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SUBJECT: GUIDANCE ON THE REGULATORY REQUIREMENTS  
FOR CRITICALITY ANALYSIS OF FUEL STORAGE AT  
LIGHT-WATER REACTOR POWER PLANTS

Attached is a copy of guidance concerning regulatory requirements for criticality analysis of new and spent fuel storage at light-water reactor power plants used by the Reactor Systems Branch. The principal objective of this guidance is to clarify and document current and past NRC staff positions that may have been incompletely or ambiguously stated in safety evaluation reports or other NRC documents. It also describes and compiles, in a single document, NRC staff positions on more recently proposed storage configurations and characteristics in spent fuel rerack or enrichment upgrade requests. This guidance is not applicable to fuel storage in casks, nor does it consider the mechanical, chemical, thermal, radiological, and other aspects of the storage of new and spent fuel.

Attachment:  
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UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
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GUIDANCE ON THE REGULATORY REQUIREMENTS FOR  
CRITICALITY ANALYSIS OF FUEL STORAGE  
AT LIGHT-WATER REACTOR POWER PLANTS

1. INTRODUCTION

This document defines the NRC Reactor Systems Branch guidance for the assurance of criticality safety in the storage of new (unirradiated or fresh) and spent (irradiated) fuel at light-water reactor (LWR) power stations. Safety analyses submitted in support of licensing actions should consider, among other things, normal operation, incidents, and postulated accidents that may occur in the course of handling, transferring, and storing fuel assemblies and should establish that an acceptable margin exists for the prevention of criticality under all credible conditions.

This guidance is not applicable to fuel storage in casks, nor does it consider the mechanical, chemical, thermal, radiological, and other aspects of the storage of new and spent fuel. The guidance considers only the criticality safety aspects of new and spent LWR fuel assemblies and of fuel that has been consolidated; that is, fuel with fuel rods reassembled in a more closely packed array.

The guidance stated here is based, in part, on (a) the criticality positions of Standard Review Plan (SRP) Section 9.1.1 (Ref. 1) and SRP 9.1.2 (Ref. 2), (b) a previous NRC position paper sent to all licensees (Ref. 3), and (c) past and present practices of the staff in its safety evaluation reports (SERs). The guidance also meets General Design Criterion 62 (Ref. 4), which states:

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

The principal objective of this guidance is to clarify and document current and past staff positions that may have been incompletely or ambiguously stated in SERs or other staff documents. A second purpose is to state staff positions on recently proposed storage configurations and characteristics in spent fuel rerack or enrichment upgrade requests (for example, multiple-region spent fuel storage racks, checkerboard loading patterns for new and spent fuel storage, credit for burnup in the spent fuel to be stored, and credit for non-removable poison inserts). Although these statements are not new staff positions, this document compiles them in a single paper. In addition, a recently approved staff position for pressurized-water reactors (PWRs) would allow partial credit for soluble boron in the pool water (Ref. 5).

The guidance stated here is applicable to both PWRs and boiling-water reactors (BWRs). The most notable difference between PWR and BWR fuel storage facilities is the larger size of the fuel assemblies and the presence of soluble boron in the spent fuel pool water of PWRs.

The determination of the effective multiplication factor,  $k_{\text{eff}}$ , for the new or spent fuel storage racks should consider and clearly identify the following:

- a. fuel rod parameters, including:
  1. rod diameter
  2. cladding material and cladding thickness
  3. fuel rod pellet or stack density and initial uranium-235 (U-235) enrichment of each fuel rod in the assembly (a bounding enrichment is acceptable)
- b. fuel assembly parameters, including:
  1. assembly length and planar dimensions
  2. fuel rod pitch
  3. total number of fuel rods in the assembly
  4. locations in the fuel assembly lattice that are empty or contain nonfuel material
  5. integral neutron absorber (burnable poison) content of various fuel rods and locations in fuel assembly
  6. structural materials (e.g., grids) that are an integral part of the fuel assembly

The criticality safety analysis should explicitly address the treatment of axial and planar variations of fuel assembly characteristics such as fuel enrichment and integral neutron absorber (burnable poison), if present (e.g., gadolinia in certain fuel rods of BWR and PWR assemblies or integral fuel burnable absorber (IFBA) coatings in certain fuel rods of PWR assemblies).

Whenever reactivity equivalencing (i.e., burnup credit or credit for imbedded burnable absorbers) is employed, or if a correlation with the reactivity of assemblies in a standard core geometry is used ( $k_{\infty}$ ), such as is typically done for BWR racks, the equivalent reactivities must be evaluated in the storage rack configuration. In this latter approach, sufficient uncertainty should be incorporated into the  $k_{\infty}$  limit to account for the reactivity effects of (1) nonuniform enrichment variation in the assembly, (2) uncertainty in the calculation of  $k_{\infty}$ , and (3) uncertainty in average assembly enrichment.

If various locations in a storage rack are prohibited from containing any fuel, they should be physically or administratively blocked or restricted to non-fuel material. If the criticality safety of the storage racks relies on administrative procedures, these procedures should be explicitly identified and implemented in operating procedures and/or technical specification limits.

## 2. CRITICALITY ANALYSIS METHODS AND COMPUTER CODES

A variety of methods may be used for criticality analyses provided the cross-section data and geometric capability of the analytical model accurately represent all important neutronic and geometrical aspects of the storage racks. In general, transport methods of analysis are necessary for acceptable results. Storage rack characteristics such as boron carbide ( $B_4C$ ) particle size and thin layers of structural and neutron absorbing material (poisons) need to be carefully considered and accurately described in the analytical model. Where possible, the primary method of analysis should be verified by a second, independent method of analysis. Acceptable computer codes include, but are not necessarily limited to, the following:

- CASMO - a multigroup transport theory code in two dimensions
- NITAWL-KENO5a - a multigroup transport theory code in three dimensions, using the Monte Carlo technique
- PHOENIX-P - a multigroup transport theory code in two dimensions, using discrete ordinates
- MONK6B - a multigroup transport theory code in three dimensions, using the Monte Carlo technique
- DOT - a multigroup transport theory code in two dimensions, using discrete ordinates

Similarly, a variety of cross-section libraries is available. Acceptable cross-section libraries include the 27-group, 123-group, and 218-group libraries from the SCALE system developed by the Oak Ridge National Laboratory and the 8220-group United Kingdom Nuclear Data Library (UKNDL). However, empirical cross-section compilations, such as the Hansen-Roach library, are not acceptable for criticality safety analyses (see NRC Information Notice No. 91-26). Other computer codes and cross-section libraries may be acceptable provided they conform to the requirements of this position statement and are adequately benchmarked.

The proposed analysis methods and neutron cross-section data should be benchmarked, by the analyst or organization performing the analysis, by comparison with critical experiments. This qualifies both the ability of the analyst and the computer environment. The critical experiments used for benchmarking should include, to the extent possible, configurations having neutronic and geometric characteristics as nearly comparable to those of the proposed storage facility as possible. The Babcock & Wilcox series of critical experiments (Ref. 6) provides an acceptable basis for benchmarking storage racks with thin strong absorber panels for reactivity control. Similarly, the Babcock & Wilcox critical experiments on close-packed arrays of fuel (Ref. 7) provide an acceptable experimental basis for benchmark analyses for consolidated fuel arrays. A comparison with methods of analysis of similar sophistication (e.g., transport theory) may be used to augment or extend the range of applicable critical experiment data.

The benchmarking analyses should establish both a bias (defined as the mean difference between experiment and calculation) and an uncertainty of the mean with a one-sided tolerance factor for 95-percent probability at the 95-percent confidence level (Ref. 8).

The maximum  $k_{\text{eff}}$  shall be evaluated from the following expression:

$$k_{\text{eff}} = k(\text{calc}) + \delta k(\text{bias}) + \delta k(\text{uncert}) + \delta k(\text{burnup}),$$

where

$k(\text{calc})$	= calculated nominal value of $k_{\text{eff}}$ ,
$\delta k(\text{bias})$	= bias in criticality analysis methods,
$\delta k(\text{uncert})$	= manufacturing and calculational uncertainties, and
$\delta k(\text{burnup})$	= correction for the effect of the axial distribution in burnup, when credit for burnup is taken.

A bias that reduces the calculated value of  $k_{\text{eff}}$  should not be applied. Uncertainties should be determined for the proposed storage facilities and fuel assemblies to account for tolerances in the mechanical and material specifications. An acceptable method for determining the maximum reactivity may be either (1) a worst-case combination with mechanical and material conditions set to maximize  $k_{\text{eff}}$  or (2) a sensitivity study of the reactivity effects of tolerance variations. If used, a sensitivity study should include all possible significant variations (tolerances) in the material and mechanical specifications of the racks; the results may be combined statistically provided they are independent variations. Combinations of the two methods may also be used.

### 3. ABNORMAL CONDITIONS AND THE DOUBLE-CONTINGENCY PRINCIPLE

The criticality safety analysis should consider all credible incidents and postulated accidents. However, by virtue of the double-contingency principle, two unlikely independent and concurrent incidents or postulated accidents are beyond the scope of the required analysis. The double-contingency principle means that a realistic condition may be assumed for the criticality analysis in calculating the effects of incidents or postulated accidents. For example, if soluble boron is normally present in the spent fuel pool water, the loss of soluble boron is considered as one accident condition and a second concurrent accident need not be assumed. Therefore, credit for the presence of the soluble boron may be assumed in evaluating other accident conditions.

### 4. NEW FUEL STORAGE FACILITY (VAULT)

Normally, fresh fuel is stored temporarily in racks in a dry environment (new fuel storage vault) pending transfer into the spent fuel pool and then into the reactor core. However, moderator may be introduced into the vault under abnormal situations, such as flooding or the introduction of foam or water mist (for example, as a result of fire fighting operations). Foam or mist affects the neutron moderation in the array and can result in a peak in reactivity at low moderator density (called "optimum" moderation, Ref. 9). Therefore, criticality safety analyses must address two independent accident conditions, which should be incorporated into plant technical specifications:

- a. With the new fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with pure water, the maximum  $k_{\text{eff}}$  shall be no greater than 0.95, including

mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.

- b. With the new fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with moderator at the (low) density corresponding to optimum moderation, the maximum  $k_{\text{eff}}$  shall be no greater than 0.98, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.

An evaluation need not be performed for the new fuel storage facility for racks flooded with low-density or full-density water if it can be clearly demonstrated that design features and/or administrative controls prevent such flooding.

Under the double-contingency principle, the accident conditions identified above are the principle conditions that require evaluation. The simultaneous occurrence of other accident conditions need not be considered.

Usually, the storage racks in the new fuel vault are designed with large lattice spacing sufficient to maintain a low reactivity under the accident condition of flooding. Specific calculations, however, are necessary to assure the limiting  $k_{\text{eff}}$  is maintained no greater than 0.95.

At low moderator density, the presence of relatively weak absorber material (for example, stainless steel plates or angle brackets) is often sufficient to preclude neutronic coupling between assemblies, and to significantly reduce the reactivity. For this reason, the phenomenon of low-density (optimum) moderation is not significant in racks in the *spent fuel pool* under the initial conditions before the pool is flooded.

Under low-density moderator conditions, neutron leakage is a very important consideration. The new fuel storage racks should be designed to contain the highest enrichment fuel assembly to be stored without taking credit for any nonintegral neutron absorber. In the evaluation of the new fuel vaults, fuel assembly and rack characteristics upon which subcriticality depends should be explicitly identified (e.g., fuel enrichment and the presence of steel plates or braces).

## 5. SPENT FUEL STORAGE RACKS

### A. Reference Criticality Safety Analysis

1. For BWR pools or for PWR pools where no credit for soluble boron is taken, the criticality safety analyses must address the following condition, which should be incorporated into the plant technical specifications:
  - a. With the spent fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with full-density unborated water, the maximum  $k_{\text{eff}}$  shall be less than or equal to 0.95, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.

2. If partial credit for soluble boron is taken, the criticality safety analyses for PWRs must address two independent conditions, which should be incorporated into the plant technical specifications:
  - a. With the spent fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with full-density unborated water, the maximum  $k_{\text{eff}}$  shall be less than 1.0, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.
  - b. With the spent fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with full density water borated to [ \* ] ppm, the maximum  $k_{\text{eff}}$  shall be no greater than 0.95, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.<sup>1</sup>
3. The reference criticality safety analysis should also include, as a minimum, the following:
  - a. If axial and planar variations of fuel assembly characteristics are present, they should be explicitly addressed, including the locations of burnable poison rods.
  - b. For fuel assemblies containing burnable poison, the maximum reactivity should be the peak reactivity over burnup, usually when the burnable poison is nearly depleted.
  - c. The spent fuel storage racks should be assumed to be infinite in the lateral dimension or to be surrounded by a water reflector and concrete or structural material as appropriate to the design. The fuel may be assumed to be infinite in the axial dimension, or the effect of a reflector on the top and bottom of the fuel may be evaluated.
  - d. The evaluation of normal storage should be done at the temperature (water density) corresponding to the highest reactivity. In poisoned racks, the highest reactivity will usually occur at a water density of 1.0 (i.e., at 4°C). However, if the temperature coefficient of reactivity is positive, the evaluation should be done at the highest temperature expected during normal operations: i.e., equilibrium temperature under normal refueling conditions (including full-core offload), with one coolant train out of service and the pool filled with spent fuel from previous reloads.
4. The fuel assembly arrangement assumed in the criticality safety analysis of the spent fuel storage racks should also consider the following:

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<sup>1</sup> [ \* ] is the boron concentration required to maintain the 0.95 $k_{\text{eff}}$  limit without consideration of accidents.

- a. the effect of eccentric positioning of fuel assemblies within the storage cells
  - b. the reactivity consequence of including the flow channel in BWR fuel assemblies
5. If one or more separate regions are designated for the storage of spent fuel, with credit for the reactivity depletion due to fuel burnup, the following applies.
- a. The minimum required fuel burnup should be defined as a function of the initial nominal enrichment.
  - b. The spent fuel storage rack should be evaluated with spent fuel at the highest reactivity following removal from the reactor (usually after the decay of xenon-135). Operating procedures should include provision for independent confirmation of the fuel burnup, either administratively or experimentally, before the fuel is placed in storage cells of the designated region(s).
  - c. Subsequent decay of longer-life nuclides, such as Pu-241, over the rack storage time may be accounted for to reduce the minimum burnup required to meet the reactivity requirements.
  - d. A reactivity uncertainty due to uncertainty in the fuel depletion calculations should be developed and combined with other calculational uncertainties. In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption.
  - e. A correction for the effect of the axial distribution in burnup should be determined and, if positive, added to the reactivity calculated for uniform axial burnup distribution.

**B. Additional Considerations**

1. The reactivity consequences of incidents and accidents such as (1) a fuel assembly drop and (2) placement of a fuel assembly on the outside and immediately adjacent to a rack must be evaluated. Under the double-contingency principle, credit for soluble boron, if present, is acceptable for these postulated accident conditions.
2. If either credit for burnup is assumed or racks of different enrichment capability are in the same fuel pool, fuel assembly misloadings must be considered. Normally, a misloading error involving only a single assembly need be considered unless there are circumstances that make multiple loading errors credible. Under the double-contingency principle, credit for soluble boron, if present, is acceptable for these postulated accident conditions.

3. The analysis must also consider the effect on criticality of natural events (e.g., earthquakes) that may deform, and change in the relative position of, the storage racks and fuel in the spent fuel pool.
4. Abnormal temperatures (above those normally expected) and the reactivity consequences of void formation (boiling) should be evaluated to consider the effect on criticality of loss of all cooling systems or coolant flow, unless the cooling system meets the single-failure criterion. Under the double-contingency principle, credit for soluble boron, if present, is acceptable for these abnormally elevated temperature conditions.
5. Normally, credit may only be taken for neutron absorbers that are an integral (nonremovable) part of a fuel assembly or the storage racks. Credit for added absorber (rods, plates, or other configurations) will be considered on a case-by-case basis, provided it can be clearly demonstrated that design features prevent the absorbers from being removed, either inadvertently or intentionally without unusual effort such as the necessity for special equipment maintained under positive administrative control.
6. If credit for soluble boron is taken, the minimum required pool boron concentration (typically, the refueling boron concentration) should be incorporated into the plant technical specifications or operating procedures. A boron dilution analysis should be performed to ensure that sufficient time is available to detect and suppress the worst dilution event that can occur from the minimum technical specification boron concentration to the boron concentration required to maintain the  $0.95k_{\text{eff}}$  design basis limit. The analysis should consider all possible dilution initiating events (including operator error), dilution sources, dilution flow rates, boration sources, instrumentation, administrative procedures, and piping. This analysis should justify the surveillance interval for verifying the technical specification minimum pool boron concentration.
7. Consolidated fuel assemblies usually result in low values of reactivity (undermoderated lattice). Nevertheless, criticality calculations, using an explicit geometric description (usually triangular pitch) or as near an explicit description as possible, should be performed to assure a  $k_{\text{eff}}$  less than 0.95.

## 6. REFERENCES

1. NRC, "Standard Review Plan," NUREG-0800, Rev.2, Section 9.1.1, "New Fuel Storage," July 1981.
2. NRC, "Standard Review Plan," NUREG-0800, Rev. 2, Section 9.1.2, "Spent Fuel Storage," July 1981.
3. Brian K. Grimes, NRC, letter to all power reactor licensees, with enclosure, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978.

4. *Code of Federal Regulations*, Title 10, Part 50, Appendix A, General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling."
5. Westinghouse Electric Corporation, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," WCAP-14416-NP-A, November 1996.
6. Babcock & Wilcox Company, "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel," BAW-1484-7, July 1979.
7. Babcock & Wilcox Company, "Critical Experiments Supporting Underwater Storage of Tightly Packed Configurations of Spent Fuel Pins," BAW-1645-4, 1981.
8. National Bureau of Standards, *Experimental Statistics*, Handbook 91, August 1963.
9. J. M. Cano, R. Caro, and J. M. Martinez Val, "Supercriticality Through Optimum Moderation in Nuclear Fuel Storage," *Nuclear Technology*, Volume 48, May 1980.