, as cliff edge effects and that have failed, due to the measures taken in the design (“justification and application of design margins, as well as implementation of a full-scale defense-in-depth” and high quality of operating water-cooled power reactors), to evolve (most unlikely to occur!) with a negative radiation effect on the personnel, the plant, the environment and the public.

Physicochemical fuel cladding – coolant interaction with potential for cladding failure and escape of radionuclides into the coolant

Since the potential source for radiation effects are primarily fuel rods that contain radioactive fission products, it is obvious that the experience of their service in reactor cores, as well as the results of tests and experiments with fuel rods and their dummies shall be analyzed when simulating ‘accident’ modes.

The fuel matrix and the fuel cladding are parts of the system of physical barriers to the spread of radiations and radioactive materials. Therefore, degradation of the protective properties of these components, implementation of the conditions for the radionuclide and irradiated fuel spreading due to the cladding failure, and the presence of radiation specific to these phenomena will characterize the hazard from the phenomena under consideration. The characteristics of the above barriers are expected to degrade as a result of the fuel cladding and fuel matrix temperature growth. This does not rule out other phenomena and processes, and does not narrow down the domain of ‘cliff edge effects’ in the light of the IAEA definition, but defines more exactly the objective of the study and makes it possible to trace how the quality of design limits or excludes the evolution of the process that can potentially lead to a ‘cliff edge effect’ (STI/PUB/1715 2016, NP-001-15 2015, IAEA-TECDOC-1791 2016).

The fuel rod properties are expected to degrade due to the interaction of the cladding material with the coolant and fuel at a high temperature. A temperature growth intensifies the processes of interaction. Oxidation and embrittlement of the cladding material (zirconium alloys) due both to restructuring and formation of hydrides with their undesired radial orientation, that affects the cladding strength, can be determining in terms of fuel failure and, ultimately, the escape of fission products from fuel rods into the coolant (Reshetnikov et al. 1995). As a result of numerous studies and analyzing the experience of operation, conditions are defined and maintained when, in the normal operation mode, the fuel cladding retains its properties and characteristics as a safety barrier. The determining parameters and characteristics are: the thickness of the oxide film on the cladding surface is equal to

(~ 10 ηm for the fuel lifetime) and is five to six times as small as the admissible limit value (60 ηm); the content of hydrogen is much smaller than the limit value; and there are no problems with the cladding hydration (hydrides form in small quantities and are arranged chaotically) (Reshetnikov et al. 1995). The thickness of the deposits on the cladding surface is not large and does not exceed 1 ηm for the fuel lifetime (Reshetnikov et al. 1995). The above characteristics are achieved by the initial state of the cladding material and by complying with the water chemistry (WC) requirements. One of the key WC characteristics, the content of oxygen in the coolant, shall not be exceed 5 ηg/kg. The content of hydrogen in the initial state of the cladding material is also limited (to not more than $1.5.10^{-3}$%) (Reshetnikov et al. 1995). Therefore, high quality of the cladding material in the initial state and standard operating procedures, including complying with the specified WC standards, lead, as experience of operation shows, to design margins for the cladding corrosive damage. In normal operating conditions, there is a margin for the fulfillment of requirements to the cladding: the corrosion criterion not exceeded both in terms of the oxide film thickness, and in terms of the hydrogen content in the cladding, the hydride orientation in the cladding, etc. This 'design margin' can be used for taking measures.

In operational states and in accident conditions, it is required to ensure not only that uranium fission products are localized and the absorbed dose is not exceeded for the public and personnel, but also that the core can be dismantled after the emergency process is over, excluding the processes taking place during severe beyond-design-basis accidents.

There is an example of the PWR operation when minor variations in one parameter, the fuel rod power density with a particular combination of the WC characteristics and the reactor lifetime, led to an Axial Offset Anomaly (AOA), that is, an anomaly of the axial power density distribution, a phenomenon earlier unknown in PWRs (Bennett et al. 2004, Henshaw et al. 2006, Kritsky et al. 2011, IAEA-TECDOC-1666 2011, Kritsky et al. 2012). Optionally, the term 'Crud Induced Power Shift' (CIPS) is used (Kritsky et al. 2011). The axial offset (AO) in PWRs is defined as the ratio of the difference in the power between the core's upper half and lower half related to total power (IAEA-TECDOC-1666 2011). The axial offset (AO) variations are defined in a range of 3 to 17% and are explained by formation of abnormally high deposits (thickness of ~ 0.1 mm) and concentration of boron compounds in such deposits. This example is important since it demonstrates the sufficiency of the measures taken in the design to limit the earlier unknown processes involving degradation of the fuel cladding properties, namely a decrease in the protective properties of the cladding as a safety barrier. The AO reduction measures have been successfully developed and implemented due to the measures taken, primarily through reduction of power, an analysis of the status, and detection of factors that affect the processes (Kritsky et al. 2011, IAEA-TECDOC-1666 2011).

Such 'earlier unknown processes' manifested themselves when the reactor life was extended and the power of individual FAs was increased. The AOA was caused by the PWR operation cycle, i.e. the reactor life, extended to 16 to 18 months and by fuel with an increased enrichment used to that end, that is, when the earlier adopted design conditions were changed. As a result, such changes in the design conditions led to an increase in the power density in a number of fuel rod in the core and, sequentially, to a greater intensity of the surface boiling in the upper part of the reactor core. At the same time, as found in simulation tests in the Halden research reactor (Bennett et al. 2004), a power reduction leads to some compounds, e.g. lithium metaborate, that can occur in the form of deposits on the cladding surface in the boiling region, dissolved or 'washed out' after the boiling is over, and cannot be therefore recorded in a post-irradiation examination. The dissolution of lithium metaborate in the coolant at the stage of the reactor power reduction (accordingly, when changing from a mode with coolant boiling to a mode without coolant boiling) and during further operation at a reduced power was recorded from the increased content of lithium in the reactor coolant. Besides, more stable ('durable') compounds such as Ni_2FeBO_5 (the Callaway PWR) also formed on the surface.

Porous deposits of corrosion products form on the fuel rod upper part surface in the mode under consideration, in which boron compounds concentrate and formation of such compounds as lithium metborate is possible (Fig. 1). Three layers can be identified in Fig. 1a, the intermediate layer being lighter in color (zirconium oxide). An increase in the concentration of corrective and impurity components (boric acid, lithium, borates, etc.) takes place in the outer layer and in the intermediate layer due to the so-called "wick" effect in the pores as coolant evaporates (Fig. 1b). The thickness of these deposits for the boiling regions is 30 to 130 ηm (the porosity of the deposits is about 80%). Such deposits do not only worsen the heat transfer and lead to a higher rate of the cladding material corrosion due to the cladding temperature growth beneath the deposit layer but also cause problems due to the asymmetrical change of the reactor core power density field (AOA) (IAEA-TECDOC-1666 2011).

Abnormal deposits of corrosion products also form in the FA upper part and the maximums of the deposit distribution match the violent coolant boiling areas in respective FAs. More high-energy assemblies have more deposits in the boiling areas. Coolant boiling is the key factor that induces the formation of deposits on the fuel rod surface. The considered process differs from the crud (coolant impurity) deposition in that the crud deposition can take place on the lower grids and lead to a growth in the pressure drop in the reactor core (Kritsky et al. 2011).

The formation of abnormal deposits and the AOA effect were observed not in all PWRs operating in the mode of an increased energy region. It has been found that, apart from surface boiling, the deposit growth rate is greatly influenced by:

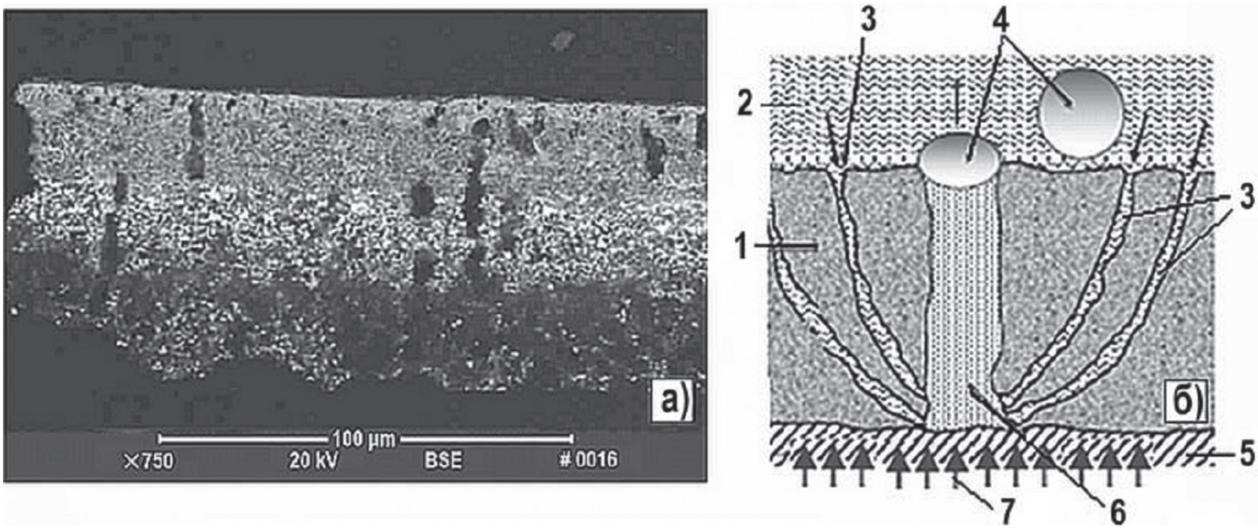

Figure 1. Steam channels in deposits in high-energy areas of the PWR FA fuel rod surface **a.** A photograph with three regions (intermediate, light – ZrO_2) (Henshaw et al. 2006); **b.** Idealized diagram: 1 – oxides; 2 – coolant; 3 – water channel; 4 – steam bubbles; 5 – fuel cladding; 6 – steam channel, 7 – heat supply (Kritsky et al. 2011).

- the concentration of corrosion products in the coolant (with the pH varying from 6.9 to 7.2, the amount of deposits changes from 2.8 to 0.5 ηg/cm^2, that is, by a factor of five (IAEA-TECDOC-1666 2011));
- the composition of the structural material and the corrosive state of the circuit equipment surface (nickel and mixtures of chromium, nickel and iron oxides are found in deposits on the fuel cladding (IAEA-TECDOC-1666 2011));
- the initial concentration of boric acid and, accordingly, the content of LiOH in the coolant (the deposit layer increases with the growth in the concentration of lithium, potassium and fluorine ions (IAEA-TECDOC-1666 2011));
- the cycle duration during power operation.

For power units with no AOA, the concentration of nickel in the circuit coolant is in a range of 1 to 5 ppb; for comparison: the concentration of nickel for one of the power units with an AOA of 17% is 11 to 53 ppb.

Therefore, an increase in the FA energy with coolant boiling can lead to the surface deposits and cause overheating and oxidation of the fuel cladding with degradation of its protective properties. Processes are possible in deposits with concentration of the boron compounds leading to an AOA.

The adopted axial offset monitoring procedure makes it possible to record the AOA phenomenon. A reduction in the local power density after the boiling stops leads to the dissolution of borates. Specifically, the presence of lithium in the coolant confirms the dissolution of lithium metaborate and elimination of the anomaly. Therefore, there are means both for the process diagnostics and control.

Noteworthy is the complexity of the processes taking place in conditions of developed surface coolant boiling (see Fig. 1b). There are results both with deposition of boron compounds on the fuel cladding and with the surface cleaning with boric acid during coolant boiling (such mode "with cleaning" was implemented in the VK-50 reactor (Zabelin et al. 1968)).

It became possible to study and reproduce the phenomenon under consideration in a research reactor (Halden) (Bennett et al. 2004). It has been found that the formation of abnormal deposits can be reduced by increasing the coolant's pH at the beginning of the reactor operating cycle with dosing of zinc in the coolant and cleaning of fuel rods and the circuit of corrosion products. When taken in PWRs, these measures lead to a reduced rate of the deposit accumulation and to a reduced AOA. The most effective way to prevent the formation of abnormal deposits on fuel rods is to reduce the rate of vaporization by reducing the fuel rod local energy and intensifying heat exchange by mixing the coolant flows with different enthalpies.

The phenomenon with formation of abnormal deposits on the fuel cladding surface was not known earlier (at the design development stage). The process was stopped at early stages and explored through the design approaches taken (axial offset measurement and monitoring), as well as through engineering and organizational solutions practically implemented in the course of redesign in connection with a longer reactor life and the reactor power increase, namely, more accurate WC determination, boiling rate reduction, etc. It will be reasonable to note that the initial step was the reduction of power for investigating the phenomena. After the model representations were explored and developed, tests were conducted in the Halden reactor and proposals were prepared for excluding the phenomenon in question. This example demonstrates that it is possible to control the process at initial stages while avoiding the power emergency condition.

In-pile tests of fuel rods

Tests in the AM-1 reactor, Obninsk

Channels with thermally profiled fuel rods were tested in the PV-2 loop facility of an NPP with the AM-1 reactor in Obninsk (Zenkevich et al. 1969). The dryout phenomenon was investigated. One of the test objectives was to compare the dryout occurrence conditions in reactor and bench conditions. There were 16 in-pile tests held and bench experiments were conducted. Test parameters: pressure of 11.8 to 13.7 MPa; inlet coolant temperature of 288 to 307 °C; mass rate of 1500 to 2590 kg/(m2s); steam quality of 18 to 26%; channel power of 37 to 60 kW. The channel axial power density form is $q(z) = q_{max} \cos (\pi \cdot z/2200)$. The power density form was the same in the bench conditions. The authors recognize that the time point taken as the crisis in the in-pile tests and characterized by the temperature growth to 500 °C is conventional, being preceded by a 'heat-exchange drop' mode with temperature fluctuations with an amplitude of up to 50 °C. The experiment was terminated when the cladding temperature reached 500 °C, that is, without causing the fuel rod to lose integrity, that is, a state with fission products and fuel particles entering the loop circuit. It would be more correct to treat the 'crisis' as a mode with the so-called heat-exchange 'drop' with the fuel cladding temperature fluctuations that precedes the mode with a major temperature growth. It has been found that the maximum difference of the critical thermal power, N, determined in the bench conditions, N_c, and in in-pile conditions, N_p, is $(N_p - N_c)/N_c = 13\%$. Normally, $N_p - N_c > 0$ except one case in 16. The crisis was detected in the upper cross-sections of the dispersion fuel rods (at the coolant channel outlet). A further power increase leads to the 'heat-exchange drop' mode spreading to the inlet region. As applied to the problem under consideration, the protection against a 'cliff edge effect' is ensured by selecting the operating limits for the channel power and respective scram settings.

The design (safety) margin is estimated in this case as the difference in the power at the onset of the 'heat-exchange drop' and the critical power (~ 10%). One of the key ways to reduce the negative influence of the cliff edge effect on the NPP safety is to justify and use design margins (see Appendix 2 (NP-001-15 2015)). It is exactly this interval that can be assessed as being limiting for deciding on the power reduction when it is not possible to monitor the critical heat flux ratio using local parameters.

Tests in the SM-2 reactor, Dimitrovgrad

Of practical interest is an experiment conducted in the SM-2 reactor involving a process with local cladding and fuel melting. The purpose of the tests was to determine the fuel rod power at which the fuel rod fails in the DNB conditions (Bobrov 2004, Bobrov et al. 1997, 1998, 2004). The in-pile experiment was expected to "reveal the outcome of the fuel-cladding thermomechanical and chemical interaction, assess the radiation effects of the fuel rods being in the crisis conditions, and determine the amount of fuel entering the coolant" (Bobrov et al. 1998). The fuel rod behavior was investigated in the actual conditions of service when there is overpower with a DNB and with causing local fuel rod melting, and the quantity of the fuel matrix to have entered the loop circuit was determined. The loop facility coolant activity limit defined the permissible extent of the fuel rod melting ('degradation' of the composition).

The test fuel rod had the same shape and composition as the upgraded SM fuel rod (the content of ^{235}U in the fuel rod is 6 g). The fuel rod is cross-shaped, and has a thin-wall stainless-steel cladding and UO_2 fuel in a matrix of copper-beryllium bronze. There was a thermocouple installed in the fuel rod's kernel the reliable contact with which is ensured by the fabrication technology (press molding and annealing). The fabrication technology also provides for the reliable thermal contact of the fuel matrix and the cladding. The SM fuel rod constant with water cooling at a rate of ~ 10 m/s is estimated at less than 0.1 s.

The test procedures, the fuel rod design and the test conditions are described in detail in (Bobrov 2004, Bobrov et al. 1997, 1998, 2004). The test results, as we believe, are the best for demonstrating the evolution of a cliff edge effect since, in the event of a DNB, the fuel rod (the cladding and the matrix) is likely to melt in a short time with the matrix melt and the particles of uranium dioxide likely to interact with the coolant with a major growth in the circulating circuit activity, that is, with radiation effects. The initial phases of this process were simulated in the experiment. Besides, means were provided for the process termination at the beginning of the process development.

An overall view of the irradiation device is presented in Figs 2, 3 shows the measured fuel rod temperature values as a function of the fuel rod power. A power of less than 45 kW leads to convective heat exchange (without coolant boiling), and a higher power leads to a mode with the surface coolant boiling. A further power increase with $Q = 71 \pm 3.5$ kW leads to a DNB (Fig. 4).

In Fig. 4, the initial segment ($0 < \tau < 50$ s) is characterized by the kernel temperature fluctuations of not more than 5 °C with a much lower frequency than the neutron power fluctuations of about 1% (the ionization chamber readings). The time point the DNB was reached at was recorded from an abrupt disproportional increase in the fuel kernel temperature relative to the power.

Violent fuel kernel temperature fluctuations were observed in Segment A (Fig. 4). The fluctuation amplitude in the segment with $\tau > 50$ s is 15 °C and, as the authors of (Bobrov et al. 1998) believe, reflects the vapor film formation and breakdown. With the fluctuation start time ($\tau = 50$ s) taken as the DNB onset time, one needs to take into account that the fuel rod was in operation for 6 min and retained its integrity in this mode. The fuel rod

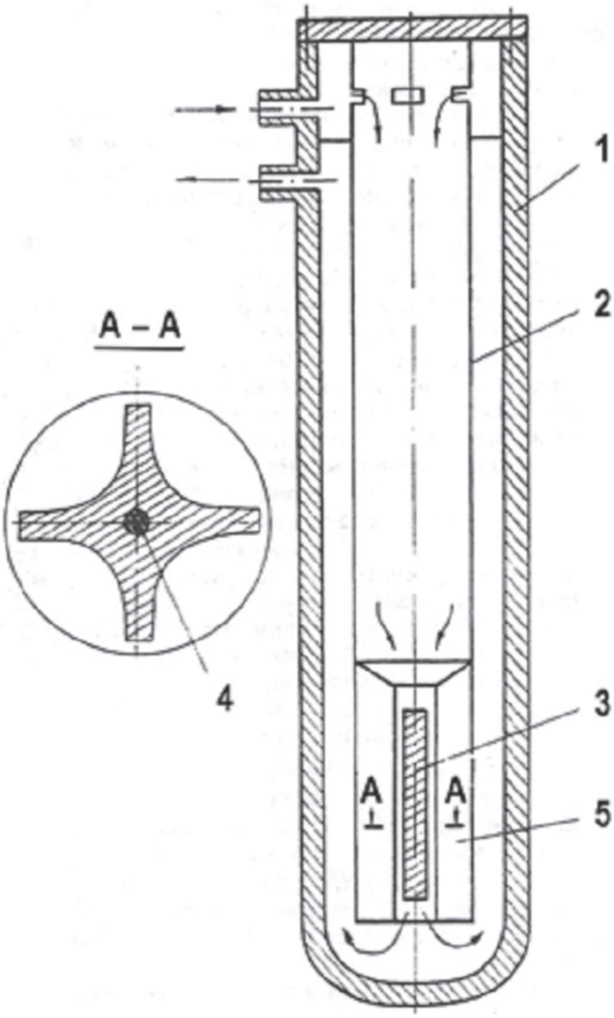

Figure 2. Irradiation device, overall view: 1 – channel body; 2 – flow splitter; 3 – fuel rod; 4 – cross-section with fuel rod, splitter and thermocouple (thermocouple is shown by a 'dot' in the central part); 5 – thermal barrier.

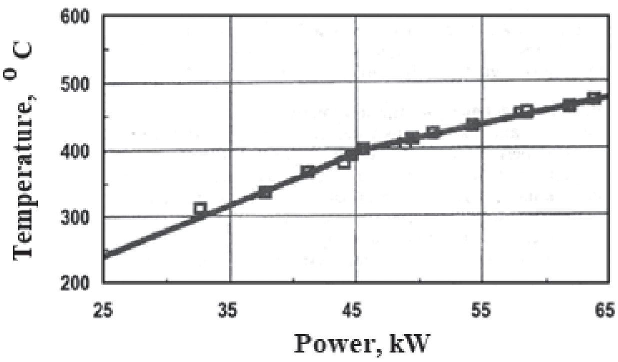

Figure 3. Fuel temperature variation as a function of power.

burnout took place at a 3% power increase (Segment B) with the specific heat flux being 14.3 MW/m^2. The temperature setting was reached with a rapid temperature growth followed by scram ($\Delta\tau_{delay}$~ 0.15 s). After about 300 s (the time for which the coolant is delivered to the gamma detector), the coolant radioactivity was recorded and estimated at not more than ~ $1\cdot10^{-4}$ Ci/l (the operating limit for a loop facility).

With $\tau \geq 50$ s, the test parameters are as follows:

- Coolant: channel inlet temperature of 82 ± 2 °C; mass rate of ρV= 9250 ± 470 kg/(m^2 s); pressure of 5.0 ± 0.25 MPa; activity of up to ~ 1 10^{-4} Ci/l.
- Fuel rod: power of 71 ± 3.5 kW; perimeter-average surface heat flux density of 14.3 MW/m^2 (estimated with regard for the measured fuel rod power value).

The design (safety) margin is estimated in this case as the difference in the fuel rod power in Segment B and at the beginning of Segment A (fuel rod temperature fluctuation onset, Fig. 4): 2 to 3 kW.

The post-test inspection of the fuel element appearance showed a local damage area of about 40 mm long found 50 mm below the central section, and ~ 225 mm from the fuel rod top, at the computationally predicted point (Fig. 5). There are cladding melting traces visible on the fuel rod cross-section slices (Bobrov 2004, Bobrov et al. 1998, 2004). There were pores observed in the fuel composition.

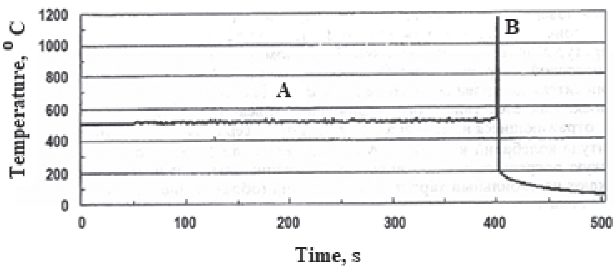

Figure 4. Dependence of fuel temperature in the crisis mode: A – τ = 50 – 400 s; B – temperature growth with scram.

The adopted test termination procedure via scram, after the thermocouple reaches the emergency setting of 709 °C, (the scram delay is estimated at 0.14 s) limited the fuel melting and localized the damage with not more than 0.05% of the accumulated radionuclides having entered the coolant. The radioactivity of the nuclides that had left the fuel did not exceed the permissible activity value (the maximum value is not more than $1\cdot10^{-4}$ Ci/l).

Therefore, the results of two in-pile DNB test types have been discussed. Using the terminology in (Kirillov 1974, Isachenko et al. 1975), the test conditions match the 'slow-rate crisis' or dryout or 'liquid film dryout' and the 'fast-rate crisis' or premature burnout with the transition of bubble boiling to film boiling simulated in the experiment under consideration in a hollow between the fins (Fig. 5). Slow-rate crisis is observed with comparatively high steam qualities and comparatively low mass flow rate values.

The in-pile test results for thermally profiled dispersion fuel rods indicate to close values of the critical fuel rod power determined immediately in the in-pile tests, N_p, and in bench conditions, N_c. The difference in the values N_p and N_c is not large and amounts to $(N_p - N_c)/N_c \leq 13\%$.

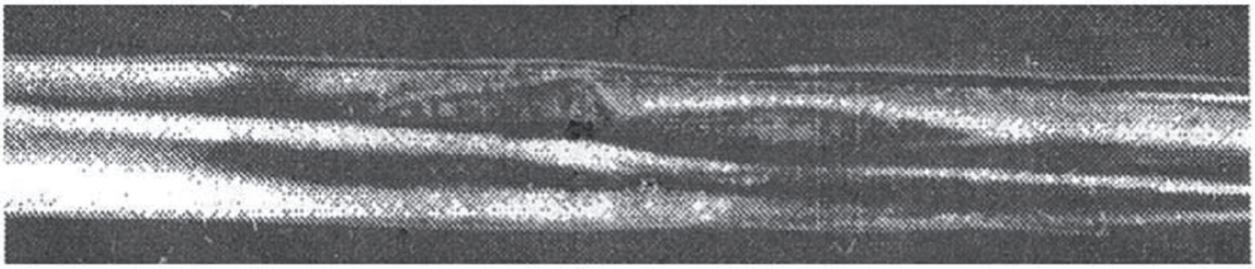

Figure 5. Appearance of the fuel rod segment with failure.

Premature burnout occurs in harder conditions (in terms of temperature growth and the maximum temperature of the fuel composition). The pre-irradiation (bench) tests made it possible to calculate fairly correctly the location of the segment with local fuel rod melting and to minimize the radiation effects.

Noteworthy is the low fuel rod inertia and the fast rate of transition to the cladding and fuel matrix melting stage caused, primarily, by the small value of the fuel rod constant (~ 0.1 s) and by the high heat flux surface density value. The SM fuel rod constant is much smaller than the VVER fuel rod constant (by an estimated factor of 30). This example shows that the said conditions with a low-inertia fuel rod allow the process to be controlled without causing major fuel rod damage and a large quantity of radionuclides entering the circuit with the coolant.

Of practical interest is the possibility for the self-regulation of the process. The crisis is experienced by the segment with the maximum power and neutron flux density. A negative reactivity is inserted with the reactor power reduction as the fuel composition melts and is entrained by the coolant, that is, a local crisis with the fuel composition melting is self-regulated which is specific to small cores. The process regulations reflect the need for the shutdown at intermediate power levels with acquisition and analysis of the key sensor data. No reaching the subsequent power level is allowed before data on the radiation situation is available. This excludes the development of the emergency process after a local DNB occurs, the fuel matrix is washed out, and the reactor power is reduced. The presented data characterizes the dynamics of these processes. The peculiarities discussed indicate that it is possible to regulate or control the process.

DNB and post-DNB peculiarities are considered in (Averyanov et al. 1989, 1992) for fuel rod simulator bundles. Four versions of variously shaped simulators are considered: a conditional 'square' with extended 'ears' (two types), a 'cross' (see Fig. 2), and a 'ring' (cylindrically shaped). FAs with 7, 19 and 61 simulators were used. As noted by the authors, the "experiments have confirmed once again the observations that "the DNB in the rod bundles is relatively mild by nature" (Averyanov et al. 1989, 1992).

Therefore, this conclusion proves that there is an additional FA critical heat flux margin estimated at ~ 10% and 3% in the considered AM and SM reactor tests. The difference in the design margin in the considered tests (10% and 3%) is defined by the 'crisis' types and by the difference in the fuel rod design and test modes.

Experience of commissioning new reactor facilities (early period of nuclear power)

An example of both the occurrence of the 'cliff edge effect' characteristics and the measures taken to avoid it in further operations is the experience of a power increase for phase 1 of the Beloyarsk NPP (Kochetkov 2001, 2014). In the initial period, the AMB-100 reactor operated at a power equal to 70% of the rated value set in the design. The facility operated stably at the above power level in the course of the year. As it reached the rated power (100%), that is, the power was increased as specified in (Kochetkov 2014), seven channels began to "fume", that is, the fuel rod cladding integrity was lost. 'Fuming' is characteristic of a state with radionuclides (fission fragments) entering the coolant of a single-circuit reactor facility. Accordingly, a radioactivity increase is possible in the unit rooms and in the turbine. 'Degraded conditions of heat removal' have been defined as the major reason for the power limits at both unit 1 (AMB-100) and unit 2 (AMB-200) of the Beloyarsk NPP's phase 1 (Kochetkov 2001). During that period (the 1960s and the early 1970s), there was no commonly shared conception of but there were two hypotheses to explain the process development. The VTI experts attributed the degraded conditions of heat exchange in the reactor's evaporation channel (dryout) to a high steam quality with a short supply of the water phase when the steam flow "breaks the water film from the walls" of the fuel rods. This hypothesis was further recognized as correct. To avoid dryout at a later stage and to raise the reactor power, the coolant flow rate was increased by changing the fuel tube design (the inner tube diameter was increased from 9.1 to 12 mm) (Kochetkov 2001). The considered AMB-100 tests form one of few examples of a cliff edge effect to have occurred as a result of the phenomenon having been underexplored prior to building the plant and imperfect theoretical concepts of dryout as a phenomenon, as well as due to the absence of a system for monitoring the key parameters to record intermediate states before the cliff edge effect. A comparison of the AMB-100 test and the AM and SM tests confirms that the probability for cliff edge effects to occur decreases substantially as knowledge is accumulated. Therefore, the formulations discussed at the top of the paper reflect the state of our knowledge after 50 years of accumulating this knowledge.

Conclusions

The notion of a 'cliff edge effect' is defined both in the IAEA documents and in Russian materials (NP-001-15). This is a "severely abnormal plant behavior caused by an abrupt transition from one plant status to another following a small deviation in a plant parameter". Indeed, it has been shown by an example of analyzing the characteristics of fuel cladding as a safety barrier that no characteristics reach the limit values during normal operation, and there are design margins treated as defined in Appendix 2 (NP-001-15 2015). Therefore, degradation of characteristics is possible only in the event of an anticipated operational occurrence, including emergency conditions.

Where current regulatory requirements for high quality of the reactor plant design, fabrication and operation are complied with, no 'cliff edge effect' is likely to occur. The sufficiency of the design measures taken or the design margins adopted to limit (avoid) this effect and correct the WC modes is shown by an example of an incident in new PWRs when an earlier unknown phenomenon manifested itself during operation with an axial offset anomaly (abnormal axial power density distribution in the core) and degradation of characteristics. Using the practically implemented engineering solutions adopted in the design development process (more accurate WC determination, boiling intensity reduction, etc.), the process was stopped at initial phases and investigated. After exploring the peculiarities of the coolant boiling in the FA upper part, determining more accurately the conditions for the deposition of boron compounds on the fuel rod surfaces in the boiling region, and reproducing these in a simulation experiment in Halden, measures were developed to avoid the above effect (more accurate WC determination, circuit cleaning, introduction of mixing grids in FAs, etc.).

Based on rather an extensive array of the actual AOA events, a conclusion can be made that the engineering solutions and design margins adopted in the design, as well as the requirements for the operation of NPPs with PWR reactors have ensured the safety of the NPP and excluded the negative development of the AOA effect. This effect did not occur in the VVER operation thanks to the design margins adopted in the development.

A DNB incident appears to be more hazardous. The in-pile tests were considered with two well-known DNB types (premature burnout and dryout). The tests have shown that the bench and in-pile test results are comparable (a power difference of up to 13%). The in-pile experiment conditions are predicted in a pre-irradiation computational analysis with an acceptable accuracy. No new phenomena were identified in the in-pile experiment. As compared with in-pile tests, bench test results are normally conservative, and provide, if used in the design, an additional design margin estimated as 3 and 10% for the two in-pile test modes (premature burnout and dryout). One can therefore believe that no fuel rod DNB occurs during normal operation (NO) and during operational occurrences (AOO) in the reactors, for which the tests have been conducted, thanks to the measures adopted in the design. It is required to investigate additionally the influence the transversal currents in the VVER reactor cores, made up of bare fuel assemblies, have on safety margins.

For small cores, e.g., in the SM reactor with a substantial power peaking, the occurrence of a local crisis with the fuel composition interaction with the coolant and the fuel and fission product being 'washed out' into (entering) the coolant is expected to lead to insertion of negative reactivity (self-regulation effect) and to a power reduction with limited fuel and fission product 'washout' into the coolant.

The phenomena and processes discussed herein may be 'cliff edge effects', as defined in (STI/PUB/1715 2016, NP-001-15 2015, IAEA-TECDOC-1791 2016), in the event the whole range of systems fail or no measures are taken as specified in the process regulations or respective instructions. As specified in NP-001-15, the key documents defining the operating safety of a nuclear plant is the process regulations for the NPP unit operation that contain the rules and key operation techniques, general procedures for carrying out safety-related operations, as well as safe operation limits and conditions. This document is developed in accordance with the NPP design and the NPP safety analysis report involving the reactor facility and NPP design developers.

The situations at the PWR and AMB reactors discussed in the paper occurred as the result of a power increase (AMB) or the reactor life extension (PWR), that is, when new modes were adopted. The design margins adopted in the design for the key parameters, e.g., for power, are 'selected'. These modes adopted for economic reasons shall be comprehensively analyzed and investigated.

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