STATE OFFICE
FOR NUCLEAR
SAFETY

COMMON POSITION ON LICENSING REQUIREMENTS FOR THE NEW VVER FUEL SUPPLIES

110 00 Praha 1, Senovážné náměstí 9
DS: me7aazb
podatelna@sujb.cz

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AUTHORS

Alžběta Bednářová (SÚJB – Czech Republic)

Daniel Vlček (SÚJB – Czech Republic)

CONTRIBUTORS

Elizabeth Tsvetanova (BNRA - Bulgaria)

Anti Daavittila (STUK- Finland)

Michal Melichárek (UJD – Slovakia)

Eszter Rétfalvi (HAEA – Hungary)

ABBREVIATIONS

AFD	Axial Flux Distribution
BUC	Burnup Credit
DNBR	Departure from nucleate boiling ratio
LOCA	Loss of Coolant Accident
LB LOCA	Large Break Loss of Coolant Accident
LTA	Lead Test Assembly
NI	Nuclear installation
NPP	Nuclear Power Plant
PBC	Partial Boron Credit
FR	Fuel rod
PIIP	Post Irradiation Inspection Programme
FA	Fuel assembly
RIA	Reactivity Initiated Accident
TH	Thermal and hydraulic
SFSP	Spent Fuel Storage Pool
SÚJB	State Office of Nuclear Safety

PURPOSE OF THE DOCUMENT

As the diversification of fuel supplies becomes increasingly important, the State Office for Nuclear Safety of the Czech Republic (SÚJB) identified the need to simplify and harmonize licensing requirements for new fuel supply across countries utilizing VVER-type fuel. This led to the creation of a set of requirements for licensing new fuel, followed by discussions on differences between various national approaches.

The following set of requirements was selected and reviewed by the national authorities of Bulgaria, Finland, Hungary, and Slovakia, with their insights and comments incorporated into the text. These requirements represent a minimum common framework that should be fulfilled in order to approve a new fuel supplier for Temelín NPP, Dukovany NPP, or any other VVER NPP. However, this document does not constitute a final and exhaustive set of requirements for fuel licensing. Each country may impose additional national requirements that could not be fully harmonized within this document. While this is not a complete harmonization effort, it represents an important first step toward achieving it.

The document is based on SÚJB Safety Guide BN-JB-3.2 *Design of the Pressurized Water Reactor Core* and on IAEA Safety Standard No. SSR-2/1, *Safety of Nuclear Power Plants: Design, Specific Safety Requirements, No. SSR-2/1 (Rev. 1)*, together with IAEA TECDOC No. 1720, *Operation and Licensing of Mixed Cores in Water-Cooled Reactors*.

1 OPERATIONAL EXPERIENCE

(1.1) Applicable operational experience or sufficient experimental measurements shall be gained for FAs or core components of the same design and properties or very similar properties which serve as a reference.

(1.2) The results and experience shall be gained in such a nuclear or experimental installation where they are operated and examined under the same operational conditions.

2 THERMAL HYDRAULIC ANALYSES

(2.1) The fundamental objective of thermal and hydraulic design of the reactor core shall be to ensure such transfer of heat from the core to ensure the removal of heat from the core in all operational states.

(2.2) The specific design limits shall evaluate parameters such as in particular:
- The maximum local and mean linear or overall performance of the FE;
- The minimum DNBR;
- The maximum temperature and enthalpy of the fuel pellet;
- The maximum temperature of FR cladding.

(2.3) The design analyses should adequately take account of the uncertainties in the values of process parameters, neutron-physical properties of the reactor core and the calculation methods used in the evaluation of thermal-hydraulic design limits.

(2.4) The analyses should take account of the design limits for the minimum and maximum coolant flow in the core and the limits should be respected or a detailed analysis should be presented to demonstrate that the design limits of the reactor core are not exceeded. The total flow through the core, FAs and bypass of the core shall be established in the thermal-hydraulic design of the FA.

(2.5) The analyses of the thermal-hydraulic design of the reactor core shall consider all specific design characteristics of the FAs and the associated production and operation deviations which in particular are the rod pitch, the power of the FEs, the shape and size of the sub-channels, spacer grids and mixing grids including their acceptable operational deformations.

(2.6) The thermal-hydraulic design should ensure that the determination of the DNBR takes into account the fact that the critical heat flux correlations were determined on the basis of the tests performed in stationary conditions. Consequently, the margin established for the non-stationary initiating events in normal and abnormal operation assessment should be used and clearly demonstrated. Since the critical heat flux correlation was measured under steady state stationary conditions, the margin should be adequate enough to avoid damage to the FE even under non-stationary conditions in normal and abnormal operation.

(2.7) The experiments to establish the critical heat flux correlations should be executed for a sufficiently wide range of the envisaged operating conditions and with the sufficient number of points measured to ensure the determination of the limit values of minimum critical heat flux can be statistically evaluated in accordance with the world practice. In addition, evaluation of used DNBR correlation and its limits and value of turbulent mixing shall be carried out for every DNRB correlations used.

(2.8) The DNBR shall be calculated for selected fuel loads to demonstrate the sufficient safety of the core. To demonstrate the compliance with the thermal-hydraulic design requirements, the assessment of the DNBR with the critical heat flux correlations should be ensured with a 95% probability at the 95% level of probability that the “hot” fuel rod in the core does not achieve the conditions of boiling crisis under no conditions of normal or abnormal operation.

3 NEUTRON PHYSICAL ANALYSES

(3.1) The design of the reactor core shall set the key safety parameters characterizing the neutron-physical properties of the reactor core. The set of key safety parameters should be the following:

- The fuel temperature coefficient of reactivity (Doppler coefficient);
- The moderator (coolant) temperature coefficient of reactivity;
- The void coefficient of reactivity for the coolant and the moderator;
- The rate and efficiency of change of the concentration of soluble absorber in the coolant;
- The rate and magnitude of insertion of positive reactivity caused by the movement of control rods or changes in process parameters;
- The rate and magnitude of insertion of negative reactivity associated with a reactor trip (the rate of insertion of override reactivity);
- The minimum delayed neutron fraction for beginning and end of cycle, burn-up limits;
- The shutdown margin;
- The radial and axial power peaking factors (coefficients of power distortions for FEs and FAs), including the allowance induced by xenon oscillations, where relevant;
- The relative maximum linear heat rate;

(3.2) The above-mentioned factors should be set out to cover selected events that can occur during operation of the NI for all FA types. Analyses should demonstrate for all states of the NI that the fuel design limits applicable to the specified states of the NI are not exceeded.

(3.3) The design of the reactor core shall include the calculations of stationary and non-stationary spatial distributions of neutron flux and of the heat power, neutron-physical characteristics and the efficiency of the means of reactivity control for selected states of the NI.

(3.4) Key safety parameters such as reactivity coefficients shall be calculated for the selected operational states (e.g. the zero power, nominal power, beginning of life, end of life) and for selected fuel loads. Their dependence on the fuel charge and the burnup of fuel shall be analysed.

(3.5) The major initiating events associated with the reactivity insertion should be considered such as:

- The ejection of the most effective control rod;
- The uncontrolled entry of clean condensate (dilution of boric acid);
- The uncontrolled withdrawal of one group of the control rods.
- The drop of the control rod;

(3.6) The limitation on the rate and amount of reactivity insertion shall be established on the basis of the analyses demonstrating compliance with the design criteria of fuel

(3.7) The compatibility of the FAs with the in-core measurement system, including compatibility between the sensors readings and the results of the software calculations shall be assessed. In addition, the neutron fluence on the reactor pressure vessel, reactor internals and the templates should be calculated.

4 MECHANICAL AND THERMOMECHANICAL ANALYSES

(4.1) The design of the FA and the NI shall ensure that no damage is done in the conditions of normal and abnormal operation to the FAs due to the mechanical load induced particularly by the following effects:

- All handling of FAs and FRs including loading;
- The intentional and unintentional changes in power level;
- The contact forces on the FA (which compensate for the hydraulic-dynamic lift forces and the changes in the geometry of the reactor internals and the FAs; due to irradiation and thermal expansion);
- The temperature gradients;
- The hydraulic forces including transverse flow among the deformed FAs or the different FAs in the mixed core;
- The irradiation effects on the materials (e. g. radiation induced growth and swelling);
- The hydraulic-dynamic vibrations of the FR and the vibration fretting of the FR cladding (Grid-to-Rod fretting);
- The non-elastic deformations of the FA skeleton that may lead to excessive deformation of the FA – bowing and twisting.

(4.2) The criterion for the failure of the FR cladding during power ramps shall be supported by one of the following approaches or their combination, taking account of the characteristics of the specific type of the FE, in particular fuel and FR cladding material:

- In-core tests, the so-called in-pile power-ramp tests;
- In-core test modelling; or
- Experiments conducted on the irradiated FR cladding.

(4.3) The integrity of the FEs due to the hydraulic-dynamic effects (e.g. flow induced fretting and vibration, pressure drops) shall be demonstrated by means of the tests performed on the qualified hydraulic loops, using the dummy fuel assemblies (full-dimension is preferred) and in the prototype test conditions (e.g. pressure, temperature and transverse flow).

(4.4) The design of the reactor core shall include stress analyses such as an assessment of the relaxation of the springs of spacer grids during FA irradiation to prevent the possibility of grid-to-rod-fretting. The stress and geometrical stability analyses shall include the effect of irradiation on the FE, FA and core components, on mechanical properties of the materials used such as the tensile strength, ductility, radiation induced growth, creep, radiation hardening and relaxation.

(4.5) The compatibility of the FAs with the current water chemistry in the primary side for various operational states shall be assessed.

5 DESIGN BASIS ACCIDENTS

(5.1) In design basis accident conditions, the FR cladding shall not be damaged to the extent that it had failed to withstand mechanical stress during accident (e.g. load induced by LB LOCA). The assessment shall take into consideration the rate of oxidation of the FR cladding both before the transition and during the transition (from the outside of the FR cladding and if necessary, from the inside of the FR cladding for failed FEs); it shall further include chemical interactions between the pellet and the FR cladding. The recommended design basis criteria shall limit:

- Oxidation of the cladding;
- Hydridization of the cladding;
- Increase of fuel radial average enthalpy; and
- Cladding temperature.

(5.2) The specific effects of fuel behaviour considering expansion and deformation such as ballooning, rupture of the FR cladding, and twisting and bowing of FEs and FAs, spike effects or flow induced vibrations shall be included in the above mentioned analyses and their impact on heat removal and the subsequent release of fission products from the FR shall be assessed.

(5.3) The design of the reactor core shall ensure that the changes in the geometry of the FA and the FR are limited to avoid the contact or the mechanical interaction between the FR and the top part of the FA and between the FR and the bottom part of the FA. Furthermore, that the bowings of the FRs and/or the FAs are limited and that the deformations of the control rods and their other possible interactions with the guide tubes of the FA do not affect the structural integrity of the FA and/or the safety function of the control rods (e. g. reactor control, shutdown, long-term cooling).

(5.4) The effect of irradiation on the resistance of spacer grids to compression in the horizontal direction shall be respected in an assessment of the seismic events or LOCA (analysis of the mechanical response of the FA to seismic/LOCA load – evidence of maintaining the coolability of the FR and the FA geometry).

(5.5) The FA as a part of the core shall be evaluated to the seismic events.

(5.6) If required, selected beyond design basis accident scenarios shall be assessed.

6 CONTROL SYSTEM

(6.1) The design of the FA and of the reactor core shall facilitate and ensure the use of the necessary instrumentation and detectors for the monitoring of core.

(6.2) The control rods shall be designed to be capable of rendering the reactor subcritical even in the conditions of design basis accidents and the design extension conditions without the core melting.

(6.3) The design of the control rods and other core components shall ensure that they withstand all handling operations during refuelling, transport and storage without being damaged.

(6.4) The control rods design shall take into account their wear and the irradiation effect in operation, e.g. absorber burnup, changes in physical properties and production of fission gaseous products.

7 FUEL MANAGEMENT

(7.1) The analyses of the design of fuel loading and associated calculations shall cover cases with limiting values of parameters for operational events and operational states during selected campaigns.

(7.2) The safety assessment shall describe initiating events that could cause unexpected criticality in fuel handling during refuelling.

8 STORAGE

(8.1) The design of the NI and of the cask shall be to meet the requirements for subcriticality for the cask charge in transfer, transport, storage and disposal of fuel including the adequate uncertainties and margins for all the types of FA used.

(8.2) The design of the NI for storage and of the cask shall ensure that the value of effective multiplication factor is not exceeded:

- 0.95 for all states; and
- 0.98 for the conditions of optimum moderation (e. g. flooding) for the casks in dry or for wet storage.

(8.3) The temperature limits for the FA shall not be exceeded during wet and dry fuel storage and shall be evaluated.

(8.4) The design of the storage grids shall include analyses for fresh fuel determining the value of subcriticality for the group. The analyses shall be particularly conducted for the following conditions:

- Nuclear fuel flooding with clean water with maximum density;

- Drop of the FAs in handling of FAs (e. g. on the top parts of other FAs stored in the SFSP).

9 TRANSPORT

(9.1) The analyses of subcriticality for the casks (transport and storage, irradiated and fresh fuel) shall be conducted for the conditions of design basis accidents and the conditions of optimum moderation. In case of flooding (or optimum moderation) the effective multiplication factor shall not exceed 0.98, in other cases 0.95. The transport casks may be licensed separately or already licenced cask can be used within the specification frame.

10 MIXED CORE

(10.1) If the applicable operational experience and experimental measurements in the operation of the mixed core cannot be demonstrated (e. g. co-location of the new and operated type of nuclear fuel in the core), the LTA programme shall be proposed.

(10.2) The design of the mixed core shall include the analyses of the mechanical and thermo-hydraulic compatibility between the current and the new FA together with reactor physics calculations as well as the assessment of the deformation conditions.

(10.3) The design of the mixed core should include calculations of pressure gradient redistribution between the new and the current type of the FA, which is used for determination of the force applied to the hold-down spring of the FA.

(10.4) The design of the mixed core should prevent excessive deformation of FA geometry and shall not exceed the predefined design difference in hydraulic resistance of the FAs of different types (in particular of the new type of the FA adjacent to the current type of the FA).

(10.5) The design of the new type of the FA should include evaluation of the dependence of the vibrations of individual components on the burnup, vibration intensity and the rate of fretting having regard to the conditions in the mixed core.

(10.6) The mixed core analysis should include evaluation of transversal flow between FAs.

(10.7) The penalization of $F\Delta h$ and AFD should be evaluated for every FA in the mixed core to ensure compatibility of different fuel designs.

(10.8) The calculations should adequately include all non-homogeneities at the boundaries of individual fuel types, which may result in an unexpected change in the neutron flux density and the power distribution in the core, or in a loss of the validity of the calculation methods used.

(10.9) If the types of the FA concurrently operated in the mixed core have different structural rigidity, the design should include calculations of the forces applied to the FA during design basis accidents and should determine the rate of deformation of the geometrical stability of the FA and demonstrate the performance of fundamental safety functions. Combining the bowing of the individual types of FA should not limit the capability of dropping the control rods or cause jamming the control rods in the guide tube. The analyses or experimental data should demonstrate that the geometrical stability of the FA is not challenged and that the fundamental safety functions are performed.

11 METHODS AND PROCEDURES

(11.1) Methods and procedures of the above-mentioned analyses and conducted experiments shall be documented in such manner that they are clearly and easily identifiable, traceable and reproducible.

(11.2) The input parameters shall be clearly identified in order to reproduce the data through independent evaluation.

(11.3) The independent evaluation shall be conducted.

(11.4) The computer codes and tools used for the above-mentioned analyses shall be verified and approved by local or international authority or their validation and verification shall be documented, referred and reviewed according to the regulatory requirements.