UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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_{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_∞), 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_∞ limit to account for the reactivity effects of (1) nonuniform enrichment variation in the assembly, (2) uncertainty in the calculation of k_∞, 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_{eff} shall be evaluated from the following expression:

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

where

k(calc)	= calculated nominal value of k_{eff},
δk(bias)	= bias in criticality analysis methods,
δk(uncert)	= manufacturing and calculational uncertainties, and
δk(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_{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_{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_{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_{eff} shall be no greater than 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_{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_{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_{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 activity and flooded with full density water borated to [*] ppm, the maximum k_{eff} shall be no greater than 0.95, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.[1]

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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[1] [*] is the boron concentration required to maintain the 0 95k_{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 $0.95k_{eff}$ 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_{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. Westingnouse 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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 25, 1996

Mr. Tom Greene, Chairman
Westinghouse Owners Group
Westinghouse Electric Corporation
P.O. Box 355
Pittsburgh, PA 15230-0355

Dear Mr. Greene:

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT WCAP-14416-P, "WESTINGHOUSE SPENT FUEL RACK CRITICALITY ANALYSIS METHODOLOGY" (TAC NO. M93254)

The staff has reviewed the topical report submitted by the Westinghouse Owners Group by letter dated July 28, 1995, and supplemented by letter dated October 18, 1996. The report is acceptable for referencing in license applications to the extent specified and under the limitations stated in the enclosed U.S. Nuclear Regulatory Commission (NRC) evaluation. The evaluation defines the basis for acceptance of the report.

The staff will not repeat its review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented applies to the specific plant involved. NRC acceptance applies only to the matters described in the report. In accordance with procedures established in NUREG-0390, the NRC requests that the Westinghouse Owners Group publish accepted versions of the report, proprietary and non-proprietary, within 3 months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed evaluation between the title page and the abstract and an -A (designating accepted) should follow the report identification symbol.

If the NRC's criteria or regulations change so that its conclusion that the report is acceptable is invalidated, the Westinghouse Owners Group and/or the applicant referencing the topical report will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the topical report without revision of the respective documentation.

EXHIBIT
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dcl 11/4/96

Sincerely,

Timothy E. Collins, Acting Chief
Reactor Systems Branch
Division of Systems Safety and Analysis

Enclosure:
WCAP-14416-P Evaluation

CPL 01920002

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENCLOSURE

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO TOPICAL REPORT WCAP-14416-P

"WESTINGHOUSE SPENT FUEL RACK CRITICALITY ANALYSIS METHODOLOGY"

WESTINGHOUSE ELECTRIC CORPORATION

1.0 INTRODUCTION

In a submittal of July 28, 1995 (Ref. 1), the Westinghouse Owners Group (WOG) requested U.S. Nuclear Regulatory Commission (NRC) review and approval of topical report WCAP-14416-P, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," June 1995 (Ref. 2). The report presents the current Westinghouse methodology for calculating the effective multiplication factor, k_{eff}, of spent fuel storage racks in which no credit is taken for soluble boron except under accident conditions. The report also presents a new proposed procedure for crediting soluble boron in the spent fuel pool water when performing storage rack criticality analysis for Westinghouse fuel storage pools. A revision to the methodology was submitted on October 23, 1996 (Ref. 28), based on recommendations by the NRC Committee to Review Generic Requirements (CRGR).

General Design Criterion (GDC) 62 (Ref. 3) states that "criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations." The NRC has established a 5-percent subcriticality margin (k_{eff} no greater than 0.95) to comply with GDC 62 (Ref 4). All of the applicable biases and uncertainties should be combined with k_{eff} to provide a one-sided, upper tolerance limit on k_{eff} such that the true value will be less than the calculated value with a 95-percent probability at a 95-percent confidence level (Ref. 5). The proposed new methodology would permit the use of spent fuel pool soluble boron to offset these uncertainties to maintain k_{eff} less than or equal to 0.95. However, the spent fuel rack k_{eff} calculation would remain less than 1.0 (subcritical) when flooded with unborated water with a 95-percent probability at a 95-percent confidence level.

2.0 SUMMARY OF THE TOPICAL REPORT

Section 1.0 of the report is an introduction, stating the purpose of the report and summarizing the individual sections. Section 2.0 explains the computer codes used in the evaluation of the spent fuel rack k_{eff} calculations and presents benchmark results. In Section 3.0, the assumptions used to model the spent fuel storage racks and the reactivity effects of biases and uncertainties are presented. Section 4.0 discusses reactivity equivalencing

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methods that credit fuel assembly burnup and integral fuel burnable absorbers (IFBA). Section 5.0 describes postulated accidents that are considered in the spent fuel rack criticality analysis. Section 6.0 of the report, in conjunction with the supplement (Ref. 28), defines how credit for spent fuel pool soluble boron will be applied in the reactivity calculations.

3.0 TECHNICAL EVALUATION

The Westinghouse spent fuel rack criticality analysis methodology presented in WCAP-14416-P, and modified by Reference 28, provides a detailed description of both the current methodology, which has been used for many years by Westinghouse to calculate the reactivity of spent fuel storage racks, and a proposed new methodology with which partial credit for soluble boron in the pool water would be taken. The review of the proposed new methodology, given in Section 3.7 below, focused on the approximations and assumptions used as well as on revised technical specifications and analysis of dilution events required when crediting boron. The following evaluation is based on the material presented in the topical report, supplementary information (Ref. 28), discussions with Westinghouse staff, and responses to our requests for additional information (Refs. 14 and 26).

3.1 Computer Code Methods and Benchmarking

Reactivity calculations for the spent fuel storage racks are performed with the KENO-Va (Ref. 6) three-dimensional Monte Carlo computer code. A 227 energy group cross section library is created by NITAWL-II (Ref. 7) and XSDRNPM-S (Ref. 8) from ENDF/B-V data (Ref. 9). This method has been used to analyze a set of 32 low-enriched, water-moderated, UO_2 critical experiments to establish a method bias and uncertainty (Refs. 10, 11, 12, 13). These experiments cover a range of enrichments varying from 2.35 weight percent (w/o) to 4.31 w/o U^{235} separated by various materials (B_4C, borated aluminum, stainless steel, water) at fuel rod spacings from 0 to 6.56 cm. These experiments simulate current PWR spent fuel storage racks as realistically as possible with respect to parameters important to reactivity such as enrichment, assembly spacing, and neutron absorber worth. In response to a staff question (Ref. 14), WOG stated that no significant biases or trends were observed as a function of lattice or fuel parameters, including enrichment. The staff concludes that the KENO-Va benchmarking data is sufficiently diverse to establish that the method bias and uncertainty will apply to spent fuel storage rack conditions similar to those currently in use containing fuel rod enrichments up to 5.0 w/o U^{235}.

To minimize the statistical uncertainty of the KENO-Va calculations, at least 100,000 neutron histories are accumulated in each calculation. Experience has shown that this number of histories is sufficient to assure convergence of KENO-Va reactivity calculations. In addition, edits from the KENO-Va calculations provide a visual inspection of the overall convergence of the results.

A method bias of 0.0077 results from the comparison of KENO-Va calculations with the average measured experimental k_{eff}. The standard deviation of the bias value is 0.00136 Δk. The 95-percent probability/95-percent confidence

level (95/95) one-sided tolerance limit factor for 32 values is 2.20 (Ref. 15). Thus, there is a 95-percent probability with a 95-percent confidence level that the uncertainty in reactivity due to the method is not greater than 0.0030 Δk (2.20 x 0.00136).

The PHOENIX-P (Ref. 16) transport theory computer code is used to determine reactivity changes due to possible variations (tolerances) in material characteristics and mechanical dimensions in the fuel assembly and spent fuel racks, changes in pool conditions such as temperature and soluble boron, and fuel burnup. PHOENIX-P is a depletable, two-dimensional, multigroup, discrete-ordinates transport theory code that uses a 42 energy group nuclear data library.

PHOENIX-P has been compared with critical experiments (Refs. 17, 18, 19, 20). The PHOENIX-P reactivity predictions agree very well with the critical experiments, showing no significant bias or trends as a function of lattice or fuel parameters. The range of lattice parameters and configurations in the critical experiments encompassed present fuel storage configurations as realistically as possible.

PHOENIX-P has also been compared with isotopic measurements of fuel discharged from Yankee Core 5 (Ref. 21). The PHOENIX-P predictions agree very well with measurements for all measured isotopes throughout the burnup range.

Based on the above, we conclude that the analysis methods described are acceptable and capable of predicting the reactivity of PWR spent fuel storage racks containing assemblies with maximum fuel rod enrichments of 5.0 w/o U^{235} with a high degree of confidence.

3.2 KENO-Va Reactivity Calculations

KENO-Va is used to establish a nominal reference reactivity, using fresh (unirradiated) fuel assemblies and nominal rack dimensions, that satisfies the 0.95 k_{eff} acceptance criterion. The following assumptions are used in the calculation:

(1) The nominal spent fuel rack storage cell dimensions are used.

(2) Fuel assembly parameters for all assembly types considered for storage in the spent fuel pool are evaluated. These parameters include number of fuel rods per assembly, fuel rod clad material, fuel rod clad outer diameter, fuel rod clad thickness, fuel pellet outer diameter, fuel pellet density, fuel pellet dishing factor, fuel rod pitch, control rod guide tube material, number of guide tubes, guide tube outer diameter, guide tube thickness, instrument tube material, number of instrument tubes, instrument tube outer diameter, and instrument tube thickness.

(3) The nominal fresh fuel enrichment loaded into each fuel pin is modeled. The pin locations within a fuel assembly with multiple enrichments are considered, if applicable. The maximum fuel rod enrichment loaded into the fuel rods is limited to 5.0 w/o U^{235}.

(4) The nominal values for theoretical density and dishing fraction of the fuel pellets are modeled.

(5) If axial blankets are modeled, the length and enrichment of the blanket fuel pellets are considered.

(6) No amount of U^{234} or U^{236} is modeled in the fuel pellet.

(7) No amount of material from spacer grids or spacer sleeves is modeled in the fuel assembly.

(8) No amount of burnable absorber poison material is modeled in the fuel assembly.

(9) No amount of fission product poison material is modeled in the fuel assembly.

(10) The moderator is pure water (no boron) at a temperature of 68°F and a density of 1.0 gm/cc.

(11) If credit is taken for any fixed neutron-absorbing poison material panels present (except Boraflex), they are modeled using the as-built or manufacturer-specified poison material loadings and dimensions. Because of the significant Boraflex deterioration observed in some spent fuel racks, additional conservative assumptions are required for racks containing Boraflex as neutron absorber. These assumptions are not part of this technical review but will be reviewed on a case-by-case basis.

(12) If all storage cells are not loaded with the same fuel assembly type and enrichment, the specific storage configuration will be modeled. Different types of configurations include checkerboard patterns, empty cell locations, specific pool configurations, and other layouts as defined.

Using these assumptions, the spent fuel rack k_{eff} is calculated with KENO-Va to show that k_{eff} is less than or equal to 0.95 with no credit for soluble boron. A temperature bias, which accounts for the normal operational temperature range of the spent fuel pool water, and the method bias, determined from the benchmarking calculations, are included. In addition, if neutron absorber panels are used, a reactivity bias is added to correct for the modeling assumption that individual B^{10} atoms are homogeneously distributed within the absorber material rather than clustered around each B_4C particle. The staff concludes that these assumptions tend to maximize the rack reactivity and are, therefore, appropriately conservative and acceptable.

3.3 PHOENIX-P Tolerance/Uncertainty Calculations

PHOENIX-P is used to calculate the reactivity effects of possible variations in material characteristics and mechanical/manufacturing dimensions. The following tolerances and uncertainties are considered:

(1) Enrichment tolerance of ±0.05 w/o U^{235} about the nominal fresh reference enrichments

(2) Variation of ±2.0% about the nominal reference UO_2 theoretical density

(3) Variation in fuel pellet dishing fraction from 0% to twice the nominal dishing

(4) Tolerance about the nominal reference storage cell inner diameter, center-to-center pitch, and material thickness

(5) Tolerances about the nominal width, length, and thickness of neutron absorber panels

(6) Tolerances about the nominal poison loading of the neutron absorbing panels, if the nominal poison loading assumed in the KENO-Va model is not the minimum manufacturer-specified loading

(7) Asymmetric positioning of fuel assemblies within the storage cells

The manufacturing tolerance uncertainties are based on the reactivity difference between nominal and maximum tolerance values and, therefore, meet the 95/95 probability/confidence level requirement. These uncertainties are combined statistically with the 95/95 calculation uncertainty on the KENO-Va nominal reference k_{eff} and the 95/95 methodology uncertainty (0.0030 Δk) in the benchmarking bias determined for the KENO-Va methodology. The methodology benchmarking bias of 0.0077 Δk, the water temperature bias, and the B^{10} self-shielding bias, if applicable, are included in the final k_{eff} summation before comparison against the 0.95 k_{eff} limit. The following formula is used to determine the 95/95 k_{eff} for the spent fuel storage racks:

$$k_{eff} = k_{nominal} + B_{method} + B_{temp} + B_{self} + B_{uncert}$$

where:

$k_{nominal}$ = nominal conditions KENO-Va k_{eff}

B_{method} = method bias determined from benchmark critical comparisons

B_{temp} = temperature bias

B_{self} = B^{10} self-shielding bias, if applicable

$$B_{uncert} = \sqrt{\sum (\text{tolerance}_i \ldots \text{or} \ldots \text{uncertainty}_i)^2}$$

The staff concludes that the final k_{eff} calculated using the above methodology will satisfy the NRC guidance that the fuel storage rack reactivity be less than or equal to 0.95 when fully flooded with unborated water, including all appropriate uncertainties at the 95/95 probability/confidence level (Refs. 4, 5). Therefore, the documented methodology is acceptable.

3.3 Fuel Assembly Burnup Credit

Reactivity equivalencing is used to allow storage of fuel assemblies with higher initial enrichments (up to 5.0 w/o U^{235}) than those found acceptable using the previously described methodology. This concept is predicated upon the reactivity decrease associated with fuel depletion. For burnup credit, a series of reactivity calculations are performed with PHOENIX-P to generate a set of initial enrichment versus fuel assembly discharge burnup ordered pairs that all yield an equivalent k_{eff} (no greater than 0.95) when fuel assemblies are stored in the spent fuel storage racks.

The CINDER computer code (Ref. 22) was used to determine the most reactive time after reactor shutdown of an irradiated fuel assembly. CINDER is a point-depletion code that has been widely used and accepted in the nuclear industry to determine fission product activities. The fission products were permitted to decay for 30 years after shutdown and the fuel reactivity was found to reach a maximum at approximately 100 hours. At this time, the major fission product poison, Xe^{135}, has nearly completely decayed away. Therefore, the most reactive time for an assembly after shutdown of the reactor can be conservatively approximated by removing the Xe^{135}.

An uncertainty associated with the depletion of the fuel assembly and the reactivities computed with PHOENIX-P is accounted for in determining the reactivity equivalence limits. This uncertainty is based on the PHOENIX-P comparisons to the measured isotopics from the Yankee Core 5 experiments and is used to account for any depletion history effects or calculational uncertainties not included in the depletion conditions that are used in PHOENIX-P. The staff concludes that this uncertainty, which increases linearly with burnup from 0 at 0 burnup to 0.02 Δk at an assembly average burnup of 60,000 MWD/MTU, is conservative and acceptable.

The effect of axial burnup distribution on fuel assembly reactivity has been evaluated by modeling depleted fuel in both two dimensions and three dimensions. These evaluations show that axial burnup effects can cause assembly reactivity to increase at burnup-enrichment combinations greater than 40,000 MWD/MTU and 4.0 w/o U^{235}. Westinghouse has stated that this effect will be accounted for as an additional bias if burnup credit limits reach these combinations.

An additional conservatism is that the depletion calculations do not take credit for effects, such as Pu^{241} decay and Am^{241} growth, that are known to substantially reduce reactivity during long-term storage. However, the staff does not consider this to be a requirement.

The staff concludes that adequate conservatism has been incorporated in the methodology used to determine burnup credit.

3.4 Integral Fuel Burnable Absorber (IFBA) Credit

Another reactivity equivalencing technique for storage of fuel enrichments greater than those allowed by the previous methodology is based on the reactivity decrease associated with the addition of integral fuel burnable absorbers (IFBA) to Westinghouse fuel. IFBAs consist of neutron-absorbing material applied as a nonremovable thin zirconium diboride (ZrB_2) coating on the outside of the UO_2 pellet. PHOENIX-P is used to generate a set of initial assembly enrichment versus number of IFBA rods per assembly ordered pairs that all yield the equivalent k_{eff} (no greater than 0.95) when fuel assemblies are stored in the spent fuel storage racks. The following assumptions are used for the IFBA rod assemblies in the PHOENIX-P calculations:

(1) The fuel assembly is modeled at its most reactive point in life. This includes any time in life when the IFBA has depleted and the fuel assembly becomes more reactive.

(2) The B^{10} loading for each IFBA rod, determined from Westinghouse IFBA design specifications for the given fuel assembly type, is the minimum standard loading offered by Westinghouse for that fuel assembly type.

(3) The IFBA B^{10} loading is reduced by 5 percent to account for manufacturing tolerances and by an amount which corresponds to the minimum absorber length offered for the given fuel assembly type (e.g., a 144-inch fuel length with a minimum absorber length of 108 inches would result in a 25 percent IFBA B^{10} loading).

A calculational uncertainty of approximately 10 percent is included in the development of the IFBA requirements by adding an additional number of IFBA rods to each data point. To demonstrate that reactivity margin exists in the IFBA credit limit to accommodate future changes in IFBA patterns, calculations are also performed with nonstandard IFBA patterns. If a future change is made to the standard IFBA pattern designs, the reactivity difference between the new patterns and the old patterns will be calculated in order to assess the impact on both core reactivity and spent fuel rack IFBA credit limits.

The staff concludes that adequate conservatism has been incorporated in the methodology for determining IFBA requirements and that assemblies that comply with the enrichment-IFBA requirement curve developed by this methodology will have a k_{eff} no greater than 0.95 when placed in the spent fuel pool storage racks.

3.5 Infinite Multiplication Factor

An alternative method for determining the acceptability of fuel storage in a specific spent fuel rack is based on a PHOENIX-P calculation of the infinite multiplication factor (k_∞) for a fuel assembly in the reactor core geometry as a reference point. The fuel assembly model is based on a unit assembly configuration (infinite in the lateral and axial dimensions) in reactor geometry and is modeled at its most reactive point in life and moderated by

pure water (no boron) at a temperature of 68°F with a density of 1.0 g/cc. A 0.01 Δk reactivity bias is added to this reference k_∞ to account for calculational uncertainties. The spent fuel storage rack is then modeled with these assemblies to ensure that the storage rack reactivity will be no greater than 0.95.

The staff concludes that fuel assemblies that have a reference k_∞ less than or equal to the value calculated with the above assumptions and methodology will have a k_{eff} no greater than 0.95 when placed in the spent fuel pool storage racks.

3.6 Postulated Accidents

The criterion that k_{eff} be no greater than 0.95 exists even for postulated accidents. Two types of accidents that can occur in a spent fuel storage rack may cause a reactivity increase: (1) a fuel assembly misplacement and (2) a pool water temperature change. However, for any of these accidents, the double contingency principle (Ref. 23) can be applied. According to this principle, it is unnecessary to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these postulated accidents, the presence of soluble boron in the pool water can be assumed as a realistic initial condition since assuming its absence would be a second unlikely event. PHOENIX-P boron worth calculations are used to determine the amount of soluble boron required to offset the highest reactivity increase caused by any postulated accident and to maintain k_{eff} less than or equal to 0.95, which is also the staff's acceptance criterion for accident conditions.

3.7 Soluble Boron Credit Methodology

In the proposed methodology for performing spent fuel rack reactivity calculations with credit for soluble boron in the pool water, a 95/95 rack k_{eff} is first calculated which remains below 1.0 (subcritical) with no soluble boron credit. This k_{eff} calculation uses the same assumptions described in Section 3.2 above, including the assumption of no soluble boron in the pool water. As previously described, a temperature bias, a method bias, a B^{10} self-shielding bias, and the 95/95 uncertainties associated with the calculation uncertainty, the methodology uncertainty in the benchmarking bias, and the manufacturing tolerances are included in the k_{eff} calculation.

The final equation for determining the k_{eff} requirement is

$$k_{eff} = k_{nominal} + B_{temp} + B_{method} + B_{self} + B_{uncert} < 1.0$$

where:

$k_{nominal}$ = nominal condition KENO-Va k_{eff}

B_{temp} = temperature bias for normal operating range

B_{method} = method bias from benchmark critical comparisons

B_{self} = B^{10} self shielding bias

$$B_{uncert} = \sqrt{\sum(\text{tolerance}_i \ldots \text{or} \ldots \text{uncertainty}_i)^2}$$

To determine the amount of soluble boron required to maintain $k_{eff} \leq 0.95$, KENO-Va is used to establish a nominal reference k_{eff} and PHOENIX-P is used to evaluate the reactivity effects of possible variations in material characteristics and mechanical manufacturing dimensions. These calculations contain the same assumptions, biases, tolerances, and uncertainties previously described except for the assumption regarding the moderator soluble boron concentration. Borated water is assumed instead of pure water. The tolerance calculations are, therefore, performed assuming the presence of soluble boron. The final 95/95 k_{eff} calculation is determined as described in Section 3.2 above and must be less than or equal to 0.95 with allowances for biases, tolerances, and uncertainties including the presence of the determined concentration of soluble boron.

For enrichments higher than those assumed in the k_{eff} calculation, reactivity equivalencing methodologies are used to determine burnup or IFBA credit. However, the maximum fuel rod enrichment is limited to 5.0 w/o U^{235}. Soluble boron credit is used to offset the uncertainties associated with each of these equivalencing methodologies, as appropriate.

Postulated accidents are considered in the same manner as discussed in Section 3.6 except that the previously determined amount of soluble boron for the 95/95 k_{eff} calculation, plus the amount determined for the reactivity equivalencing calculation, if required, is assumed present. The results of PHOENIX-P calculations of the reactivity change due to the presence of soluble boron are used to determine the amount of soluble boron required to offset the maximum reactivity increase caused by postulated accident conditions.

The final soluble boron credit requirement is determined from the following summation:

$$SBC_{TOTAL} = SBC_{95/95} + SBC_{RE} + SBC_{PA}$$

where:

SBC_{TOTAL} = total soluble boron credit requirement (ppm)

$SBC_{95/95}$ = soluble boron credit required for 95/95 k_{eff} less than or equal to 0.95 (ppm)

SBC_{RE} = soluble boron credit required for reactivity equivalencing methodologies (ppm)

SBC_{PA} = soluble boron credit required for k_{eff} less than or equal to 0.95 under accident conditions (ppm)

Thus the total soluble boron credit requirement will maintain the spent fuel rack k_{eff} less than or equal to 0.95 with a 95-percent probability at a 95-percent confidence level.

The total soluble boron required to maintain k_{eff} less than or equal to 0.95 is normally well below the large amount of soluble boron which is typically in spent fuel pool water. Therefore, a significant margin to criticality would generally still exist. However, a boron dilution analysis will be performed for each plant requesting soluble boron credit to ensure that sufficient time is available to detect and mitigate the dilution before the 0.95 k_{eff} design basis is exceeded and submitted to the NRC for review (Ref. 29). The analysis should include an evaluation of the following plant-specific features:

1. Spent Fuel Pool and Related System Features
   a) dilution sources
   b) dilution flow rates
   c) boration sources
   d) instrumentation
   e) administrative procedures
   f) piping
   g) loss of offsite power impact

2. Boron Dilution Initiating Events (including operator error)

3. Boron Dilution Times and Volumes

4.0 SUMMARY AND CONCLUSIONS

The topical report WCAP-14416-P and supporting documentation provided in References 14, 26 and 28 have been reviewed in detail. A major portion of this review focused on a proposed new methodology whereby partial credit could be taken for soluble boron in the spent fuel pool to meet the NRC-recommended criterion that the spent fuel rack multiplication factor (k_{eff}) be less than or equal to 0.95, at a 95-percent probability, 95-percent confidence level.

The staff concludes that the proposed new methodology for soluble boron credit is acceptable for the following reasons:

(1) Uncertainties in mechanical tolerances and storage rack dimensions are determined at the 95-percent probability, 95-percent confidence level and are incorporated in a conservative direction.

(2) Conservative uncertainties are incorporated for depletion calculations.

(3) A substantial margin to criticality would be available since the spent fuel rack k_{eff} will be less than or equal to 0.95, at a 95-percent probability, 95-percent confidence level, with an amount of soluble boron significantly less than that amount normally available in the pool.

(4) The fuel rack k_{eff}, will remain less than 1.0 (subcritical), at a 95-percent probability, 95-percent confidence level, even with no soluble boron in the spent fuel pool, thereby conforming to Criterion 62, "Prevention of criticality in fuel storage and handling" of Appendix A to 10 CFR Part 50.

The staff concludes that the methodology documented in WCAP-14416-P and Reference 28 can be used in licensing actions with the following provisions which are stated in WCAP-14416-P and Reference 28:

(1) If axial and planar variations of fuel assembly characteristics are present, they should be explicitly addressed, including the locations of burnable absorber rods.

(2) The maximum fuel rod enrichment shall be limited to 5.0 w/o U^{235}.

(3) The spent fuel storage racks should be assumed to be infinite in lateral extent or 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 reflector on the top and bottom of the fuel may be evaluated.

(4) If credit for the reactivity depletion due to fuel burnup is taken, operating procedures should include provision for independent confirmation of the fuel burnup, either administratively or experimentally, before the fuel is placed in burnup-dependent storage cells.

(5) A reactivity uncertainty due to uncertainty in the fuel depletion history effects and depletion calculations should be included.

(6) A correction for the effect of the axial distribution in burnup should be determined and added to the reactivity calculated for uniform axial burnup distribution if it results in a positive reactivity effect.

In addition, as stated in the letter of October 18, 1996, from Westinghouse to the NRC (Ref. 28), the following items will be submitted by all licensees proposing to use the methodology described above:

(1) All licensees proposing to use the new method described above for soluble boron credit should submit a 10 CFR Part 50.36 technical specification change containing the following:

a. k_{eff} shall be less than or equal to 0.95 if fully flooded with water borated to [1050] ppm which includes an allowance for uncertainties as described in WCAP-14416-P.

b. k_{eff} shall be less than 1.0 if fully flooded with unborated water which includes an allowance for uncertainties as described in WCAP-14416-P.

c. The spent fuel pool boron concentration shall be greater than [2300] ppm and shall be verified at a frequency of [7 days].

Licensees using the Westinghouse Improved Standard Technical Specifications (ISTS) described in NUREG-1431 (Ref. 27), should adopt specification 3.7.16, "Fuel Storage Boron Concentration," and 4.3.1, Fuel Storage-Criticality," as shown in section 5.0 below.

(2) All licensees proposing to use the new method described above for soluble boron credit should identify potential events which could dilute the spent fuel pool soluble boron to the concentration required to maintain the 0.95 k_{eff} limit (as defined in (1)a above) and should quantify the time span of these dilution events to show that sufficient time is available to enable adequate detection and suppression of any dilution event. The effects of incomplete boron mixing, such as boron stratification, should be considered. This analysis should be submitted for NRC review and should also be used to justify the surveillance interval used for verification of the technical specification minimum pool boron concentration.

(3) Although Boraflex deterioration is not addressed in this topical report, appropriate analyses are required to account for Boraflex degradation in storage racks that credit the negative reactivity effect of Boraflex. These analyses should be submitted for NRC review.

(4) Plant procedures should be upgraded, as necessary, to control pool boron concentration and water inventory during both normal and accident conditions.

5.0 TECHNICAL SPECIFICATIONS

3.7 PLANT SYSTEMS

3.7.16 Fuel Storage Pool Boron Concentration

LCO 3.7.16 The fuel storage pool boron concentration shall be ≥ [2300] ppm.

APPLICABILITY: When fuel assemblies are stored in the fuel storage pool and a fuel storage pool verification has not been performed since the last movement of fuel assemblies in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	----------NOTE---------- LCO 3.0.3 is not applicable. ----------		
	A.1	Suspend movement of fuel assemblies in the fuel storage pool.	Immediately
	AND		
	A.2.1	Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately
	OR		
	A.2.2	Verify by administrative means [Region 2] fuel storage pool verification has been performed since the last movement of fuel assemblies in the fuel storage pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify the fuel storage pool boron concentration is within limit.	[7 days]

4.0 DESIGN FEATURES

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of [4.5] weight percent;

b. $k_{eff} < 1.0$ if fully flooded with unborated water which includes an allowance for uncertainties as described in WCAP-14416-P;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to [1050] ppm which includes an allowance for uncertainties as described in WCAP-14416-P;

[d. A nominal [9.15] inch center to center distance between fuel assemblies placed in [the high density fuel storage racks];]

[e. A nominal [10.95] inch center to center distance between fuel assemblies placed in [low density fuel storage racks];]

[f. New or partially spent fuel assemblies with a discharge burnup in the "acceptable range" of Figure [3.7.17-1] may be allowed unrestricted storage in [either] fuel storage rack(s); and]

[g. New or partially spent fuel assemblies with a discharge burnup in the "unacceptable range" of Figure [3.7.17-1] will be stored in compliance with the NRC approved [specific document containing the analytical methods, title, date, or specific configuration or figure].]

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CERTIFICATE OF DEPONENT

I, LAURENCE I. KOPP, do hereby certify that I have read the foregoing transcript of my deposition testimony and, with the exception of additions and corrections, if any, hereto, find it to be a true and accurate transcription thereof.

Laurence I. Kopp

12/27/99
DATE

Sworn and subscribed to before me, this the _____ day of _____, 19_____.

_____
NOTARY PUBLIC IN AND FOR

My commission expires:

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IN THE CASE OF : Carolina Power & Light
CASE # : 99-7102-02 LA

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PAGE	LINE	ERROR/AMENDMENT	REASON FOR CHANGE
5	1	change "C." to "I."	wrong middle initial
7	15	change "1993" to "1983"	typo
8	17	change "cancel" to "canal"	"
11	11	change "simply" to "simplify"	"
22	3	change "additional" to "official"	"
22	10	change "concurrences" to "concurrences"	"
29	6	change "wasn't" to "was"	"
29	12	change "200" to "with 2000"	"
29	15	change "pucl" to "fuel"	"
38	6	change "are" to "effective"	"
38	9	change "solution" to "dilution"	"
39	7	change "plans" to "plants"	"
39	15	change "plan" to "plant"	"

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