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

November 7, 1996

MEMORANDUM TO: James M. Taylor
Executive Director for Operations

FROM: Edward L. Jordan, Chairman
Committee to Review Generic Requirements

SUBJECT: MINUTES OF CRGR MEETING NUMBER 293

The Committee to Review Generic Requirements (CRGR) met on Tuesday, October 8, 1996, from 9:00 a.m. to 12:00 p.m. A list of attendees is provided in Attachment 1. The following items were discussed at the meeting:

1. G. Holahan (NRR) presented for CRGR review and endorsement staff's evaluation and acceptance of the Westinghouse Owners Group (WOG) methodology for crediting the soluble boron in Westinghouse-designed spent fuel pools (WCAP-14416-P, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," June 1995. The staff's rationale for accepting the WOG methodology was that a significant margin would be available, and thus, the probability of an inadvertent criticality in the spent fuel pool would not be increased to an appreciable extent.

   The Committee remarked on the staff's evaluation and rationale, and on ensuing acceptance of the WOG methodology. The Committee recommended very specific changes to be made to the text of the Safety Evaluation Report (SER). The staff agreed to these changes. Additionally, the Committee noted that in addition to these changes in text of the SER, they would also need to be incorporated by WOG in the subject Topical Report (WCAP-14416-P). The staff agreed to work this out with WOG. Subsequently, WOG accepted these modifications and committed to submit a revised Topical Report incorporating the CRGR recommendations delineated in the staff's SER. Attachment 2 contains details concerning this topic.

2. CRGR was also briefed by the staff of Division of Reactor Program Management, NRR, on staff's plans for implementation of the revised accident source term at operating power plants. An option paper is planned to be sent to the Commission for consideration. No decisions or formal recommendation were made by CRGR; however, the Committee expressed an interest in being involved in review of the foundation documents associated with this important policy matter.

In accordance with the EDO's July 18, 1983 directive concerning "Feedback and Closure of CRGR Review," a written response is required from the cognizant office to report agreement or disagreement with the CRGR recommendations in these minutes. The response is to be forwarded to the CRGR Chairman and if there is disagreement with the CRGR recommendations, to the EDO for decision making.

R002

Questions concerning these meeting minutes should be referred to Raji Tripathi (415-7584).

Attachments: As stated

cc: Commission (5)
SECY
J. Lieberman, OE
P. Norry, ADM
H. Bell, OIG
K. Cyr, OGC
J. Larkins, ACRS
Office Directors
Regional Administrators, RI/RII/RIII/RIV
CRGR Members
E. W. Brach
A. Thadani

Attachment 1 to the Minutes of CRGR Meeting No. 293

Attendance List

October 8, 1996

CRGR Members

D. Ross
A. Thadani (for F. Miraglia)
M. Knapp (Part time)
J. Murphy
D. Dambly
C. W. Hehl
E. W. Brach (Part time)

CRGR Staff

R. Tripathi
J. Conran

NRC Staff

G. Holahan
R. Jones
E. Weiss
L. Kopp
C. Grimes
C. Miller
T. Collins
C. Grattow
B. Wetzel
B. Zalcman
J. Wilson
A. Huffert
J. Schaperow
J. Lee
D. Nolan (ACRS)

Attachment 2 to the Minutes of CRGR Meeting No. 293

Staff's Evaluation and Acceptance of the Westinghouse Owners Group Methodology for Crediting the Soluble Boron in Westinghouse-Designed Spent Fuel Pools

October 8, 1996

TOPIC

G. Holahan (NRR), R. Jones (NRR) and L. Kopp (NRR) presented for CRGR review and endorsement the staff's evaluation and acceptance of the Westinghouse Owners Group (WOG) methodology for crediting the soluble boron in the Westinghouse-designed spent fuel pools. The staff's rationale for accepting the WOG methodology was that a significant margin would be available, and thus, the probability of an inadvertent criticality in the spent fuel pool would not be increased to an appreciable extent.

BACKGROUND

(I) The package provided for CRGR review and endorsement for this item was transmitted by memorandum, dated August 12, 1996, from F. J. Miraglia, to E. L. Jordan, "Credit for Soluble Boron in PWR Spent Fuel Pools." The package (CRGR Review Item No. 147) was distributed to the members on August 19, 1996. It contained the following documents:

1. Staff's Safety Evaluation Report of Topical Report WCAP-14416-P (non-proprietary) as an enclosure to a letter from R. C. Jones (NRC) to R. A. Newton, Chairman, Westinghouse Owners Group, "Acceptance for referencing of Licensing Topical report WCAP-14416-P, 'Westinghouse Spent Fuel Rack Criticality Analysis Methodology,'" dated August 12, 1996.

2. Topical report WCAP-14416-P (non-proprietary)

(ii) E-mail from J. Conran to the CRGR members, dated September 13, 1996

(iii) Comments from E. L. Jordan recommending CRGR review of this item, forwarded by J. Conran to the members on or about September 25, 1996

(iv) AEOD Special Study, "Assessment of Spent Fuel Pool Cooling," (AEOD/S96-02), dated September 1996, forwarded by J. Conran to the CRGR members on October 4, 1996.

A copy of the briefing material distributed at the meeting is included as Attachment 2A.

RECOMMENDATIONS/CONCLUSIONS

The Committee remarked on the staff's evaluation and rationale, and on ensuing acceptance of the WOG methodology. The Committee recommended the following changes to be made to the text of the Safety Evaluation Report (SER):

1. Change k_{eff} less than 1.0 limit to a 95 percent probability, 95 percent confidence level value rather than a best estimate.

2. Add an enrichment limit of 5.0 weight-percent (w/o).

3. Add a statement to include operator error in considering boron dilution initiating events.

4. Add a statement to upgrade plant procedures to control boron concentration and water inventory during both normal and accident conditions.

5. Add a statement to consider the effects of incomplete boron mixing, such as boron stratification, in boron dilution analysis.

6. Add a statement to find method acceptable for extension to 5.0 weight-percent (w/o) fuel based on fact that no significant biases or trends were observed as a function of enrichment from experiments.

The staff agreed to these changes. Attachment 2B contains the revised SER.

(NOTE: The text of the revised SER in Attachment 2B contains the changes made by the staff to incorporate CRGR recommendations. Only these changes have been "red-lined" to facilitate the CRGR staff's review of the Committee's recommendations as incorporated in the SER.

It is important to note that the revised SER contains additional editorial changes made by the staff based on the NRR management's comments subsequent to the CRGR review of the SER. Thus, the embellished text included herein should not be compared, word-by-word, with the as-submitted text in the CRGR Background Material Item 1 (CRGR Review Item No. 147). Since the Committee has not re-reviewed the entire revised SER, the CRGR endorsement is only of the as-submitted text plus the "red-lined" changes made based on the CRGR recommendations.)

Additionally, the Committee noted that in addition to these changes in text of the SER, they would also need to be incorporated by WOG in the subject Topical Report (WCAP-14416-P). The staff agreed to work this out with WOG. Subsequently, WOG accepted these modifications and committed to submit a revised Topical Report incorporating the CRGR recommendations delineated in the SER.

On the basis of the staff's presentation and discussion at the meeting, as well as subsequent changes made by the staff to the text of the SER and the staff's assurance that consistent changes will be made by WOG to the Topical report WCAP-14416-P and the revised report will be re-submitted to the NRC, the Committee endorsed the SER.

BACKFIT CONSIDERATIONS

A backfit analysis is not required as the use of the WOG methodology by the licensees is voluntary.

Attachment 2A

CURRENT CRITICALITY CRITERION

- ☐ k_{eff} LESS THAN OR EQUAL TO 0.95 WHEN FLOODED WITH UNBORATED WATER, INCLUDING ALL APPROPRIATE UNCERTAINTIES AT THE 95% PROBABILITY / 95% CONFIDENCE LEVEL

PROPOSED CRITERIA

- ☐ k_{eff} LESS THAN OR EQUAL TO 0.95 WHEN FLOODED WITH BORATED WATER, INCLUDING ALL APPROPRIATE UNCERTAINTIES AT THE 95% PROBABILITY / 95% CONFIDENCE LEVEL
(INCLUDES METHOD BIAS, TEMPERATURE BIAS, METHODOLOGY UNCERTAINTY, MANUFACTURING TOLERANCE UNCERTAINTY)

- ☐ k_{eff} LESS THAN 1.0 WHEN FLOODED WITH UNBORATED WATER
(BEST ESTIMATE - INCLUDES METHOD BIAS, TEMPERATURE BIAS, NO UNCERTAINTIES)

- ☐ ADDITIONAL CONSERVATISM INCLUDED IN BEST ESTIMATE CALCULATION:

NOMINAL FRESH FUEL ENRICHMENT
NO U-234 OR U-236
NO GRID OR SPACER MATERIAL
NO BURNABLE ABSORBERS
NO FISSION PRODUCT POISON
TEMPERATURE AT 68°F AND 1.0 GM/CC
NO CREDIT FOR PU-241 DECAY OR AM-241 GROWTH
INFINITE ARRAY IN LATERAL EXTENT

REGULATORY REQUIREMENTS

- GENERAL DESIGN CRITERION 62
  (REMAINS APPLICABLE)

  CRITICALITY IN THE FUEL STORAGE AND HANDLING SYSTEM SHALL BE PREVENTED BY PHYSICAL SYSTEMS OR PROCESSES, PREFERABLY BY USE OF GEOMETRICALLY SAFE CONFIGURATIONS

- STANDARD REVIEW PLAN, SECTION 9.1.2
  (REQUIRES REVISION)

  k_{eff} NOT GREATER THAN 0.95 WHEN FULLY LOADED AND FLOODED WITH NONBORATED WATER

- LETTER TO ALL LICENSEES FROM B.K. GRIMES (4/14/78)

  k_{eff} NOT GREATER THAN 0.95, INCLUDING ALL UNCERTAINTIES, WHEN FULLY LOADED AND FLOODED WITH NONBORATED WATER

  A CALCULATIONAL BIAS AND UNCERTAINTY AND MECHANICAL UNCERTAINTIES SHALL BE DETERMINED SUCH THAT THE TRUE k_{eff} WILL BE LESS THAN THE CALCULATED VALUE WITH A 95% PROBABILITY AT A 95% CONFIDENCE LEVEL

SOLUBLE BORON CREDIT MARGIN

CAN BE USED TO OFFSET:

- PARTIAL BORAFLEX DEGRADATION
- ENRICHMENT INCREASES
- INCREASED STORAGE CAPACITY
- REQUIRED BURNUP IN 2 REGION POOLS

LICENSING SUBMITTALS

50.36 T.S. SUBMITTAL CONTAINING THE FOLLOWING

- ☐ k_{eff} NOT GREATER THAN 0.95, INCLUDING ALL 95/95 UNCERTAINTIES, WHEN FULLY LOADED AND FLOODED WITH BORATED WATER

- ☐ k_{eff} LESS THAN 1.0 WHEN FULLY FLOODED BY NONBORATED WATER UNDER BEST ESTIMATE CONDITIONS

- ☐ THE MINIMUM BORON CONCENTRATION IN THE SPENT FUEL POOL SHALL BE [ ] PPM AND SHALL BE VERIFIED AT A FREQUENCY OF [ ]

BORON DILUTION ANALYSIS

ANALYSIS MUST CONSIDER:

- ☐ INITIATING EVENTS
- ☐ POTENTIAL DILUTION SOURCES
- ☐ DILUTION FLOW RATES
- ☐ BORATION SOURCES
- ☐ INSTRUMENTATION
- ☐ ADMINISTRATIVE PROCEDURES - INCLUDING EMERGENCY PROCEDURES
- ☐ IMPACT OF LOSS OF OFFSITE POWER

RESULTS MUST SHOW:

- ☐ TIME TO DILUTE BELOW 0.95 k_{eff} BORON CONCENTRATION
- ☐ SUFFICIENT TIME FOR DETECTION & SUPPRESSION
- ☐ JUSTIFICATION OF TS BORON SURVEILLANCE INTERVAL

METHODOLOGY VALIDATION

	CRITICAL EXPERIMENTS	TYPICAL W POOL
ENRICHMENT:	1.04 TO 4.31 WT.%	5.0 WT.% (MAX)
BORON:	0 TO 3392 PPM	3250 PPM (APPROX)
SEPARATION MATERIAL:	WATER/SS/BORAL/B_4C	WATER/SS/BORAL B_4C/BORAFLEX
PELLET DIA:	0.44 TO 2.35 CM	0.89 CM
LATTICE PITCH:	0.95 TO 4.95 CM	1.41 CM

PRAIRIE ISLAND

- ☐ TOTAL NOMINAL SOLUBLE BORON IN POOL = 3250 PPM
- ☐ Keff NO GREATER THAN 0.95 WITH 1050 PPM BORON
- ☐ NO SINGLE PLANT SOURCE OF WATER CAN PROVIDE QUANTITY OF WATER NEEDED FOR DILUTION TO 1380 PPM (1050 + UNCERT)
- ☐ SINCE SUCH A LARGE WATER VOLUME WOULD BE REQUIRED, DILUTION EVENT WOULD BE READILY DETECTED
- ☐ EVALUATION INDICATES TENS OF HOURS WOULD BE AVAILABLE FOR DETECTION AND RESPONSE
- ☐ PLANT PROCEDURES WILL BE UPGRADED AS NECESSARY TO CONTROL POOL BORON CONCENTRATION AND WATER INVENTORY DURING BOTH NORMAL AND ACCIDENT SITUATIONS

Attachment 2B to the Minutes of CRGR Meeting No. 293

Revised SER incorporating the CRGR Recommendations

NOTE: The text of the revised SER on the following pages contains the changes made by the staff to incorporate CRGR recommendations. Only these changes have been "red-lined" to facilitate the CRGR staff's review of the Committee's recommendations as incorporated in the SER.

It is important to note that the revised SER contains additional editorial changes made by the staff based on the NRR management's comments subsequent to the CRGR review of the SER. Thus, the embellished text included herein should not be compared, word-by-word, with the as-submitted text in the CRGR Background Material Item 1 (CRGR Review Item No. 147). Since the Committee has not re-reviewed the entire revised SER, the CRGR endorsement is only of the as-submitted text plus the "red-lined" changes made based on the CRGR recommendations.

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.

Sincerely,

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

Enclosure:
WCAP-14416-P Evaluation

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 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 18, 1996 (Ref. 28), based on recommendations by the U.S. Nuclear Regulatory Commission (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 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, 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 for each fuel pin is modeled. The pin locations within a fuel assembly with multiple enrichments are considered, if applicable.

(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

$$B_{uncert} = \sqrt{\sum (tolerance_i \ldots or \ldots uncertainty_i)^2}$$

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

$$B_{uncert} = \sqrt{\sum(tolerance_i \ldots or \ldots uncertainty_i)^2}$$

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. 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 95/95 probability/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 calculations should be developed and combined with other calculational uncertainties.

(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
	<u>AND</u>	
	A.2.1 Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately
	<u>OR</u>	
	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 <u>unborated</u> water which includes an allowance for uncertainties as described in WCAP-14416-P;

c. $k_{eff} \leq 0.95$ if fully flooded with water <u>borated</u> 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].]

6.0 REFERENCES

1) Newton, R.A. (WOG), letter to Document Control Desk (NRC), Westinghouse Owners Group Transmittal of Reports: WCAP-14416-P (Proprietary) and WCAP-14417-NP (Non-proprietary) Entitled "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," OG-95-062, July 28, 1995.

2) Newmyer, W.D., "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," WCAP-14416-P, June 1995.

3) Code of Federal Regulations, Title 10, Part 50, Appendix A, Criterion 62, "Prevention of Criticality in Fuel Storage and Handling."

4) U.S. Nuclear Regulatory Commission, Standard Review Plan, Section 9.1.2, NUREG-0800, July 1981.

5) U.S. Nuclear Regulatory Commission, Letter to All Power Reactor Licensees from B.K. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978.

6) Petrie, L.M. and Landers, N.F., "KENO Va--An Improved Monte Carlo Criticality Program With Supergrouping," NUREG/CR-0200, Vol. 2, Section F11, November 1993.

7) Greene, N.M., "NITAWL-II: SCALE System Module for Performing Resonance Shielding and Working Library Production," NUREG/CR-0200, Vol. 2, Section F2, June 1989.

8) Greene, N.M., "XSDRNPM-S: A One-Dimensional Discrete-Ordinates Code for Transport Analysis," NUREG/CR-0200, Vol. 2, Section F3, November 1993.

9) Ford, W.E., III, "CSRL-V: Processed ENDF/B-V 227-Neutron-Group and Pointwise Cross-Section Libraries for Criticality Safety, Reactor and Shielding Studies," ORNL/CSD/TM-160, June 1982.

10) Baldwin, M.N., "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel," BAW-1484-7, July 1979.

11) Bierman, S.R., and Clayton, E.D., "Criticality Separation Between Subcritical Clusters of 2.35 Wt% ^{235}U Enriched UO_2 Rods In Water with Fixed Neutron Poisons," PNL-2438, October 1977.

12) Bierman, S.R., and Clayton, E.D., "Criticality Separation Between Subcritical Clusters of 4.29 Wt% ^{235}U Enriched UO_2 Rods in Water with Fixed Neutron Poisons," PNL-2615, August 1979.

13) Bierman, S.R., and Clayton, E.D., "Criticality Experiments with Subcritical Clusters of 2.35 Wt% and 4.31 Wt% ^{235}U Enriched UO_2 Rods in Water at a Water-to-Fuel Volume Ratio of 1.6," PNL-3314, July 1980.

14) Bush, L. (WOG), letter to Document Control Desk (NRC), Westinghouse Owners Group Transmittal of Response to Request for Additional

Information (RAI) Regarding WCAP-14416-P Entitled "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," Second Round of Questions, January 17, 1996.

15) Owen, D.B., "Factors for One-Sided Tolerance Limits and For Variables Sampling Plans," Sandia Corporation, SCR-607, March 1963.

16) Nguyen, T.Q., et. al., "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," WCAP-11597-A, June 1988.

17) Strawbridge, L.E., and Barry, R.F., "Criticality Calculations for Uniform Water-Moderated Lattices," Nuclear Science & Engineering, Vol. 23, pp. 58-73 (1965).

18) Baldwin, M.N., and Stern, M.E., "Physics Verification Program Part III, Task 4: Summary Report," BAW-3647-20, March 1971.

19) Baldwin, M.N., "Physics Verification Program Part III, Task 11: Quarterly Technical Report January-March 1974," BAW-3647-30, July 1974.

20) Baldwin, M.N., "Physics Verification Program Part III, Task 11: Quarterly Technical Report July-September 1974," BAW-3647-31, February 1975.

21) Melehan, J.B., "Yankee Core Evaluation Program Final Report," WCAP-3017-6094, January 1971.

22) England, T.R., "CINDER - A One-Point Depletion and Fission Product Program," WAPD-TM-334, August 1962.

23) American Nuclear Society, "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," ANSI/ANS-8.1-1983, October 7, 1983.

24) Kopp, L. (NRC), letter to R. Newton (WOG), Request for Additional Information on Licensing Topical Report WCAP-14416-P, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," August 30, 1995.

25) Kopp, L. (NRC), letter to L. Bush (WOG), Request for Additional Information on Licensing Topical Report WCAP-14416-P, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," Second Round of Questions, October 16, 1995.

26) Bush, L. (WOG), letter to Document Control Desk (NRC), Westinghouse Owners Group Transmittal of Response to Request for Additional Information (RAI) Regarding WCAP-14416-P Entitled "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," OG-95-080, September 28, 1995.

27) U.S. Nuclear Regulatory Commission, "Standard Technical Specifications, Westinghouse Plants," NUREG-1431, Rev. 1, April 7, 1995.

28) Andrachek, J. (W), letter to L. Kopp (NRC), October 18, 1996.

Attachment 3 to the Minutes of CRGR Meeting No. 293

Staff's Plants for Implementation of the Revised Accident Source Term at Operating Power Plants

October 8, 1996

TOPIC

C. Miller (NRR) and R. Emch (NRR) briefed CRGR on staff's plans for implementation of the revised source term at operating power plants. An option paper is planned to be sent to the Commission for consideration.

BACKGROUND

There was no review package provided by the staff. A copy of the briefing material distributed at the meeting is included as Attachment 3A.

RECOMMENDATIONS/CONCLUSIONS

No decisions ot formal recommendation were made by CRGR; however, the Committee expressed an interest in being involved in review of the foundation documents associated with this important policy matter.

CRGR Briefing

Staff's Plans for Implementation of the Revised Accident Source Term at Operating Power Plants

R. Emch/DRPM-NRR

Attachment 3A

Background

- Final NUREG-1465 Issued in February 1995
- NEI Generic Framework Document Submitted to NRC for Review in November 1995
- NRC/NEI Meetings Held in October 1994, June 1995, October 1995, January 1996, and October 1996
- Commission Paper Due November 1996
- ACRS Briefing in November 1996

Summary of Staff's Review of NEI's GFD

- Objective is to Establish a Generic Methodology for Applying Revised Source Term at Operating Plants

- Plant Changes Classified in Four Groups

- Generic Framework Based on Four Principles

Summary of Staff's Review of NEI's GFD
(continued)

- **Four Groups of Plant Changes Desired**

    - **Allowable Leak Rate Changes**

    and

    - **Isolation Valve Timing Changes**

    DFR's comment on ACRS '64 remarks
    ALARA considerations? - CWT

    - **Filtration Unit Simplification**

    - **Mitigation System Actuation Timing**

Summary of Staff's Review of GFD
(continued)

- **Four Principles of GFD**

Principle 1: Existing Licensing Basis is Acceptable

- backfitting not required

Principle 2: Complete Implementation of New Source Term as Substitute for Existing Licensing Basis is Acceptable

- integrated assessment needed
- removal of accident mitigation hardware

Various:
EQ
Post-accident monit (see fig)
DCOLA
challenges to the containment

Summary of Staff's Review of GFD
(continued)

Principle 3: **Selective Implementation of Revised Source Term is Acceptable**

- timing-only applications
- if dose calculations needed, integrated assessment required

Principle 4: **Dose Calculations Based on Existing Methods and Limits is Acceptable**

- new analytical framework including TEDE and "any" two hour evaluation period

Staff Plans for Implementation

- Transmit Decision Letter on GFD to NEI

- Continue Review of Pilot Plants
  - 4+ applications
  - exemptions needed

Staff Plans for Implementation
(continued)

- **Begin Rulemaking**
  - existing Part 100 requires analyses to begin immediately following release of radioactivity
  - use of TEDE for additional radionuclides
  - provide consistency with revised Part 100
  - RES lead

Staff Plans for Implementation
(continued)

- **Begin Integrated Assessment**
  - change in source terms needs thorough evaluation in plant performance, not just dose calculations
  - multidisciplinary review (NRR lead)
  - rebaseline PWR and BWR
  - assist staff with implementation issues
  - develop technical bases for rulemaking