U.S. NUCLEAR REGULATORY COMMISSION

July 1975

REGULATORY GUIDE

OFFICE OF STANDARDS DEVELOPMENT

REGULATORY GUIDE 1.99

EFFECTS OF RESIDUAL ELEMENTS ON PREDICTED RADIATION DAMAGE TO REACTOR VESSEL MATERIALS

A. INTRODUCTION

General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires, in part, that the reactor coolant pressure boundary be designed with sufficient margin to ensure that, when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," which were added to 10 CFR Part 50 effective August 16, 1973, to implement, in part, Criterion 31, necessitate the prediction of the amount of radiation damage to the reactor vessel of water-cooled power reactors throughout its service life.

This guide describes general procedures acceptable to the NRC staff as an interim basis* for predicting the effects of the residual elements copper and phosphorus on neutron radiation damage to the low-alloy steels currently used for reactor vessels.

B. DISCUSSION

The principal examples of NRC requirements that necessitate prediction of radiation damage are:

1. Paragraph II.H of Appendix G defines the beltline in terms of a predicted adjustment of reference temperature at end of service life in excess of 50°F; paragraphs III.C and IV.B specify the additional test requirements for beltline materials that supplement the requirements for reactor vessel materials generally.

2. Paragraph II.C.3 of Appendix H establishes the required number of surveillance capsules on the basis of the predicted adjusted reference temperature at the end of service life. In addition, withdrawal of the first capsule (when four or more are required) is to occur when the predicted adjustment of reference temperature is approximately 50°F or at one-fourth of the service life, whichever is earlier.

3. Paragraph IV.C of Appendix G requires that vessels be designed to permit a thermal annealing treatment if the predicted value of adjusted reference temperature exceeds 200°F during their service life.

4. Paragraph II.B of Appendix H incorporates ASTM E185-73 by reference. Paragraph 4.1 of ASTM E185-73 requires that the materials to be placed in surveillance be those that may limit operation of the reactor during its lifetime, i.e., those expected to have the highest adjusted reference temperature or the lowest Charpy upper-shelf energy at end of life. Both measures of radiation damage must be considered.

5. Paragraph V.B. of Appendix G describes the basis for setting the upper limit for pressure as a function of temperature during heatup and cooldown for a given service period in terms of the predicted value of the

*Research and construction experience with low-residual-element compositions of these steels is accumulating rapidly and is expected to provide a firm basis for acceptable procedures in the near future.

adjusted reference temperature at the end of the service period.

The two measures of radiation damage used in this guide are obtained from the results of the Charpy V-notch impact test. Appendix G to 10 CFR Part 50 requires that a full curve of absorbed energy versus temperature be obtained through the ductile-to-brittle transition temperature region. The latter is located by the reference temperature, RT_{NDT}, which is defined in paragraph II.F of Appendix G. The "shift" of the adjusted reference temperature is defined in Appendix G as the temperature shift in the Charpy V-notch curve for the irradiated material relative to that for the unirradiated material, measured at the 50-foot-pound energy level or measured at the 35-mil lateral expansion level, whichever temperature shift is greater. In using published data that report only the temperature shift measured at the 30-foot-pound energy level, it has been assumed herein that the adjustment of the reference temperature is equal to the 30-foot-pound shift.

The second measure of radiation damage is the decrease in the Charpy upper-shelf energy level. In the absence of a standard definition, the upper-shelf energy is defined herein as the average energy value for all specimens whose test temperature is above the upper end of the transition temperature region. Normally, at least three specimens should be included; more specimens should be included when the shelf level appears to be marginal.

The measure of fluence used herein is n/cm^2 ($E > 1$ MeV), consistent with the data base* for the expression relating fluence to neutron damage. This procedure is not intended to preclude future use of data that are given in terms of neutron damage fluence.

As used herein, references to "% Cu" and "% P" mean the weight percent of copper and phosphorus as measured in the surveillance program per ASTM E 185-73. However, if such results are not available, the results of a product analysis may be used.

Use of the procedures for prediction of radiation damage given in the regulatory position should be limited to irradiation at 550 ± 25°F, because temperature is important to damage recovery processes. As a guideline, irradiation at 450°F has been shown to cause twice the adjustment of reference temperature and irradiation at 650°F, about half the adjustment produced by irradiation at 550°F for the steels cited in the regulatory position when the copper content is about 0.15%. The effects of irradiation temperature on decrease in shelf energy should be considered quantitatively similar to those cited for the adjustment of reference temperature.

Sensitivity to neutron embrittlement may be affected by other residual elements such as vanadium and by deoxidation practice, as indicated by the findings of current research. In predicting radiation damage for materials that differ in residual element content or deoxidation practice from those that make up the data base, such findings should be considered. Other residual elements, notably sulfur, impair the initial Charpy shelf energy of these materials, and their content should be kept low. Clearly, it is the remaining toughness at end of life or at some other critical period that is important. Such toughness may be given in terms of the margin between the operating temperature (nominally 550°F) and the limiting temperature based on toughness. A margin of 200 degrees is desirable to permit safe management of system transients. At full power, the limiting temperature based on toughness is generally 150-200 degrees above RT_{NDT}; hence, the latter should not exceed 150-200°F at end of life. This limit also avoids the problems of providing for annealing, per paragraph IV.C of Appendix G. The levels of residual elements such as copper, phosphorus, sulfur, and vanadium that are required to achieve the limit of 200°F adjusted reference temperature at end of life in a given reactor vessel will depend on the initial values of RT_{NDT} of the beltline materials and on the predicted fluence at the particular locations in the vessel where the materials are used.

*The data base for this guide is that given by Spencer H. Bush, "Structural Materials for Nuclear Power Plants," 1974 ASTM Gillett Memorial Lecture, published in ASTM Journal of Testing and Evaluation, Nov. 1974 and its addendum, "Radiation Damage in Pressure Vessel Steels for Commercial Light-Water Reactors."

C. REGULATORY POSITION

1. Prediction of neutron radiation damage to the beltline of reactor vessels of light water reactors should be based on the following procedures. When surveillance data from the reactor in question become available, both sets of information should be considered in making the prediction.

Reference temperature should be adjusted as a function of fluence and residual element content in accordance with the following expression, within the limits listed below:

$$A = [40 + 1000(\%\,Cu - 0.08) + 5000(\%\,P - 0.008)]\,[f/10^{19}]^{1/2}$$

where

A = predicted adjustment of reference temperature, °F

f = fluence, n/cm^2 ($E > 1$ MeV)

% Cu = weight percent of copper. If % Cu < 0.08, use 0.08

%P = weight percent of phosphorus.
If $\%P \leq 0.008$, use 0.008.

If the value of A obtained by the above expression exceeds that given by the curve labeled "Upper Limit" in Figure 1, the "Upper Limit" curve should be used. If % Cu is unknown, the "Upper Limit" curve should be used.

As illustrated in Figure 1 for selected copper and phosphorus contents, the above expression should be considered valid only for $A > 50°F$ and for $f < 6 \times 10^{19}$ n/cm^2 ($E > 1$ MeV).

Charpy upper-shelf energy should be assumed to decrease as a function of fluence and copper content as indicated in Figure 2, within the limits listed below. Interpolation is permitted.

Application of the foregoing procedures should be subject to the following limitations:

a. The procedures apply to those grades of SA-302, 336, 533, and 508 steels having minimum specified yield strengths of 50,000 psi and under and to their welds and heat-affected zones.

b. The procedures are valid for a nominal irradiation temperature of 550°F. Irradiation below 525°F should be considered to produce greater damage, and irradiation above 575°F may be considered to produce less damage. The correction factor used should be justified.

c. The expression for A is given in terms of fluence as measured by units of n/cm^2 ($E > 1$ MeV); however, the expression may be used in terms of fluence as measured by units of neutron damage fluence, provided the constant 10^{19} n/cm^2 ($E > 1$ MeV) is changed to the corresponding value of neutron damage fluence.

d. Application of these procedures to materials having residual element content beyond that represented by the current data base should be justified by submittal of data.

2. For new plants, the reactor vessel beltline materials should have the content of residual elements such as copper, phosphorus, sulfur, and vanadium controlled to low levels. The levels should be such that the predicted adjusted reference temperature at end of life is less than 200°F.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for utilizing this regulatory guide.

This guide reflects current regulatory practice. Therefore, except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the positions described in this guide will be used by the NRC staff as follows:

1. The method described in regulatory position C.1 of this guide will be used in evaluating all predictions of radiation damage called for in Appendices G and H to 10 CFR Part 50 submitted on or after January 15, 1976; however, if an applicant wishes to use the recommendations of regulatory position C.1 in developing submittals before January 15, 1976, the pertinent portions of the submittal will be evaluated on the basis of this guide.

2. The recommendations of regulatory position C.2 will be used in evaluating construction permit applications docketed on or after January 15, 1976; however, if an applicant whose application for construction permit is docketed before January 15, 1976, wishes to use the recommendations of regulatory position C.2 of this regulatory guide in developing submittals for the application, the pertinent portions of the application will be evaluated on the basis of this guide.

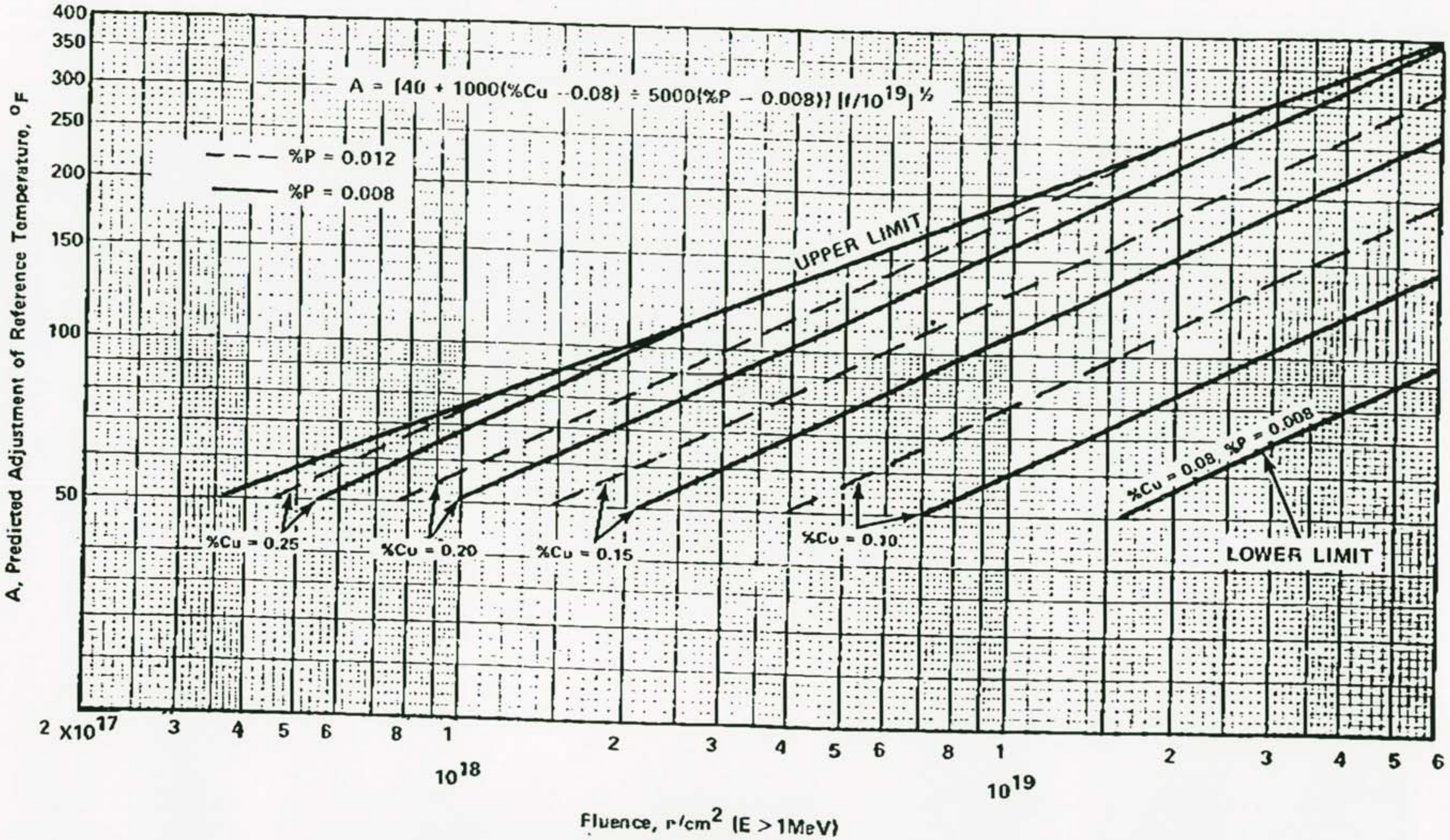

Figure 1 Predicted Adjustment of Reference Temperature as a Function of Copper and Phosphorus Content and Fluence.

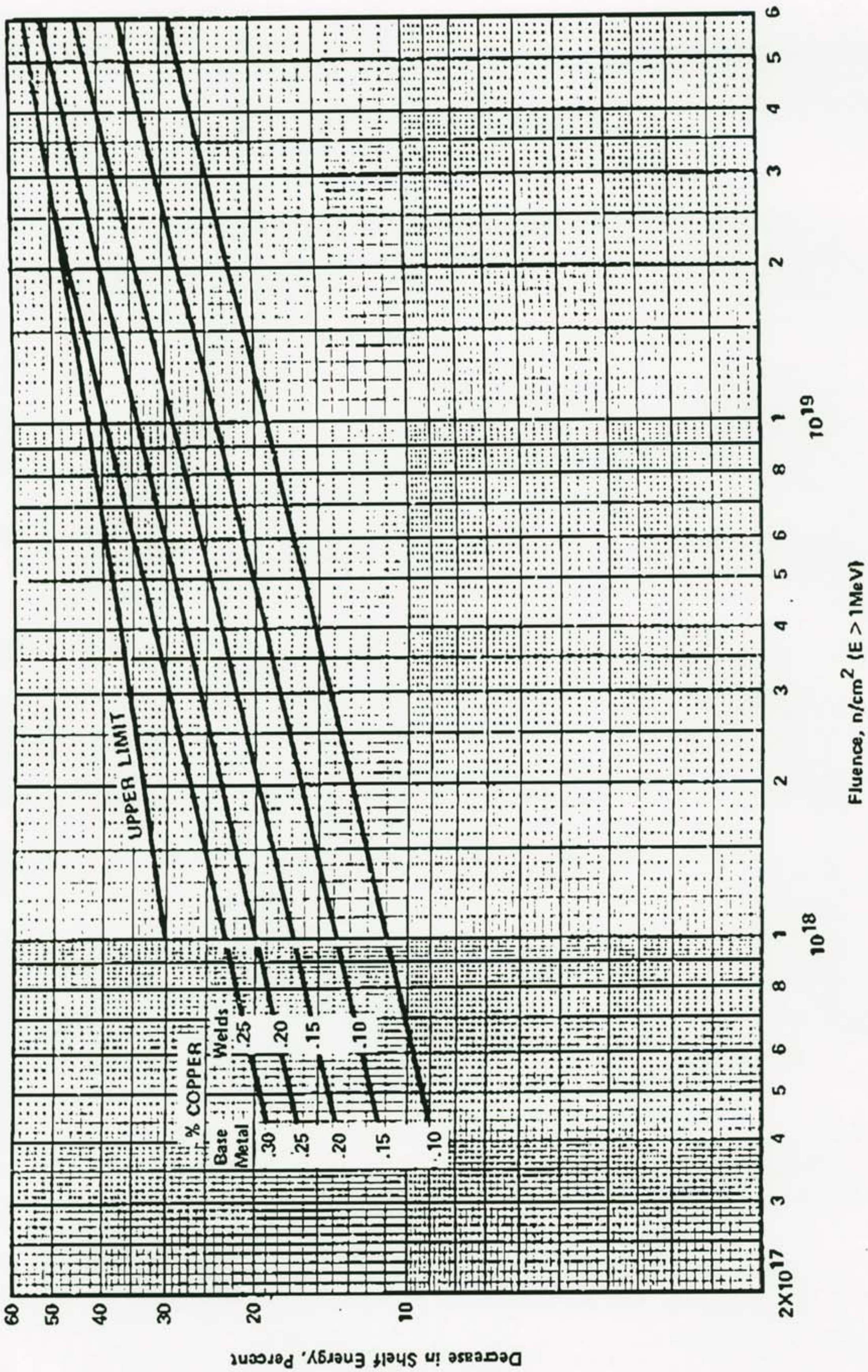

Figure 2 Predicted Decrease in Shelf Energy as a Function of Copper Content and Fluence.

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

February 1, 1977

MEMORANDUM FOR: Lee V. Gossick, Executive Director for Operations

FROM: E. G. Case, Chairman, Regulatory Requirements Review Committee

SUBJECT: SUMMARY OF REGULATORY REQUIREMENTS REVIEW COMMITTEE MEETING NO. 57, JANUARY 14, 1977

The Committee reviewed:

1. PROPOSED REGULATORY GUIDE 1.99, REVISION 1, "EFFECTS OF RESIDUAL ELEMENTS ON PREDICTED RADIATION DAMAGE TO REACTOR VESSEL MATERIALS," and recommended approval subject to the following comments:[1]

   a. The footnote on research and construction experience on page 1 of the proposed revision should be expanded to include the wording concerning the limitations of the guide suggested by the Office of Nuclear Regulatory Research.

   b. The Committee characterized paragraphs C.1, C.2, and C.4 of the proposed Regulatory Position as Category 3 - backfit required; and paragraph C.3 as Category 1 - no backfit.

2. PROPOSED REGULATORY GUIDE 1.XX, "ACCEPTABLE MODEL AND RELATED STATISTICAL METHODS FOR ANALYSIS OF FUEL DENSIFICATION," and found that the proposed Regulatory Guide represented an acceptable model and appropriate methods for analysis of fuel densification effects. Utilization of such models and methods in the licensing process should be implemented in accordance with 10 CFR 50.46 and the related Appendix K to 10 CFR 50.46.

3. DRAFT REGULATORY GUIDE 1.XXX, "INSPECTION OF WATER CONTROL STRUCTURES ASSOCIATED WITH NUCLEAR PLANT FACILITIES" (DRAFT "A", DATED SEPTEMBER 17, 1976) and recommended approval subject to the following comments:[2]

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[1]Committee comments refer to the draft distributed at the meeting dated January 1977, rather than the draft distributed with the agenda dated September 1976.

[2]The Committee comments refer to the draft guide distributed at the meeting dated December 16, 1976, rather than the draft distributed with the agenda dated September 17, 1976.

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a. The Committee review of this proposed guide was limited to its applicability and acceptability with regard to water control structures which could affect radiological safety. The proposed guide should be amended throughout (including the title) to set forth this limitation.

b. The Committee leaves to the staff the development of appropriate methods and procedures to be utilized for assuring that the inspection, surveillance, and/or test requirements set forth in the guide are met by licensees.

The Committee characterized this guide as Category 3 - backfit required.

4. PROPOSED REVISION 2 TO REGULATORY GUIDE 1.32, "CRITERIA FOR SAFETY RELATED ELECTRIC POWER SYSTEMS FOR NUCLEAR POWER PLANTS," and recommended approval.

The Committee characterized this guide as Category 1 - no backfit.

5. PROPOSED REGULATORY GUIDE 10.XX, "GUIDANCE ON IMPLEMENTATION OF REGULATORY GUIDES TO STANDARDIZED NUCLEAR POWER PLANTS," and concluded the following:

a. Guidance on the implementation of regulatory guides for standardized nuclear power plants should be included in a revision to WASH-1341, rather than in a regulatory guide.

b. The present system for Committee characterization of the backfit potential of proposed guides for operating plants can and should be applied to standardized plants; i.e., Category 1 - need not be applied to standardized plants; Category 2 - applicability to standardized plants must be decided by NRR on a case-by-case basis; Category 3 - must be applied to standardized plants. The Executive Secretary should develop and circulate for Committee comments within 2 weeks a proposed revision to the definitions of backfit categories previously established by the Committee in Meeting No. 31 (see memorandum to L. V. Gossick dated September 24, 1975) that includes these additions concerning standardized plants, and clarifies, as necessary, the applicability of Committee-considered guides to custom plants with CPs.

c. NRR (DPM) should develop a program and procedures for implementing the Committee's recommendations concerning backfit potential of changes in regulatory requirements to those specific plants not having an operating license which are affected.

E. G. Case, Chairman
Regulatory Requirements Review Committee