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Safety verification requirements for changes in the core fuel management mode of pressurized water reactor nuclear power plants

压水堆核电厂堆芯燃料管理模式变更的安全论证要求

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5 Overview

Changes in the reactor core fuel management model of a pressurized water reactor nuclear power plant refer to one or more of the following combinations.

a) Changes in core fuel cycle length. For example, the core fuel refueling cycle is changed from 12 months to 18 months or 24 months.

b) Changes in the initial fuel enrichment of newly added fuel assemblies. For example, newly added fuel assemblies may have a lower fuel enrichment. Change to one or more higher fuel enrichment.

c) Changes in the fuel assembly loading mode of the new reactor core. For example, changing from a high-neutron leaking core to a low-neutron leaking core.

d) Changes in the structure and materials of newly added fuel assemblies alter neutronics or thermal-hydraulic properties. For example, two or more different fuel assemblies may exhibit different properties. A "hybrid reactor core" is constructed by loading fuel assemblies that are incompatible but have significant differences in neutronics or thermal-hydraulic properties.

e) Other changes that deviate from the established core fuel management model. For example, single-cycle or multi-cycle flexible core fuel pipes. Changes in operating mode, long-term low-power operation, extended operation, changes in the configuration of control rods or flammable poison rods, and certain specific compatible materials. For irradiation tests, etc., safety assessments may be conducted with reference to the provisions of this document when necessary.

6 General Requirements for Safety Demonstration

6.1 Changes in the reactor core fuel management model of pressurized water reactor nuclear power plants will lead to changes in the initial assumptions and requirements of the Final Safety Analysis Report (FSAR). Changes in the input may affect the analysis results or conclusions of FSAR and other related documents.

6.2 Nuclear power plant operators and design units shall conduct safety assessments in accordance with relevant nuclear safety requirements and review principles, and submit them to the National Nuclear Safety Administration. The relevant department submitted an application for change of core fuel management mode and related analysis report. The justification for changing the core fuel management mode should meet (but is not limited to) the following general requirements and related contents.

a) Conduct a comprehensive safety assessment throughout the entire design process of the core fuel management mode change to ensure safety during the nuclear power plant's operational lifespan. Each stage during the period meets the design safety requirements.

b) Select initial and assumed conditions in accordance with nuclear safety regulations, including but not limited to. 1) Reactor thermal power and safety analysis involving thermal parameters (such as coolant flow rate), taking into account measurement errors; 2) Stop characteristics, such as the longest stop delay time and the most valuable bundle of control rods stuck outside the core; 3) Core neutronics parameters, affecting the moderator temperature reactivity coefficient, Doppler power, and temperature reactivity coefficient. The combination of cavitation reactivity coefficient, core axial power distribution, and radial power distribution is the most unfavorable. 4) Instrument allowable error in protection parameters and settings; 5) Reactor coolant system and auxiliary systems response, reactor protection system functional characteristics and operating characteristics, operator intervention. Pre-action, worst-case single fault, etc.

c) Safety analysis software approved by the national nuclear safety regulatory authority shall be used, and its analysis methods, mathematical models, and computer software shall be [compliant/compliant/etc.]. The calculations are verified, reasonable, applicable, and self-consistent with the design inputs and other supporting calculation software, and their accuracy is [not specified]. It meets the requirements of the engineering design.

d) The analysis should cover sufficient representative or enveloped operating conditions or state points, and select the appropriate acceptance criteria according to the operating condition classification.

e) The analysis results include core nuclear power and heat flux density, reactor coolant temperature and pressure, fuel rod cladding, and fuel pellet peaks. Value temperature, minimum deviation nucleus boiling ratio (DNBR), fuel rod burn-out fraction, and the resulting radioactive consequences, etc.

7.1 Core Fuel Management and Nuclear Design

7.1.1 When the fuel management mode changes, the actual configuration of the unit and external demand should be comprehensively considered, and the nuclear power plant's reaction should be determined based on the following conditions. Core loading schemes for each fuel cycle of the reactor.

a) Types and fuel enrichment of new fuel assemblies;

b) The number of new fuel assemblies loaded into the reactor core for each fuel cycle;

c) New fuel assembly loading methods (e.g., "low-leakage" core loading mode);

d) Types, placement, and quantities of flammable and toxic substances;

e) Core fuel cycle length;

f) Flexible operating mode;

g) Main operating parameters of the reactor.

7.1.2 Nuclear designs based on core fuel management models shall meet (but are not limited to) the following design limits.

a) Design limits for nuclear enthalpy rise thermal channel factor (F NDeltaH);

b) Design limits for peak heat flux density (FQ);

c) Limits on the temperature reactivity coefficient of moderators;

d) Minimum shutdown margin limit;

e) Maximum fuel consumption limits for fuel assemblies and/or fuel rods;

7.2 Parameters required for accident analysis, fuel and system validation

7.2.1 General Rules When the core fuel management mode changes, the parameters required for accident analysis, fuel and system validation should be adjusted accordingly. The characteristics and requirements determine the parameters, which typically include.

a) General critical safety parameters;

b) Specific key security parameters;

c) Parameters required for transient analysis during normal operation;

d) Parameters required for design verification of fuel rods and fuel assemblies;

e) Parameters required for system validation;

f) Parameters required for analyzing the long-term transient effects of boron concentration on Loss-of-Cooler Accidents (LOCA);

g) Parameters required for decay heat calculation;

h) Parameters required for fuel pellet-cladding interaction (PCI) analysis;

i) Parameters required for Expected Transient (ATWS) analysis in cases where emergency shutdown fails.

7.2.2 General Key Safety Parameters Typical common key safety parameters include.

a) Maximum and minimum values of the temperature (or density) reactivity coefficient of the moderator;

b) The maximum and minimum values of the Doppler temperature reactivity coefficient;

c) Maximum and minimum values of the Doppler power coefficient;

7.7 Accident Analysis and Assessment of Radioactive Consequences

7.7.1 General Rules Accident analysis and radiological consequences assessment should cover the accidents analyzed in FSAR, using the parameters provided in

7.2 and their... He designed the input data, taking full account of its conservative requirements.

7.7.2 Accident analysis

7.7.2.1 For specific accidents where key safety parameters change or exceed the original FSAR assumption boundaries after a change in fuel management mode. For typical accidents, conduct accident analysis and evaluation to confirm that the accident consequences meet safety limits. Analysis must be performed using specific accident-specific key safety parameters. Typical accidents include.

a) Analysis of boron dilution accidents;

b) Analysis of control rod drop accidents;

c) Accident analysis of control rod assembly runaway under subcritical or low-power start-up conditions;

d) Accident analysis of single-beam control rod malfunction under power operation conditions;

e) Analysis of control stick ejection incidents;

f) Analysis of a main steam pipeline rupture accident;

g) Accident analysis of rod assembly runaway during power operation;

h) Analysis of fuel assembly misloading accidents;

i) LOCA incident analysis;

j) Non-LOCA incident analysis;

k) Mass-energy release calculation and containment response analysis.