Westinghouse Class 3

WCAP-10126-NP-A

EXTENDED BURNUP EVALUATION OF WESTINGHOUSE FUEL

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Edited by:	Edited by:
P. J. Kersting	S. L. Davidson W. R. Kramer

Contributors

M. R. Adler
S. I. Dederer
E. E. DeMario
S. Fyfitch
J. L. Grover
S. K. Kapil
P. J. Kersting
G. E. Paul
J. Skaritka
H. W. Wilson

Approved: RSKaiser
R. S. Kaiser
Performance Analysis

8601230266 860107
PDR TOPRP EMVWEST
B PDR

WESTINGHOUSE ELECTRIC CORPORATION
Nuclear Energy Systems
P. O. Box 355
Pittsburgh, Pennsylvania 15230

4222L:6-851216

WCAP-10126-NP-A

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

SECTION A

NRC ACCEPTANCE LETTER
DATED OCTOBER 11, 1985

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

October 11, 1985

Mr. E. P. Rahe, Jr., Manager
Nuclear Safety Department
Westinghouse Electric Corporation
Box 355
Pittsburgh, Pennsylvania 15230

Dear Mr. Rahe:

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT WCAP-10125(P), "EXTENDED BURNUP EVALUATION OF WESTINGHOUSE FUEL"

We have completed our review of the subject topical report submitted by the Westinghouse Electric Corporation (Westinghouse) by letter dated July 28, 1982. We find the report to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our 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 is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that Westinghouse publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, Westinghouse and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

Cecil O. Thomas

Cecil O. Thomas, Chief
Standardization and Special
Projects Branch
Division of Licensing

Enclosure:
As stated

SECTION B

NRC SAFETY EVALUATION REPORT (SER)
DATED MAY 1985

SAFETY EVALUATION REPORT ON WESTINGHOUSE ELECTRIC CORPORATION

EXTENDED BURNUP TOPICAL REPORT - WCAP-10125 (Proprietary)

May 1985

Prepared by
Core Performance Branch and Accident Evaluation Branch
Division of Systems Integration
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

Table of Contents

1.0 INTRODUCTION

Economics and prudent utilization of resources has led utilities to seek more efficient use of current generation light water reactors (LWRs). Improved fuel utilization is one of the avenues being pursued for greater efficiency. One of the greater improvements in fuel utilization is to increase the fuel discharge exposure which is currently at batch average burnups of approximately 28 MWd/kgM for BWRs and approximately 33 MWd/kgM for PWRs to batch average burnups of approximately 40 MWd/kgM and 50 MWd/kgM or above, respectively.

In response to this trend for extended burnup fuel operation, the Nuclear Regulatory Commission (NRC) has requested each fuel vendor to prepare and submit a topical report for review and approval that covers extended burnup experience, methods and test data to provide a generic basis for operation at extended burnups (Reference 1).

Westinghouse Electric Corporation Westinghouse has submitted such a report (Reference 2) requesting generic licensing approval of their criteria and methods used for licensing their fuel designs, for application at extended burnups. In addition, Westinghouse has also provided responses (References 3, 4, 5, 6) to NRC questions concerning this submittal.

This review considered only Zircaloy-clad Westinghouse fuel and did not consider the effects of extended burnup on Westinghouse stainless steel clad fuel.

This technical review and evaluation has been performed by Pacific Northwest Laboratory (PNL) under contract (FIN B2533) with the United States NRC. The review has been based on References 2 through 6 and Section 4.2 of the Standard Review Plan (SRP) (Reference 7) and covers the fuel assembly, fuel rods, and burnable poison rods but does not include the rod cluster control assemblies for extended burnup operation.

This report follows the intent of Section 4.2 of the SRP, where appropriate for a generic review, to insure that all licensing requirements of the fuel

system are reviewed with respect to extended burnup operation. The objective of Section 4.2 and this review is to provide assurance that, as a result of extended burnup operation, (a) the system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained. "Not damaged" is defined as meaning that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. This objective implements General Design Criterion (GDC) 10 of 10 CFR Part 50, Appendix A ("General Design Criteria for Nuclear Power Plants") and the design limits that accomplish this are called Specified Acceptable Fuel Design Limits (SAFDLs). "Fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10 CFR Part 100 ("Reactor Site Criteria") for postulated accidents. "Coolability," which is sometimes termed "coolable geometry," means, in general, that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channels to permit removal of residual heat after a severe accident. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the General Design Criteria (GDC e.g., 27 and 35). Specific coolability requirements for the loss-of-coolant accidents are given in 10 CFR Part 50.46 ("Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors").

In order to meet the above stated objectives and follow the format of Section 4.2, this review covers the following three categories: (1) <u>Fuel System Damage Mechanisms</u>, which are most applicable to normal operation and anticipated operational occurrences, (2) <u>Fuel Rod Failure Mechanisms</u>, which apply to normal operation, anticipated operational occurrences and postulated accidents and; (3) <u>Fuel Coolability</u>, which apply to postulated accidents.

Because the purposes of each vendor's high burnup topical report are slightly different, it is useful to quote Westinghouse's goal in preparing this report, as stated in Reference 2.

"The purposes of this topical is to justify operation of Westinghouse designed fuel to [the target] lead fuel rod average burnup..."

In addition, Westinghouse stated that "The information supplied in this report supports the conlcusion that Westinghouse design methods and safety analyses are valid for operation to "the [proprietary] lead rod average burnup target. No performance limitations have been identified which would preclude the design of Westinghouse fuel to this target burnup, and it has been shown that current design and safety evaluation criteria can be applied with no modification to these criteria".

The criteria sections in this review address limiting values for fuel damage that are acceptable under the three major categories of failure mechanisms listed above and in the SRP. The purpose of this review is to determine if the Westinghouse criteria are applicable to extended burnup operation of their fuel. These criteria along with certain definitions for fuel failure constitute the SAFDLs required by GDC 10.

The evaluation sections review the methods that Westinghouse uses to demonstrate that the design criteria have been met for extended burnup operation and thus are reviewed with respect to their applicability to the proposed range of extended burnup operation. These methods and data may include operating experience, prototype testing and analytical techniques. The determination that specific Westinghouse designs meet the stated criteria is not addressed in this review but will be addressed in specific licensing applications.

Westinghouse uses the ANS classification of plant conditions which divides plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

1. Condition I: Normal Operation and Operational Transients
2. Condition II: Faults of Moderate Frequency
3. Condition III: Infrequent Faults
4. Condition IV: Limiting Faults

This approach is used consistently in Westinghouse analyses and will be used in this safety evaluation report.

2.0 FUEL SYSTEM DAMAGE

The design criteria in this section should not be exceeded during normal operation including anticipated operational occurrences (AOOs). The evaluation portion of each damage mechanism demonstrates that the design criteria are not exceeded during normal operation and AOOs.

(a) Design Stress

Bases/Criteria - The Westinghouse design basis for fuel assembly, fuel rod, burnable poison rod, and upper end fitting spring stresses is that the fuel system will be functional and will not be damaged due to excessive stresses. The design limit for fuel rod cladding stress under Condition I and II modes of operation is that the volume averaged effective stress calculated with the von Mises equation, considering interference due to uniform cylindrical pellet-to-cladding contact (caused by pellet thermal expansion and swelling, uniform cladding creep, and fuel rod/coolant system pressure differences), is less than the Zircaloy 0.2 percent offset yield stress with consideration of temperature and irradiation effects as described in Reference 8. This report has been approved by the NRC (Reference 9). This criterion is applicable to extended burnup operation.

Evaluation - The Performance-Analysis-and-Design (PAD) code Version 3.3 (Reference 10) is used by Westinghouse to assure that the above criterion is met. This code has been verified against fuel rod data with rod average burnups up to approximately 57 MWd/kgM. This code takes into account those parameters important for determining cladding stresses at extended burnups, such as pellet thermal expansion and swelling, cladding creep and fuel rod/coolant system pressure differences. The NRC has approved (Reference 11) the use of this code for licensing applications without burnup restrictions. Consequently, this code is found acceptable for determining cladding stress on fuel rods with extended burnups up to those requested by Westinghouse in Reference 2.

It is noted that cladding stresses and strains due to transients (Condition II events) at extended burnups are not expected to be limiting because the power capability is reduced due to fissile material burnout, limiting the power excursion and thus stresses experienced by these rods.

(b) Cladding Design Strain

Bases/Criteria - The Westinghouse design basis for fuel rod cladding strain is that the fuel system will not be damaged due to excessive cladding strain. In order to meet this design basis the Westinghouse design limit for cladding strain during steady-state operation is that the total plastic tensile creep and uniform cylindrical fuel pellet expansion due to fuel swelling and thermal expansion is less than 1 percent from the unirradiated condition. For Condition II transients, the design limit for cladding strain is that the total tensile strain due to uniform cylindrical pellet thermal expansion during the transient is less than 1 percent of the pretransient value. These design strain bases and limits have been presented previously by Westinghouse (Reference 12) approved by the NRC (Reference 13) for application to current burnup fuel.

The material property that could have a significant impact on the cladding strain limit at extended burnup levels is cladding ductility. The strain criterion could be impacted if cladding ductility were decreased, as a result of extended burnup operation, to a level that would allow cladding failure without the Condition I and II cladding strain criterion being exceeded in the Westinghouse analyses. From examination of irradiated Zircaloy cladding ductility data (References 14, 15), it has been concluded that ductility decreases with increasing fluence at low burnup levels, i.e., less than 12 MWd/kgM, but asymptotically approaches either a constant value or a small fluence dependence beyond these low burnups. Consequently, cladding ductility has either little or no change for the increased burnup levels projected for Westinghouse extended burnup operation. In addition, Westinghouse has irradiated experimental and lead test rods with average burnups up to approximately 62 MWd/kgM with no adverse effects in cladding ductility.

From the above, we can conclude that the strain limit proposed by Westinghouse is applicable for extended burnup application.

Evaluation - The NRC-approved Westinghouse fuel performance code, PAD 3.3 (Reference 10), is used to assure that Westinghouse fuel meets the above criterion. As noted in the Design Stress section, this code has been verified against fuel rod data with rod average burnups up to approximately 57 MWd/kgM and takes into account those parameters important for determining cladding stresses and strains at extended burnups. Consequently, this code is found acceptable for determining cladding strains on fuel rods with extended burnups up to those requested by Westinghouse (Reference 2).

(c) Strain Fatigue

Bases/Criteria - The Westinghouse design basis for fuel rod cladding fatigue is that the fuel system will not be damaged due to cladding strain fatigue. In order to assure that this design basis is met, Westinghouse imposes a design limit for strain fatigue such that the fatigue life usage factor is less than 1.0. That is, for a given strain range, the number of strain fatigue cycles are less than those required for failure when a minimum safety factor of 2 on the stress amplitude or a minimum safety factor of 20 on the number of cycles, whichever is the more conservative, is imposed. This criterion is the same as that given in Section 4.2 of the Standard Review Plan.

As noted in the Cladding Design Strain section, the material property that could have a significant effect on cladding strain and thus strain fatigue at extended burnups is cladding ductility. However, as discussed above, extended burnup operation has shown little or no observable effects on cladding ductility and performance. From this, it is concluded that extended burnup operation does not reduce the applicability of the fatigue limits and thus the Westinghouse criterion is found acceptable for use in extended burnup applications.

Evaluation - The NRC-approved Westinghouse fuel performance code, PAD 3.3, is used to determine the strain range for the fatigue usage analysis. The Langer O'Donnell fatigue model (Reference 16) with the empirical factors in this model modified in order to conservatively bound the Westinghouse test data, is used with the strains from PAD 3.3 to assure that the above criterion is met. A description of this methodology and the Westinghouse data base is presented in WCAP-9500 (Reference 12) which has been approved by the NRC (Reference 13).

This methodology takes into account daily load follow operation and the additional fatigue load cycles that may result from extended burnup operation. In addition, the PAD 3.3 code accounts for those parameters important for determing cladding strains at extended burnups, see Sections 2.0(a) and 2.0(b). Therefore, the above methodology is found to model operational and material behavior parameters important for determining strain fatigue at extended burnups and thus is acceptable for extended burnup application.

(d) Fretting Wear

Bases/Criteria - Fretting wear is a concern for fuel and burnable poison rods, and the Zircaloy guide tubes. Fretting, or wear, may occur on the fuel and/or burnable rod cladding surfaces in contact with the spacer grids if there is a reduction in grid spacing loads in combination with small amplitude, flow-induced, vibratory forces. Guide tube wear may result when there is flow induced motion between the control rod ends and the inner wall of the guide tube.

While the Standard Review Plan (SRP), Section 4.2, (Reference 7) does not provide numerical bounding-value acceptance criteria for fretting wear, it does stipulate that the allowable fretting wear should be stated in the safety analysis report and that the stress/strain and fatigue limits should presume the existence of this wear.

The Westinghouse design basis for fuel rod fretting wear is that fuel rods shall be designed not to fail due to fretting wear during Condition I and II events. In order to meet this basis, Westinghouse uses a general guide for wall thickness reduction which is a percent of the original wall thickness (the specific value is proprietary) for evaluating cladding imperfections, including wear marks. Westinghouse indicates that the cladding stress and fatigue limits, discussed here in Sections 2.0(a) and 2.0(c), apply to fretting wear. Westinghouse also indicates (Reference 2) that fretting wear will not have a significant effect on cladding stresses and thus need not be considered in stress related analyses. We have confirmed that fretting wear effects on the stress analysis are insignificant as long as the fretting wear in Westinghouse designed rods remains below the general guide for cladding imperfections stated in terms of percent wall thickness in the extended burnup topical report (Reference 2). These design

bases and criteria are found to be acceptable for extended burnup application. The Westinghouse design bases and criteria for guide thimble tube wear is that no perforation of the tube wall should occur and that the integrity of the guide thimble tube be maintained throughout the normal life of a fuel assembly. As an additional design limit on guide thimble tubes, Westinghouse has determined (Reference 12) from stress analyses that the limiting load on the fuel assembly structure is that which might occur during a fuel handling accident. A design criterion of 6 g is used for the analysis of this accident and this has previously been approved by the NRC (Reference 13). These design bases and criteria are also found to be acceptable for extended burnup application.

Evaluation - Westinghouse utilizes empirical data taken from operating reactors and out-of-reactor wear tests to provide assurance that the above criteria are met for both Zircaloy and Inconel grid designs. Fuel rod fretting is affected by the increased fluence and in-reactor residence time associated with extended burnup. The increased fluence results in a slight decrease in grid spring forces and the increased residence time may result in a small increase in wear volume.

Fretting type failures have been observed at the bottom (Inconel) grid location of several rods in one of the Westinghouse 14x14 OFA assemblies. Westinghouse has stated (Reference 3) that the cause of these failures was traced to non-standard installation of the rods in the assembly during fabrication, rather than to a generic problem in rod or grid design. To support this, Westinghouse has shown (Reference 3) that the remaining 14x14 and 17x17 OFA assemblies with assembly average burnups up to 39 MWd/kgM have shown no indication of fretting wear indicating that Zircaloy grid spring forces continue to preclude fretting. Out-of-reactor tests on OFA assemblies have indicated (Reference 2) that fuel rod fretting wear will not be a limiting concern up to the extended burnup level requested. From this it is concluded that fuel rod fretting is not expected to be a problem for OFA fuel designs with Zircaloy grids; however, in order to confirm this conclusion, it is recommended that additional fuel rod fretting data be obtained on Zircaloy grid assemblies up to the extended burnup level requested by Westinghouse.

Westinghouse 15x15 and 17x17 assemblies using the Inconel grid design have been irradiated for five cycles of operation (assembly average and lead rod average burnups of approximately 55 MWd/kgM and 60 MWd/kgM, respectively) and for four

cycles of operation, respectively. Detailed visual examinations of these assemblies have indicated no evidence of cladding fretting. Consequently, Westinghouse fuel designs with Inconel grids are found to be acceptable for extended burnup operation.

(e) Oxidation and Crud Buildup

Bases/Criteria - The Westinghouse design basis for cladding oxidation is that the fuel system will not be damaged due to excessive cladding oxidation. In order to preclude a condition of accelerated oxidation, Westinghouse imposes specific temperature limits on the cladding. The temperature limits applied to cladding oxidation are that calculated cladding temperatures (at the oxide-to-metal interface) shall be less than a specific (proprietary) value during steady-state operation, and for Condition II transients the metal-to-oxide interface shall not exceed a higher (proprietary) value. These criteria have been approved by NRC (Reference 13) for current burnup levels and are also found to be applicable to extended burnup operation.

Evaluation - The SRP states that the effects of cladding crud and oxidation need to be addressed in safety analyses, such as thermal and mechanical analyses. The major means of controlling cladding crud and oxidation is through primary coolant chemistry controls; however, this does not eliminate the need to include their effects in safety analyses at extended burnups.

Westinghouse has presented (Reference 3) two sets of oxide thickness data: 1) those induced by thick crud deposits, along with a bounding curve for the data, and 2) nominal oxide and crud thickness data along with a bounding curve (labeled for this discussion as best estimate) for these data. Westinghouse has indicated that they have primary water chemistry controls that limit the amount of crud deposits and only those plants that have operated outside of these chemistry controls have been observed to have the thick crud deposits and abnormally high oxide thicknesses. This is consistent with past industry experience.

Westinghouse has indicated that a best estimate (proprietary) value of crud is input to PAD 3.3 and PAD 3.3, Addenda 2 and the best estimate bounding curve (from Figure 1 of Reference 3) for cladding oxide thickness is modeled in the PAD 3.3 and PAD 3.3 Addenda 2 codes. This is found to be acceptable for thermal

evaluations of extended burnup fuel because Westinghouse imposes water chemistry controls on their plants to maintain crud and oxide thicknesses to nominal values up to the extended burnup range requested.

For mechanical analyses Westinghouse has indicated that they reduce their cladding wall thickness by a specified (proprietary) amount to account for cladding defects and cladding oxidation. This amount is found to more than bound the cladding thickness reduction due to cladding oxidation at the extended burnups requested. Therefore, this methodology is acceptable for extended burnup application.

(f) Rod Bowing

Bases/Criteria - Fuel and burnable poison rod bowing are phenomena that alter the design-pitch dimensions between adjacent rods. Bowing affects local nuclear power peaking and the local heat ransfer to the coolant. Rather than placing design limits on the amount of bowing that is permiteed, the effects of bowing are included in the safety analysis. This is consistent with the Standard Review Plan and the NRC has approved (Reference 17) this for current burnups. It remains acceptable for extended burnups. The methods used for predicting the degree of rod bowing at extended burnups are evaluated below.

Evaluation - The Westinghouse methods for evaluating fuel and burnable poison rod bowing in 14x14, 15x15 and 17x17 assembly designs has been addressed in Reference 18 which has been approved by the NRC (Reference 17) for current burnup levels. In response to NRC questions in this review Westinghouse has shown (Reference 3) that their conservative upper 95th percentile worst span closure curve from Reference 18 conservatively bounds their rod bow data with regional average burnups up to 48 MWd/kgM, i.e., peak rod average burnups approximately 53 MWd/kgM. In addition, Westinghouse has indicated (Reference 2) that Westinghouse fuel assemblies will not be capable of achieving limiting power peaking factors at extended burnups due to the reduced power capabilities of these assemblies. For example, the extended burnup assemblies are not limited by rod bow imposed penalties above assembly average burnups of approximately 33 MWd/kgM because the self-imposed decrease in power capabilities is greater than the penalty. Therefore, the operation of Westinghouse fuel assemblies is found acceptable for this burnup range with respect to rod bowing.

(g) Axial Growth

Bases/Criteria - The core components requiring axial dimensional analyses are the neutron soruce rods, burnable poison rods, fuel rods, and fuel assemblies (thimble plugging rods are omitted because they are short and not axial growth limited). The axial growth of the first two of these components is primarily dependent upon the behavior of poison, source, or spacer pellets and their Type 304 stainless-steel cladding. The growth of the last two is mainly governed by fuel-pellet contact, and creep and irradiation growth of the Zircaloy-4 cladding, and Zircaloy-4 guide thimble tubes. Failure to adequately design for axial growth of these components can lead to fuel rod-to-nozzle gap closure, rod bowing and perhaps fuel rod failure. In addition, growth of the guide thimble tubes can result in collapse of the assembly holddown springs.

The Westinghouse design bases for core component rods are that (a) dimensional stability and cladding integrity are maintained during Condition I and II events and (b) these components do not interfere with shutdown during Condition III and IV events.

Westinghouse does not, per se, have design limits on the axial growth of their control, source, and burnable poison rods. However, allowances are made to accommodate (a) pellet swelling due to gas production and (b) relative thermal expansion between the stainless-steel cladding and the encapsulated material. Westinghouse does not account for irradiation growth of the stainless-steel cladding and has cited experiments (Reference 19) as justification for the insignificance of irradiation growth of stainless-steel at PWR operating conditions. This is also found to be true for extended burnup operation with the Zircaloy clad fuel rods providing the limiting conditions for irradiation growth.

For the Zircaloy cladding and fuel assembly components, the axial-dimensional tolerances that require controlling are (a) the spacing between the top and bottom of the fuel rods and the top and bottom fuel assembly nozzles, respectively, and (b) the spacing between the fuel assemblies and the upper and lower core plates. As noted earlier, failure to adequately design for the former may result in fuel rod bowing, and for the latter may result in collapse

of the assembly holddown springs. With regard to a design basis for both rod-to-nozzle gap spacings and fuel assembly to core spacings, Westinghouse withdrew the proposed design limit in their extended burnup report (Reference 2) and indicated that they will continue to use the design limit approved in WCAP-9500 (Reference 12) which states that no axial interference shall take place due to closure of either the rod-to-nozzle gap spacing or the fuel assembly to core spacing.

Evaluation - From the Westinghouse topical report on extended burnup (Reference 2) and responses (Reference 3) to NRC questions, Westinghouse has shown that they have both rod and assembly growth data near the burnups and fluences requested for extended burnup operation. These data indicate that the rod-to-nozzle gap spacings on Westinghouse fuel designs are approaching the above criteria at extended burnups and thus should be monitored in their fuel surveillance program. The models used by Westinghouse to predict rod and assembly growth appear to bound the extended burnup data and thus are found to be accpetable for extended burnup applications.

(h) Rod Internal Pressure

Design Bases/Criteria - The Westinghouse design basis for fuel rod internal pressure is that the fuel system will not be damaged due to excessive fuel rod internal pressure. The Westinghouse design limits used to meet this design basis are that the internal pressure of the lead rod in the reactor will be limited to a value below that which could result in (1) the diametral gap increasing due to outward cladding creep during steady-state operation and (2) extensive DNB propagation (References 2 and 20) This design basis and the associated limits have been found acceptable by the NRC (Reference 21) for current burnup levels, and they are not found to be affected by extended burnup operation. Therefore, they are also found to be acceptable for extended burnup application.

Evaluation - The models and methods used by Westinghouse to evaluate whether their designs meet the above basis and limits are examined in this section. The models used by Westinghouse are contained in the PAD 3.3 code (Reference 10) which has been approved by the NRC (Refernce 11) without restrictions of its use to high burnup fuel. As noted in Section 2.0(a) this code has been verified against fuel rod data with rod average burnups up to approximately 57 MWd/kgM. The NRC review of this code paid particular attention to those parameters important to internal rod pressure predictions, i.e., the thermal and fission gas release models.

Therefore, the PAD 3.3 code is approved for use in the evaluation of rod internal pressures of extended burnup fuel to the target Westinghouse burnup.

An important parameter of the methodology used in the internal rod pressure evaluations is the power history used as input to the PAD code. Power history is very important because fission gas release and thus internal rod pressures are strongly dependent on the fuel thermal history. In response to an NRC question on the power histories used, Westinghouse has indicated (References 5 and 6) that the power histories input to the PAD 3.3 code for calculating internal rod pressures are based on best estimate peak power histories from their fuel management calculations. The peak rod power histories are chosen by Westinghouse from their fuel management calculations based on those rods which have experienced the highest rod powers during each reactor cycle of operation, e.g., cycle 1, 2, 3, etc., along with the power history of the peak burnup rod of the fuel batch for a total of (number of cycles +1) power histories. For example, a batch of fuel that will experience four cycles of operation will have at the maximum five peak power histories (sometimes a rod with a peak power during a specific reactor cycle corresponds to the peak burnup rod which would give four peak power histories for this case). Each of these five peak power histories are then input separately into PAD 3.3 to calculate five different end-of-life internal rod pressures with the highest pressure being subject to the above criteria.

In response to a subsequent NRC question concerning the conservatism in the rod power history methodology used by Westinghouse, Westinghouse presented analytical calculations (Reference 6) to demonstrate that the models used by Westinghouse are conservatively biased to bound any power uncertainties and Condition II power excursions that their fuel may experience as a result of extended burnup operation. As a check on the conservatism in the Westinghouse methodology, we have performed audit calculations using the GT2R2 code (Reference 22) along with bounding power histories of Westinghouse fuel to show that the Westinghouse methodology predicts bounding end-of-life rod pressures up to the extended burnups requested by Westinghouse (Reference 2). From the above evaluation, it is concluded that Westinghouse models and methodology for determining end-of-life rod pressures are adequately conservative and thus acceptable up to the extended burnups requested by Westinghouse (Reference 2).

(i) Assembly Liftoff

Design/Bases - The SRP calls for the fuel assembly holddown capability (wet weight and spring forces) to exceed worst-case hydraulic loads for normal operation, which includes anticipated operational occurrences. Westinghouse has stated (References 2 and 12) that they meet this criterion for all Condition I and II events with the exception of the turbine overspeed transient associated with a loss of external load. The NRC has accepted (Reference 13) this condition in the past for current burnup levels as long as the affected fuel assemblies can be shown to reseat properly in the core plate without damage or other adverse effects during the event. This also remains acceptable for extended burnup assemblies as long as the same criteria are met as for current burnup fuel.

Evaluation - The fuel assembly liftoff forces are a function of primary coolant flow, spring forces and assembly dimensional changes. Westinghouse has indicated (Reference 2) that extended burnups will result in 1) additional irradiation relaxation of the holddown springs and 2) assembly length increases. These two phenomena have opposing effects on assembly holddown forces; however, Westinghouse predicts that there is a net increase in force with increased irradiation because fuel assembly growth is the dominant effect which more than compensates for the decrease in spring force. This is consistent with industry experience and thus assembly liftoff is not judged to be a problem at extended burnups.

(j) Control Material Leaching

Control rods are not within the scope of this review since they are treated separately and may be removed or installed in a core independent of fuel assembly burnup.

3.0 FUEL ROD FAILURE

In the following paragraphs, fuel rod failure thresholds and analysis methods for the failure mechanisms listed in the Standard Review Plan are reviewed. When the failure thresholds are applied to normal operation including anticipated operational occurrences, they are used as limits (and hence SAFDLs) since fuel failure under those conditions should not occur according to the traditional conservative interpretation of General Design Criterion 10.

When these thresholds are used for postulated accidents, fuel failures are permitted, but they must be accounted for in the dose calculations required by 10 CFR 100. The basis or reason for establishing these failure thresholds is thus established by GDC 10 and Part 100 and only the threshold values and the analysis methods used to assure that they are met and reviewed below.

(a) Hydriding

Bases/Criteria - Internal hydriding as a cladding failure mechanism is precluded by controlling the level of hydrogen impurities in the fuel during fabrication. The moisture level in the uranium dioxide fuel is limited by Westinghouse (Reference 12) to less than or equal to 20 ppm, and this specification is compatible with the ASTM specification (Reference 23) which allows two micrograms of hydrogen per gram of uranium (i.e., 2 ppm). This is the same as the limit described in the Standard Review Plan and has been found acceptable by NRC (Reference 12) and continues to be acceptable for extended burnup application.

In addition, for extended burnup fuel, Westinghouse has introduced (Reference 2) a design limit on the hydrogen pickup level the value of which is proprietary. Westinghouse has indicated that their test results show that the mechanical properties of Zircaloy-4 are not adversely affected at this level of hydrogen. We agree with this assessment as long as hydride platelet orientation remains in the circumferential direction. Westinghouse has also stated that process controls and texture acceptance tests assure that Westinghouse cladding maintains the proper hydride platelet orientation. This design limit on hydrogen pickup level is found acceptable for extended burnup applications.

Evaluation - The hydrogen uptake of Zircaloy-4 during normal reactor operation to the extended burnup levels requested by Westinghouse is typically much lower than the Westinghouse criterion. The exception to this is when an abnormal amount of cladding oxidation is encountered that results in cladding failure. Cladding oxidation is addressed in Section 2.0(e). In this review Westinghouse has provided data (References 2 and 3) on hydrogen uptake from commercial reactor operation to burnups that bound the extended burnup level requested by Westinghouse. These data have shown that extended burnup operation up to the level

requested by Westinghouse remains significantly below their criterion for hydrogen uptake. From this it is concluded that hydriding is not a likely failure mechanism for Westinghouse fuel at extended burnups.

(b) Cladding Collapse

Bases/Criteria - If axial gaps in the fuel pellet column were to occur due to densification, the cladding would have the potential of collapsing into a gap (i.e., flattening). Because of the large local strains that would result from collapse, the cladding is assumed to fail. It is a Westinghouse design basis that fuel rod failures due to flattening will not occur. In order to meet this design basis Westinghouse imposes a design limit for fuel rod clad flattening such that "the core residence time will not exceed the calculated core residence time which corresponds to a flattened rod frequency of 1.0". This design basis and its associated criterion are essentially the same as those specified in the SRP and are found acceptable for extended burnup application.

Evaluation - The longer in-reactor residence times associated with extended-burnup fuel will increase the amount of creep of an unsupported fuel cladding. Extensive postirradiation examinations of both test and commercial fuel designs of current vintage by Westinghouse have not shown any evidence of cladding collapse or large local ovalities at rod average burnups up to approximately 62 MWd/kgM. This is primarily the result of the use of prepressurized rods and stable fuel in current generation designs.

Westinghouse utilizes a cladding collapse model (Reference 24) to show that the longer inreactor residence time associated with extended burnup fuel will not result in the collapse of an unsupported cladding with their fuel design. This method is very conservative in relation to the stable fuel employed in current designs, because it assumes a gap has formed in the fuel column and the tube is unsupported. This method has been approved by the NRC (Reference 25) and because it explicitly accounts for the longer in-reactor residence times of extended burnup fuel, it is also found acceptable for extended burnup applications.

(c) Overheating of Cladding

Bases/Criteria - The Westinghouse design limit for the prevention of fuel failures due to overheating is that there will be at least 95% probability at

a 95% confidence level that departure from nucleate boiling (DNB) will not occur on a fuel rod having the minimum DNBR during normal operation and anticipated operational occurrences (Condition I and II events). This design limit is consistent with the thermal margin criterion of SRP Section 4.2 and thus has been found acceptable by the NRC (Reference 13) for use at current burnup levels. It is also judged to remain acceptable for extended burnup applications.

Evaluation - As stated in SRP Section 4.2, adequate cooling is assumed to exist when the thermal margin criterion to limit the departure from nucleate boiling (DNB) in the core is satisfied.

(d) Overheating of Fuel Pellets

Bases/Criteria - As a second method of avoiding cladding failure due to overheating, Westinghouse has as a design basis that the fuel rod will not fail due to fuel centerline melting during Condition I and II operation. In order to assure that this basis is met, Westinghouse imposes a design limit on fuel temperatures such that there is a 95% probability that the peak linear heating rate (kW/ft) fuel rod will not exceed the UO2 melting temperature (Reference 2). The melting temperature of the UO2 is assumed to be 5080°F unirradiated and is decreased by 58°F per 10 MWd/kgM of exposure. A calculated centerline temperature of 4700°F has been selected by Westinghouse as the overpower limit. The fuel melting temperature dependence with fuel burnup is identical to that proposed by Christiansen (Reference 26). Christiansen presented two sets of fuel melting data versus fuel burnup with the above relationship presented by Westinghouse being the largest decrease with burnup and believed to be the better of the two. From this it is concluded that the Westinghouse criterion for fuel melting adequately acounts for the effects of extended burnup on fuel melting and thus is acceptable for extended burnup applications.

(e) Pellet/Cladding Interaction

Bases/Criteria - As indicated in SRP Section 4.2, there are no generally applicable criteria for pellet/cladding interaction (PCI) failure. However, two acceptance criteria of limited application are presented in the SRP for PCI: (1) less than 1% transient-induced cladding strain and (2) no centerline

fuel melting. Both of these limits have been adopted by Westinghouse for use in evaluating their fuel designs (References 2 and 12) and have been approved by the NRC (Reference 13) for current burnup applications. These are also found acceptable for extended burnup application.

Evaluation - Westinghouse uses the PAD 3.3 code (Reference 10) to show their fuel meets both the cladding strain and fuel melt criteria as discussed in Sections 2.0(b) and 3.0(d), respectively. As noted earlier, this code has been found acceptable for application to extended burnup fuel.

In addition, Westinghouse has presented (Reference 3) various power ramp data with rod average burnups up to approximately 46 MWd/kgM to show that PCI susceptibility does not increase at extended burnups. In fact, the trend of these data suggests that PCI susceptibility may decrease at extended burnups.

From the above, it is concluded that Westinghouse methods adequately address the effects of PCI at extended burnups.

(f) Cladding Rupture

Bases/Criteria - There are no specific design limits associated with cladding rupture other than the 10 CFR50 Appendix K requirement that the incident of rupture not be underestimated. The rupture model is an integral portion of the approved Westinghouse ECCS evaluation model (Reference 27). This is found acceptable for extended burnups.

Evaluation - The cladding deformation and rupture models used by Westinghouse in their LOCA-ECCS analysis are directly coupled to their models for cladding ballooning and flow blockage. A more detailed discussion of these models and their relation to extended burnup operation is provided in the section that addresses cladding ballooning and flow blockage, see Section 4.0(c). These models have been approved by the NRC (Reference 27) for current burnup levels and for the reasons stated in Section 4.0(c) are also found acceptable for extended burnup application.

Other parameters that are important to the LOCA analysis are those input to this analysis from the steady-state fuel performance code, PAD. There are two versions

of PAD, 3.3 (Reference 10) and PAD 3.3 Addenda 2 (Reference 28), used by Westinghouse to evaluate steady-state fuel performance of their designs. The PAD 3.3 version has been verified against fuel performance data with rod average burnups up to 57 MWd/kgM and, as noted earlier, has been approved for extended burnup analysis applications (including LOCA). The PAD 3.3 Addenda 2 version is a modification of PAD 3.3, in which conservatisms in the thermal model have been reduced. The PAD 3.3 Addenda 2 code has been verified against only thermal performance data at low burnups. Consequently, this code has been used for predicting early-in-life fuel thermal performance such as input for the LOCA analysis. Westinghouse has justified (Reference 28) the use of the (low burnup) PAD 3.3 Addenda 2 code for initializing steady-state thermal conditions input to their LOCA analysis of fuel designs at current burnup levels by showing that:

"the maximum peak clad temperature during a LOCA occurred using fuel parameters and initial conditions consistent with the time in life which exhibits the highest pellet average temperatures, near the beginning of life" (Reference 28).

Therefore, the use of the PAD 3.3 Addenda 2 code for initializing LOCA input has been approved by the NRC (Reference 29) for current burnup levels. In response to an NRC question during this review, Westinghouse has responded (Reference 4) that their original statement (given above) from the PAD 3.3 Addenda 2 review remains valid for all Westinghouse plant configurations and approved ECCS Evaluation Models for extended burnups up to those requested in this review. Westinghouse has stated that this has been verified by performing a series of calculations with PAD 3.3 on those parameters sensitive to extended burnup. These calculations have taken into account the reduced fuel rod powers at extended burnups due to fissile material burnout. This effect is a real phenomenon at extended burnups, because if extended burnup fuel rods were driven to the rod powers allowed by the Technical Specifications, other lower burnup fuel in the core would exceed the Technical Specifications on power distribution peaking factors. The PAD 3.3 code continues to be used for fuel design calculations that are burnup dependent.

(h) Fuel Rod Mechanical Fracturing

Bases/Criteria - The term "mechanical fracture" refers to a cladding defect that is caused by an externally applied force such as a load derived from core-plate motion or a hydraulic load. These loads are bounded by the loads of a safe-shutdown earthquake (SSE) and LOCA, and the mechanical fracturing analysis is usually done as a part of the SSE-LOCA loads analysis (see Section 5.0(d) of this SER).

Evaluation - The discussion of the SSE-LOCA loading analysis is given in Section 5.0(d) of this SER.

4.0 FUEL COOLABILITY

For accidents in which severe fuel damage might occur, core coolability must be maintained as required by several General Design Criteria (e.g., GDC 27 and 35). In the following paragraphs, limits and methods to assure that coolability is maintained are reviewed for the severe damage mechanisms listed in the Standard Review Plan.

(a) Fragmentation of Embrittled Cladding

Bases/Criteria - The most severe occurrence of cladding oxidation and possible fragmentation during a Condition III and IV accident results from a LOCA. Westinghouse uses the acceptance criteria of 2200°F on peak cladding temperature and 17% on maximum cladding oxidation as prescribed by 10CFR 50.46.

For events other than the LOCA, there are no separately established temperature or oxidation criteria. However, it is clear that for short-term events such as a locked rotor accident, the 2200°F peak cladding temperature and 17 percent oxidation LOCA criteria are not really meaningful, because the temperature history for such an event is much shorter than that of a LOCA. For events such as a locked rotor accident, Westinghouse uses (Reference 12) a peak cladding temperature (PCT) criterion of 2700°F.

The Westinghouse 2700°F PCT limit was selected taking into consideration the short time (a few seconds) that the fuel is calculated to be in DNB for a locked-rotor type event and the fact that the peak cladding temperature and total metal-water reaction at the fuel hot spot is not expected to impact fuel coolable geometry. The NRC has concluded (Reference 13) that the 2700°F peak cladding temperature limit for short-term undercooling events such as the locked rotor is an acceptable coolability limit for Westinghouse fuel designs at current burnup levels. This coolability limit is not judged to be changed by extended burnup operation and thus is found to be acceptable for extended burnup application.

Evaluation - The cladding oxidation models used to determine the amount of cladding fragmentation and embrittlement during the LOCA are not affected by extended burnup operation; however, the steady-state fuel performance PAD codes, used to provide input to the LOCA analysis, are burnup dependent. As noted earlier, Westinghouse has stated that early-in-life steady-state conditions are the most conservative for all their fuel designs. Consequently, Westinghouse has demonstrated that their LOCA analyses are insensitive to extended burnup operation because early-in-life conditions are most limiting. Therefore, the use of PAD 3.4 is found to be acceptable for LOCA analyses of Westinghouse extended burnup fuel.

(b) Violent Expulsion of Fuel

Bases/Criteria - In a severe reactivity initiated accident (RIA) such as a control rod ejection accident, the large and rapid deposition of energy in the fuel could result in melting, fragmentation, and dispersal of fuel. The mechanical action associated with fuel dispersal might be sufficient to destroy fuel cladding and the rod-bundle geometry and to provide significant pressure pulses in the primary system. To limit the effects of an RIA event, Regulatory Guide 1.77 recommends that the radially-averaged energy deposition at the hottest axial location be restricted to less than 280 cal/g.

The Westinghouse design limits for this event are:

(a) Average fuel pellet enthalpy at the hot spot will be below 225 cal/g for unirradiated fuel and 200 cal/gm for irradiated fuel.

(b) Average cladding temperature at the hot spot will be below the temperature at which cladding embrittlement may be expected (2700°F).

(c) Peak reactor coolant pressure will be less than that which could cause pressures to exceed the faulted condition stress limits.

(d) Fuel melting will be limited to less than 10 percent of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits in (a), above.

These limits are more conservative than the single 280 cal/g limit given in Regulatory Guide 1.77. They have been previously approved in the review of WCAP-7588 (Reference 31) and based on the above evaluation are found to be conservative and thus acceptable for extended burnup applications.

Evaluation - As discussed in Section 5 of this safety evaluation report, the methods used to calculate energy deposition as a result of reactivity insertion accidents (reactor physics codes) are also applicable to extended burnups. The extended burnup fuel is not expected to approach the 280 cal/gm criterion because fissile material burnout at extended burnups lowers the maximum possible fuel enthalpies when compared to maximum fuel enthalpies at lower burnups.

(c) Clading Ballooning

Bases/Criteria - Zircaloy cladding will balloon (swell) under certain combinations of temperature, heating rate, and stress during a LOCA. There are no specific design limits associated with cladding ballooning other than the 10 CFR 50 Appendix K requirement that the degree of swelling not be underestimated.

Evaluation - The Westinghouse cladding ballooning model is directly coupled to the cladding rupture temperature model for the LOCA-ECCS analysis and these are addressed in Revision 1 of WCAP-9220-P-A and WCAP-9221-A (Reference 27). These models have been approved by the NRC (Reference 27) for current burnup levels. The Westinghouse extended burnup topical report (Reference 2) references this approved report as being applicable to the LOCA analysis of extended burnup fuel.

There is evidence that cladding oxidation at extended burnup levels and LOCA temperatures may result in reduced cladding strains (Reference 32) from those traditionally predicted for LOCA. These data are not conclusive, however, because these tests were not performed with an oxidizing atmosphere nor under irradiation conditions. Irrespective of whether these data are applicable to a LOCA, reduced cladding strains would result in less flow blockage and thus the current analysis methods would be more conservative with respect to this criterion. In addition, the high cladding temperatures associated with the LOCA analysis will anneal any irradiation damage effects on cladding properties.

The steady-state operational input that is provided to the LOCA analysis from the PAD fuel performance codes is most limiting at early-in-life and thus are insensitive to extended burnup operation. Consequently, the ECCS models approved for application to current burnup fuel are also found to be acceptable for application to fuel at extended burnup levels.

From this evaluation, it is concluded that the Westinghouse methodology for calculating cladding ballooning during a LOCA is acceptable for extended burnup applications.

(d) Fuel Assembly Structural Damage From External Forces

Bases/Criteria - Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. SRP Section 4.2 and associated Appendix A state that fuel system coolability should be maintained and that damage should not be so severe as to prevent control rod insertion when required during these low probability accidents.

The Westinghouse design basis is that the fuel assembly will maintain a geometry that is capable of being cooled under the worst case accident Condition IV event and that no interference between control rods and thimble tubes will occur during a safe shutdown earthquake (Reference 12). This is nearly identical to the design basis presented in the SRP and is therefore acceptable for extended burnup operation.

Evaluation - Generic analysis methods for performing combined SSE-LOCA loading analyses have been described by Westinghouse in WCAP-9401-P-A (and WCAP-9402-A) (Reference 33). These analysis methods not only include the fuel assembly structural response but also fuel rod cladding loads. These methods have been approved by the NRC (Reference 33) for current burnup levels.

The material property that could have an impact on these analyses at extended burnup levels is material ductility. These analyses could be impacted if cladding or assembly ductility were decreased, as a result of extended burnup operation, to a level that would allow cladding or assembly failure not accounted for in the analysis. As noted in Section 2.0(a), the decrease in material ductility is expected to be negligible for the increased exposure and burnup levels requested and no adverse effects have been observed for rod average burnups up to 62 MWd/kgM.

From the above evaluation, it is concluded that the above analysis methods are acceptable for extended applications.

It should be noted that this analysis is plant specific, because it requires site specific input ground motions and thus cannot be completed in a generic manner. Therefore, an applicant for an operating license proposing to reference the extended burnup report (Reference 2) and the SSE-LOCA analysis methods must perform site specific analyses using References 33 and 34 analysis methods in order to address the above criteria and Appendix A to SRP Section 4.2 guidelines.

5.0 NUCLEAR DESIGN

Typical extended fuel burnup and increased fuel cycle length core designs utilize higher fuel enrichments, low leakage patterns and/or axial blankets. Higher fuel enrichment is required to reduce the number of feed assemblies and offset the reactivity loss resulting from the higher fission product inventory. The core neutron economy is improved by reducing the radial leakage using low leakage loading patterns in which the high burnup fuel is located on the core periphery. Axial blankets are used to flatten the axial burnup distribution and improve fuel utilization. The increased power peaking resulting from the

large reactivity differences between the fresh and high burnup fuel and the use of low leakage loading patterns is generally controlled using burnable poison rods.

These features affect the physics characteristics of high burnup core designs. The increased fuel depletion in high burnup cores results in an increase in the plutonium fission fraction and the fission product inventory, the higher plutonium fission fraction in turn hardens the neutron spectrum and increases the neutron production per unit energy. The increase fission product inventory and use of burnable absorbers tends to increase absorption and also harden the neutron spectrum.

While the increased fuel burnup does affect the core physics characteristics, the changes are relatively small and the physics parameters are determined using standard calculational methods and procedures. The high burnup neutronic effects enter through the microscopic cross sections and fuel assembly lattice group constants. The present calculations of these parameters account for substantial levels of plutonium, fission products and burnable absorbers, and these methods are expected to adequately treat the neutronics change associated with extended burnup. The depletion methods used to track the plutonium and fission product isotopics and various normalization procedures are also expected to be equally valid for high burnup fuel configurations.

The high burnup fuel physics characteristics and core configuration affect the core nuclear safety parameters. The major effect is to increase the power in the low burnup and/or centrally located fuel assemblies and to decrease the power in the high burnup and/or peripherally located fuel assemblies. The resulting increase in the number and power of the peak powered rods is typically controlled by use of burnable poison rods.

The increased fission product inventory and use of burnable absorbers increases thermal and epithermal absorption and hardens the core neutron spectrum. These factors combine to reduce the boron and control rod worth, prompt neutron lifetime and Doppler coefficient. The moderator temperature coefficient may increase or decrease depending on the particular high burnup design, and is also controlled using burnable absorbers as in present core designs. The delayed neutron fraction is also reduced as a result of the increased plutonium fission fraction.

In addition to improving the neutron economy, the low leakage patterns reduce the pressure vessel damage fluence by shifting the power toward the center of the core and away from the vessel. This fluence reduction is partially offset, however, by the harder neutron spectrum and increased neutron production (per MeV) of the high burnup fuel.

The calculation of the high burnup core safety parameters is carried out using the same core and lattice methods and procedures used for present core designs. The changes in the core safety parameters resulting from the higher fuel burnup designs tend to be relatively small as a result of the low relative importance of the high burnup fuel and the tendency for the increase in plutonium fission rates and fission product inventory to saturate. These calculated safety parameters provide the core neutronics input to the required plant transient and accident analysis.

As the above discussion indicates, the effect of high burnup on the physics design is expected to result in relatively small changes in the predicted characteristics of the core, and also relatively small extensions in range of the methods used to calculate the characteristics. Because high burnup fuel is not subject to limiting duty and because of its low relative importance in determining the core characteristics, we conclude that present methods are adequate for high burnup designs. To provide added assurance that these methods are adequate, we recommend that Westinghouse pay special attention to comparisons of predicted and measured physics parameters (particularly power distributions) which are monitored during the reactor cycle. A systematic pattern of deviation between predictions and measurements would provide an indication of potential problems. We intend to take an active role in following these comparisons.

6.0 RADIOLOGICAL CONSIDERATIONS OF POSTULATED ACCIDENTS WITH EXTENDED BURNUP OPERATION

To ensure that accidents involving the movement of fuel do not constitute an offsite health and safety issue, design events are assessed. Analyses of fuel handling accidents assume release of the entire volatile radionuclide fuel assembly gap and plenum inventory under nominally 23 feet of water after the

assembly has cooled substantially (usually at least 24 hours for BWR assemblies, 72 or 100 hours for PWR assemblies). For assemblies with burnup up to 38,000 MWd/t batch average at discharge, Regulatory Guide 1.25 assumptions are used. These stipulate an inventory of ten percent of the total fuel assembly iodines and noble gases (with the exception of 30 percent for ^{85}Kr) in the gap and plenum volumes released upon clad perforation. An iodine decontamination factor (DF) of 100 ("Evaluation of Fission Product Release and Transport for a Fuel Handling Accident," G. Burley, USAEC, Revised October 5, 1971) is assumed for 23 feet of water cover, and appropriate airborne radionuclide filtration/mixing, if any, is applied in the analysis before release to the atmosphere. The decontamination factor is based, in part, on an analysis of work presented in WCAP-7518-L, "Radiological Consequences of a Fuel Handling Accident," M. J. Bell, et al, June 1970, NES Proprietary Class 2.

For fuel handling accident offsite radiological consequence evaluations involving fuel assemblies with burnup > 38,000 MWd/t batch average at discharge (extended burnup assemblies), the analysis is presently performed using Regulatory Guide 1.25 assumptions, but with modified gap and plenum fractional volatile radionuclide inventories. The fractional inventories range from a few percent (less than the R. G. 1.25 ten percent recommendation) to as much as 40-50 percent for certain high burnups/radionuclide combinations. The gap and plenum fractional inventories for the highest-power assembly are computed as a function of at least burnup, and at most time, temperature, and burnup using the GAPCON-THERMAL-2 computer code in conjunction with the ANS 5.4 fission gas release standard (model) proposed by the American Nuclear Society in "Radioactive Gas Release from LWR Fuel," C. E. Beyer, draft NUREG CR-2715, April 1982. In generating these estimated fractional inventories, the conservative assumption of fuel assembly operation at a constant maximum-allowed peak linear heat generation rate (LHGR) for PWR's or MAPLHGR for BWR's is made. This assumption appears to be conservative within a factor of 2-3 for gap and plenum volatile inventories.

In addition to the conservative assumption regarding fuel assembly power operation noted above, there are two other significant sources of conservatism in the staff's analysis. The iodine decontamination factor (DF) assigned to the pool is taken to be a factor of 100. It can be inferred from the report upon which this factor is based (WCAP-7518-L) that this value is probably

conservative by about a factor of three. Finally, plateout of volatile iodine released from the fuel into the gap and fuel rod plenum has been entirely neglected. Although not well quantified, a tentative estimate suggests that about 10 percent or less of the iodine released into the gap will remain volatile at the fairly low temperatures after the fuel has been allowed to cool for about a day or more.

Because of the significance of these conservatisms, the staff intends to study and quantify them in more detail and to use the results of such evaluations to appropriately revise the staff's Standard Review Plan (SRP), NUREG-0800. In the interim, the staff concludes that consideration of all three factors together noted above may permit a significant reduction of estimated thyroid doses compared to existing analyses. Adequate justification by licensees on a case-by-case basis, or by vendors on a generic basis, are likely to provide sufficient bases for departing from SRP criteria until such time as detailed changes can be made. A reduction by a factor of two is likely to be appropriate and conservative. Consequently, with regard to evaluation of thyroid doses for fuel-handling accidents involving extended-burnup fuel ($>$38,000 MWd/tonne), and pending SRP revision, it is likely that justification can be provided for lower estimates of thyroid doses from fuel handling accidents by a factor of two in departures from SRP review criteria.

7.0 REGULATORY POSITION

The review of Westinghouse Electric Company's submittal, as described in WCAP-10125 (Proprietary) and responses to NRC questions in Reference 3 through 6, for application of their design criteria and analysis methods to extended burnups has been completed. As a result of our review, we conclude that these criteria and analysis methods are applicable to licensing of Westinghouse Zircaloy-clad fuel designs up to the extended burnup level requested in WCAP-10125.

From this evaluation, we have concluded that Westinghouse criteria and analysis methods, as described in the extended burnup topical report and response to questions, Reference 2 through 6, for extended burnup application are adequate such that 1) fuel damage is not expected to occur as a result of normal operation and anticipated operational occurrences (Condition I and II events), 2)

fuel damage during postulated accidents (Condition III and IV events) would not be severe enough to prevent control rod insertion when it is required, and 3) core coolability will always be maintained even after postulated accidents (Condition III and IV events).

This conclusion is based on two primary factors:

1) Westinghouse provided sufficient evidence that the design citeria will allow for safe operation of Westinghouse design fuel at the proposed extended burnup level; and
2) The Westinghouse analysis methods used to assure that these criteria are met have been based on adequate extended burnup operating experience and prototype testing.

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SECTION C

WESTINGHOUSE LETTER FROM E. P. RAHE,
"EXTENDED BURNUP EVALUATION OF WESTINGHOUSE FUEL,"
JULY 1982;
WCAP-10125 (PROPRIETARY), NS-EPR-2629,
DATED JULY 28, 1982,
TO NRC, J. R. MILLER

Westinghouse Electric Corporation
Water Reactor Divisions
Nuclear Technology Division
Box 355
Pittsburgh Pennsylvania 15230

July 28, 1982

NS-EPR-2629

Mr. James R. Miller, Chief
Special Projects Branch
Division of Project Management
U.S. Nuclear Regulatory Commission
Phillips Building
7920 Norfolk Avenue
Bethesda, Maryland 20014

Subject: "Extended Burnup Evaluation of Westinghouse Fuel", July 1982, WCAP-10125 (Proprietary)

Reference: Letter from L. S. Rubenstein (NRC) to T. M. Anderson (W), dated June 2, 1981

Dear Mr. Miller:

Enclosed are:

1. Twenty-five (25) copies of Westinghouse topical report, "Extended Burnup Evaluation of Westinghouse Fuel", July 1982, WCAP-10125 (Proprietary).

Also enclosed are:

1. One (1) copy of Application for Withholding (Non-Proprietary).
2. One (1) copy of original Affidavit (Non-Proprietary).

The enclosed topical report is in response to an NRC request (reference) for a topical report which justifies the validity of Westinghouse methods and criteria to evaluate Westinghouse fuel at extended burnups. This report generally follows your suggested outline attached to the reference letter. In addition, the acceptance criteria given in Section 4.2, "Fuel System Design" of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" have been addressed.

Note that at this time only the proprietary version of this topical report, appropriately bracketed to identify proprietary information, has been provided. This is consistent with the guidelines given in NUREG-0390, Vol. 6, No. 1, "Topical Report Review Status". Westinghouse will submit a corresponding non-proprietary version upon receipt of an NRC acceptance letter and SER, and/or when the licensing topical report is referenced in a specific license application in accordance with NUREG-0390.

This submittal contains proprietary information of Westinghouse Electric Corporation. In conformance with the requirements of 10CFR2.790, as amended, of the Commission's regulations, we are enclosing with this submittal an application for withholding from public disclosure and an affidavit. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission.

Correspondence with respect to the affidavit or application for withholding should reference AW-82-44 and should be addressed to R. A. Wiesemann, Manager of Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P.O. Box 355, Pittsburgh, Pennsylvania 15230.

Very truly yours,

E. P. Rahe, Jr., Manager
Nuclear Safety Department

MDB/kk
Enclosures

Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Technology Division

Box 355
Pittsburgh Pennsylvania 15230

July 28, 1982
AW-82-44

Mr. James R. Miller, Chief
Special Projects Branch
Division of Project Management
U. S. Nuclear Regulatory Commission
Phillips Building
7920 Norfolk Avenue
Bethesda, Maryland 20014

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

SUBJECT: "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125, July 1982 (Proprietary)

REF: Westinghouse Letter No. NS-EPR-2629, Rahe to Miller, dated July 28, 1982

Dear Mr. Miller:

The proprietary material transmitted by the reference letter is of the same technical type as material previously submitted concerning the Westinghouse reload safety evaluation methodology (Reference: NS-CE-1731, dated March 22, 1978). Further, the affidavit submitted to justify the material previously submitted, AW-78-27, is equally applicable to this material.

Approval of the application for withholding and affidavit AW-78-27 was received by NRC letter, Check to Wiesemann, dated October 23, 1978.

Accordingly, withholding the subject information from public disclosure is requested in accordance with the previously approved affidavit, AW-78-27, dated March 22, 1978, a copy of which is attached.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-82-44 and should be addressed to the undersigned.

Very truly yours,

for Robert A. Wiesemann, Manager
Regulatory & Legislative Affairs

/bek
Attachment

cc: E. C. Shomaker, Esq.
Office of the Executive Legal Director, NRC

Westinghouse Electric Corporation Power Systems

Nuclear Fuel Division

Box 355
Pittsburgh Pennsylvania 15230

March 22, 1978

AW-78-27

Mr. John F. Stolz, Chief
Light Water Reactors Branch No. 1
Division of Project Management
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Dear Mr. Stoltz:

SUBJECT: WCAP-9272, "Westinghouse Reload Safety Evaluation Methodology"

REF: Westinghouse Letter No. NS-CE-1731, Eicheldinger to Stolz, dated March 22, 1978

The proprietary material being transmitted by the referenced letter supplements the proprietary material previously submitted concerning the Westinghouse reload safety evaluation methodology. Further, the affidavit submitted to justify the material previously submitted, AW-76-31, is equally applicable to this material.

Accordingly, withholding the subject information from public disclosure is requested in accordance with our previously submitted affidavit and application for withholding, AW-76-31, dated July 27, 1976, a copy of which is attached.

Correspondence with respect to the proprietary aspects of this application should reference AW-78-27 and should be addressed to the undersigned.

Very truly yours,

C. Eicheldinger

Robert A. Wiesemann, Manager
Licensing Programs

Attachment

cc: Mr. J. A. Cooke, Esq.
Office of the Executive Legal Director, NRC

Westinghouse
Electric Corporation

Power Systems
Company

PWR Systems Division

Box 355
Pittsburgh Pennsylvania 15230

July 27, 1976
AW-76-31

Mr. D. G. Eisenhut
Assistant Director for Operational Technology
Division of Operating Reactors
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

APPLICATION FOR WITHHOLDING PROPRIETARY

INFORMATION FROM PUBLIC DISCLOSURE

SUBJECT: Westinghouse Reload Safety Evaluation Methodology

REF: Westinghouse Letter No. NS-CE-1142 Eicheldinger to Eisenhut dated July 27, 1976

Dear Mr. Eisenhut:

This application for withholding is submitted by Westinghouse Electric Corporation ("Westinghouse") pursuant to the provisions of paragraph (b)(1) of Section 2.790 of the Commission's regulations. Withholding from public disclosure is requested with respect to the subject information which is further identified in the affidavit accompanying this application.

The undersigned has reviewed the information sought to be withheld and is authorized to apply for its withholding on behalf of Westinghouse, WRD, notification of which was sent to the Secretary of the Commission on April 19, 1976.

The affidavit accompanying this application sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations.

Accordingly it is respectfully requested that the subject information which is proprietary to Westinghouse and which is further identified in the affidavit be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should be addressed to the undersigned.

Very truly yours,

Robert A. Wiesemann

Robert A. Wiesemann, Manager
Licensing Programs

/smh

Enclosure

cc: J. W. Maynard, Esq.
Office of the Executive Legal Director, NRC

AW-76-31

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Robert A. Wiesemann, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Corporation ("Westinghouse") and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Robert A. Wiesemann

Robert A. Wiesemann, Manager
Licensing Programs

Sworn to and subscribed
before me this 5 day
of July 1976.

Notary Public

MY COMMISSION EXPIRES APR. 15, 1978

(1) I am Manager, Licensing Programs, in the Pressurized Water Reactor Systems Division, of Westinghouse Electric Corporation and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing or rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Water Reactor Divisions.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse Nuclear Energy Systems in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(g) It is not the property of Westinghouse, but must be treated as proprietary by Westinghouse according to agreements with the owner.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition in those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information is not available in public sources to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in the attachment to Westinghouse letter No. NS-CE-1142, Eicheldinger to Eisenhut dated July 27, 1976 concerning reproductions of view-graphs used in the Westinghouse presentation to the NRC during the meeting on July 27, 1976 on the subject of Westinghouse Reload Safety Evaluation Methodology.

This information enables Westinghouse to:

(a) Justify the design for the reload core

(b) Assist its customers to obtain licenses

(c) Meet contractual requirements

(d) Provide greater flexibility to customers assuring them of safe and reliable operation.

Further, this information has substantial commercial value as follows:

(a) Westinghouse sells the use of the information to its customers for purposes of meeting NRC requirements for licensing documentation.

(b) Westinghouse uses the information to perform and justify analyses which are sold to customers.

(c) Westinghouse uses the information to sell nuclear fuel and related services to its customers.

Public disclosure of this information is likely to cause substantial harm to the competitive position of Westinghouse in selling nuclear fuel and related services.

Westinghouse retains a marketing advantage by virtue of the knowledge, experience and competence it has gained through long involvement and considerable investment in all aspects of the nuclear power generation industry. In particular Westinghouse has developed a unique understanding of the factors and parameters which are variable in the process of design of nuclear fuel and which do affect the in service performance of the fuel and its suitability for the purpose for which it was provided.

In all cases that purpose is to generate energy in a safe and efficient manner while enabling the operating nuclear generating station to meet all regulatory requirements affected by the core loading of nuclear fuel. Confidence in being able to accomplish this comes from the exercise of judgement based on experience.

Thus, the essence of the competitive advantage in this field lies in an understanding of which analyses should be performed and in the methods and models used to perform these analyses. A substantial part of this competitive advantage will be lost if the competitors of Westinghouse are able to use the results of the Westinghouse experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions. Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design and licensing of a similar product.

This information is a product of Westinghouse design technology. As such, it is broadly applicable to the sale and licensing of fuel in pressurized water reactors. The development of this information is the result of many years of Westinghouse effort and the expenditure of a considerable sum of money. In order for competitors of Westinghouse to duplicate this process

would require the investment of substantially the same amount of effort and expertise that Westinghouse possesses and which was acquired over a period of more than fifteen years and by the investment of millions of dollars.

Further the deponent sayeth not.

SECTION D

WCAP-10126-NP-A
TEXT

Table of Contents

Table of Contents (Cont'd)

Table of Contents (Cont'd)

2209F:6

<u>List of Illustrations</u>

List of Tables

Abstract

At the request of the Nuclear Regulatory Commission (NRC), this report provides information which justifies the validity of Westinghouse fuel system design and safety analyses methods and criteria to a lead rod average burnup of [ ]$^{+}$ MWD/MTU. (a,c

SUMMARY

This report evaluates the impact of extended burnup on the design and operation of Westinghouse fuel. Each area of potential concern is examined in Section 2.0 with respect to the impact of extended burnup to show that applicable design criteria, performance models and methodology are sufficient for design to a target lead rod average burnup of [ ]$^{+}$ MWD/MTU. In Section 3.0, the Westinghouse extended burnup (a,c) operating experience and performance data base is summarized. Westinghouse performance models and methodology are qualified with respect to these data for design to the target burnup. Current and future experimental programs which will provide confirmatory data at extended burnup are discussed in Section 4.0, and conclusions are presented in Section 5.0.

The information supplied in this report supports the conclusion that Westinghouse design methods and safety analyses are valid for operation to a lead rod average burnup of [ ]$^{+}$ MWD/MTU. No performance (a,c) limitations have been identified which would preclude the design of Westinghouse fuel to this target burnup, and it has been shown that current design and safety evaluation criteria can be applied with no modification to these criteria. In addition Westinghouse has already acquired sufficient data at extended burnup to justify operation to lead rod average burnup of [ ]$^{+}$ MWD/MTU. It is therefore concluded (a,c) that Westinghouse designed fuel can achieve a lead rod average burnup of (a,c) [ ]$^{+}$ MWD/MTU.

1.0 INTRODUCTION

1.1 BACKGROUND

Based upon economic and resource considerations, it may become desirable to increase Westinghouse fuel region average discharge burnups beyond the traditional levels of approximately 33000 MWD/MTU. This would result in improved uranium utilization, potential fuel cycle cost savings and a reduced demand on spent fuel storage capacity. The Department of Energy (DOE) has been funding programs with all domestic fuel vendors in order to pursue the improved uranium utilization of high burnup.

As shown in Section 3.1 and in reference 1, Westinghouse has had considerable fuel operating experience at extended burnup. Experience to date suggests that in progressing to extended burnups, no sudden or unexpected change in performance occurs. No fuel failures, attributable solely to lifetime in reactor or achieved burnups, have been observed in a power reactor. Fuel performance is more sensitive to power and operating clad temperatures than to lifetime in a reactor. With Westinghouse experience as of January 1982, approximately 230 commercial design fuel assemblies have successfully achieved burnups in excess of 36000 MWD/MTU, with 78 in excess of 38000 MWD/MTU. Several utilities have a number of fuel assemblies irradiated to burnups of greater than 40000 MWD/MTU. Recently in a cooperative program between DOE, the Virginia Electric Power Company, and Westinghouse, a 17x17 demonstration assembly in the Surry Unit 2 reactor achieved a burnup of 43200 MWD/MTU, while in a joint program with the Electric Power Research Institute (EPRI), Commonwealth Edison Company, and Westinghouse in the Zion Unit 1 reactor, four standard 15x15 design fuel assemblies have been discharged after achieving an average burnup of 55000 MWD/MTU. Several high power fuel rods in the Jose Cabrera Reactor have achieved rod average burnup greater than 55000 MWD/MTU, and in the ongoing DOE-sponsored examination of Westinghouse fuel rods, rods have been irradiated in the Centre D'Etude De L'Energie Nucleaire (CEN) BR-3 reactor to rod average burnups as high as 61500 MWD/MTU.

Early in 1981, the NRC initiated an evaluation of the current state of the art and the additional data needed for licensing LWR fuel to extended burnups. Westinghouse extended burnup experience, including the burnup impact on design models and on reload analyses, were addressed at a March 27, 1981 meeting with the NRC. Subsequent to informational meetings with each of the five U.S. fuel vendors, the NRC requested[2] that each vendor submit an extended burnup topical report which covers extended burnup experience, methods and test data. This report is in response to that request.

1.2 OBJECTIVES

The purpose of this topical is to justify operation of Westinghouse designed fuel to a lead fuel rod average burnup of [ ]+ MWD/MTU. (a,c)
This report provides the necessary information to assure the validity of the current NRC approved Westinghouse design methods and safety analyses to a target lead fuel rod average burnup of [ ]+ MWD/MTU. (a,c)

2.0 ISSUES CONCERNING EXTENDED BURNUP

2.1 INTRODUCTION

Included in this section is a discussion of the specific parameters affected by extended burnup operation. The changes in fuel duty associated with extended burnup include increased residence time, fuel burnup, fast fluence, and some small changes in power history. Fuel rod and fuel assembly structural performance parameters have been evaluated with respect to the above changes in fuel duty to assure that all design criteria can be met for lead rod burnups of [ ]+ MWD/MTU. In (a,c)
addition to fuel performance parameters, nuclear design parameters have also been evaluated with respect to extended burnup operation to show that all of the effects of extended burnup operation are accommodated. Finally, the effects of extended burnup operation identified in all of the above areas have been assessed with respect to the current Westinghouse safety evaluation methodology to show that current safety analysis limits and procedures are sufficient.

2.2 FUEL ROD PERFORMANCE

2.2.1 Clad Oxidation and Hydriding

The design basis for clad oxidation and hydriding is that the fuel system will not be damaged due to excessive clad oxidation and hydriding.

2.2.1.1 Clad Oxidation

The design limits applied to clad oxidation evaluations are that calculated clad temperature (oxide to metal interface) shall be less than
[$]^{+o}$F during steady state operation, and for Condition II trans- (a,c)
ients the metal to oxide interface temperature shall not exceed
[$]^{+o}$F. These clad temperature limits are required to preclude a (a,c)
condition of accelerated oxidation.

The primary impact of extended burnup operation on clad oxidation is due to the increased residence time at operating conditions. The factors that are considered to control in-reactor corrosion for Zircaloy-4 are the temperature (metal-to-oxide interface) and irradiation enhancement to the degree applicable. Extensive experience [3] has shown that irradiation enhancement of the corrosion rate does not occur under normal operating conditions in current Westinghouse PWR's. The controlling factor for the in reactor Zircaloy corrosion rate is therefore the oxide to metal interface temperature, which is controlled by the above clad temperature criteria.

The NRC approved Westinghouse fuel performance code[4] is used to evaluate the clad surface temperature criterion. A value of [ ]$^{+}$ (a,c)
mils crud thickness (typical of that observed in operating reactors) is assumed in the clad temperature evaluation. Typical results at extended burnup show that the peak clad temperature throughout life is less than
[ ]$^{+o}$F for steady state operation and less than [ ]$^{+o}$F for (a,c)
Condition II transients.

Data on the corrosion behavior of Westinghouse fuel cladding is presented in Section 3.2.1. These data show that bounding limits of clad oxidation thickness for both normal and high power rods are acceptable at burnups in excess of [ ]$^{+}$ MWD/MTU. Both normal and high (a,c)
crud thickness rods are included in the data base.

2.2.1.2 Clad Hydriding

Clad hydrogen pickup limits are required to prevent loss of mechanical properties of the clad due to hydrogen embrittlement by the formation of Zirconium hydride platelets. The design limit is that the hydrogen pickup level shall be less than or equal to [ ]$^{+}$ ppm at the end of (a,c)
the projected life. Westinghouse test results indicate that the
[ ]$^{+}$ ppm criterion does not adversely affect the mechanical properties of Zircaloy-4. (a,c)

Hydride platelet orientation can significantly affect the ductility of Zircaloy 4 components. The clad texture determines whether the preferential hydride orientation is circumferential or radial (radial hydride orientation reducing fracture ductility). Process controls and texture acceptance tests preclude this problem for Westinghouse cladding.

Extended burnup operation impacts clad hydriding primarily due to increased residence time. Conservative calculations at extended burnup show typical maximum hydrogen pickup to be [ ]$^+$ ppm, (a,c)
which is well within the [ ]$^+$ criterion. (a,c)

2.2.2 Rod Internal Pressure

The design basis for fuel rod internal pressure is that the fuel system will not be damaged due to excessive fuel rod internal pressure.

The current NRC approved design limit for fuel rod internal pressure is that the internal pressure of the lead rod in the reactor will be limited to a value below that which could cause (1) the diametral gap to increase due to outward cladding creep during steady state operation and (2) extensive DNB propagation to occur.[5] This limit is applied to extended burnup fuel rod design.

Typical design fuel rod internal pressure limit values which preclude gap increase and DNB propagation are [ ]$^+$ psia above (a,c)
system pressure for 17 x 17 designs and [ ]$^+$ psia (a,c)
above system pressure for 15 x 15 and 14 x 14 designs.

Fuel rod internal pressure is more limiting at extended burnup due to the additional fission gas inventory at extended burnup and potentially, to a change in the release characteristics of the fuel. Data also show that the fission gas release increases as a function of rod power. Since current peaking factor limits will be met in extended burnup design, extended burnup fuel will not experience peak power levels higher than those which are evaluated for current fuel rod design. However, extended burnup fuel at higher initial enrichment may

operate at these power levels for a longer time, potentially resulting in an increased fission gas release fraction. Data on fission gas release at extended burnup is discussed in detail in Section 3.2.2.

Evaluation of the fuel rod internal gas pressure as a function of irradiation time is performed using the Westinghouse burnup dependent fission gas release model presented in Reference 4. This model is based on measured fission gas release data which includes fuel rods operated at high power levels at burnups ranging from several MWD/MTU to greater than 57000 MWD/MTU. Limiting power histories consistent with extended burnup operation are used in the rod internal pressure evaluation to assure that limits are met.

A typical value of fuel rod internal pressure calculated for a 15 x 15 fuel rod operated to a burnup in excess of [ ]$^{+}$ MWD/MTU, is (a,c)
[ ]$^{+}$ psia. This value is compared to the limiting value of (a,c)
[ ]$^{+}$ psia for the 15 x 15 fuel rod design. (a,c)

2.2.3 Clad Stress

The design basis for fuel rod clad stress is that the fuel system will not be damaged due to excessive fuel cladding stresses.

The design limit for fuel rod clad stress is that the volume average effective stress calculated with the Von Mises equation considering interference due to uniform pellet - cladding contact, caused by pellet thermal expansion, pellet swelling, uniform cladding creep, and pressure differential is less than the Zircaloy 0.2% offset yield strength, with due consideration to temperature and irradiation effects under Condition I and II modes of operation. While the cladding has some capability for accomodating plastic strain, the yield stress has been accepted as a conservative design limit.

The NRC approved Westinghouse fuel performance code [4] is used for evaluating clad stress limits. Both steady-state (Condition I) and transient (Condition II) conditions are evaluated; however, analyses

have shown that transient clad stresses are most limiting and that the limiting time in life is near the end of Cycle 2 of operation. Transient clad stress at high burnup is not limiting since the rod power capability is reduced as a function of burnup, thus limiting the power excursion which can be experienced by an extended burnup rod during a Condition II event. Fuel rod clad stress is not, therefore, a concern for extended burnup fuel rod design.

Typical design values of clad effective stress during Condition I operation are [ ]$^+$ psi. A typical design value of the irradiated 0.2% offset yield stress for a Condition II transient is [ ]$^+$ psi. Typical design values of clad effective stress for a Condition II transient are [ ]$^+$ psi.

(a,c)
(a,c)
(a,c)

2.2.4 Clad Strain

The design basis for fuel rod clad strain is that the fuel system will not be damaged due to excessive fuel cladding strain.

The design limit for fuel rod clad strain during steady state operation is that the total plastic tensile creep and uniform cylindrical fuel pellet expansion due to fuel swelling and thermal expansion is less than 1 percent from the unirradiated condition. For Condition II transients, the design limit for cladding strain is that the total tensile strain due to uniform cylindrical pellet thermal expansion during the transient is less than 1 percent from the pretransient value. These limits are consistent with proven practice.

During steady state operation, tensile clad creep strain results primarily from cladding stresses caused by pellet swelling and thermal expansion following the closing of the pellet-clad gap. Since rod power levels, and hence fuel temperature, decrease as a function of burnup the fuel pellet diameter increase at extended burnup caused by the fuel swelling effect is somewhat mitigated by the reduced thermal expansion. Evaluation of clad strain during steady state operation is performed using the NRC approved Westinghouse fuel performance code [4].

Analyses show that steady state clad strain for rods irradiated to [ ]$^+$ MWD/MTU are well within the design limit, with a typical limiting value of total tensile creep strain of [ ]$^+$ percent. (a,c (a,c

For Condition II transients analyses have shown that tensile creep strains, resulting from pellet thermal expansion, are most limiting at the end of the second cycle of operation when transient clad stress is most limiting. Results also show that transient strain criteria are less limiting than transient stress criteria, and therefore transient strain limits are always met when transient stress limits are met. These evaluations are performed using the NRC approved Westinghouse fuel performance code [4], and a typical limiting value of transient clad strain is [ ]$^+$ percent. Since the limiting condition occurs near the end of the second cycle of operation, transient clad strain is not a concern for extended burnup fuel design. (a,c

2.2.5 Clad Fatigue

The design basis for fuel rod clad fatigue is that the fuel system will not be damaged due to clad strain fatigue.

The design limit for clad strain fatigue is that the fatigue life usage factor is less than 1.0. That is, for a given strain range the number of strain fatigue cycles are less than those required for failure, considering a minimum safety factor of 2 on the stress amplitude or a minimum safety factor of 20 on the number of cycles, whichever is more conservative.

Extended fuel rod, burnup up to [ ]$^+$ MWD/MTU may result in additional fatigue load cycles. Therefore, the strain fatigue evaluation must span the projected extended burnup design lifetime. (a,c

The evaluation of the fatigue life usage factor for extended burnup design conservatively assumes daily load follow operation over the life of the fuel rod. The NRC aproved Westinghouse fuel performance code [4]

is used to determine the strain range for the fatigue life usage analysis. The Langer - O'Donnell fatigue model [6] constitutes the basic approach taken in the fatigue analysis, with the empirical factors of their correlation modified in order to conservatively bound the results of the Westinghouse testing program.

Westinghouse fatigue test programs and the resulting data on which the design strain fatigue life is based are presented in Reference 7. The design equations which are used follow the concept for the fatigue design criterion according to the ASME Boiler and Pressure Vessel Code, Section III.

A typical value of strain fatigue life usage for Westinghouse fuel rods operated to [ ]$^+$ MWD/MTU burnup is [ ]$^+$. This (a,c) value is based on evaluations which assumed daily load follow over the life of the fuel rod, where the load follow operation is assumed to be a typical ramp pattern described as a reference "12-3-6-3" daily load cycle. The plant runs at full power for 12 hours, ramps down to the lower power level over three hours, maintains the lower power level for six hours, and ramps back to full power in three hours. For this evaluation the power level changes for a load cycle are conservatively assumed to be from 100% full power to 15% of full power.

2.2.6 Fuel Temperature

The design basis for fuel temperature is that the fuel rod will not fail due to fuel centerline melt during Condition I and Condition II operation.

The design limit for fuel temperature analyses is that during Condition I and Condition II modes of operation, there is at least a 95% probability that the peak KW/ft fuel rods will not exceed the UO_2 melting temperature. The melting temperature of unirradiated UO_2 is taken as 5080°F, decreasing by 58°F per 10000 MWD/MTU exposure. At

[ ]+ MWD/MTU, this corresponds to a melting temperature of (a,c
[ ]+°F. A calculated centerline fuel temperature of 4700°F has (a,c
been selected as the overpower limit.

Design evaluations of fuel temperature are performed using the NRC approved Westinghouse fuel performance code [4]. Analyses have shown that fuel temperatures are maximum at beginning of life where the pellet-clad gap is a maximum. Following pellet-clad contact, fuel temperature decreases significantly.

Since fuel rod internal pressure criteria preclude the opening of the pellet-clad gap during normal operation at high burnup, it is concluded that fuel temperature is limiting at beginning of life. Therefore evaluation of fuel temperature limits is not affected by extended burnup design.

2.2.7 Clad Flattening

The design basis for fuel rod clad flattening is that fuel rod failures will not occur due to clad flattening.

The design limit for fuel rod clad flattening is that the core residence time will not exceed the calculated core residence time which corresponds to a flattened rod frequency of 1.0.

Clad flattening is dependent on residence time at operation conditions, and not on burnup per se. Longer residence time is typically required to achieve higher fuel rod burnup, and therefore the clad flattening analysis must bound the projected extended burnup residence time. The
lead rod average burnup of [ ]+ MWD/MTU may correspond to resi- (a,c)
dence time up to [ ]+. (a,c

Evaluation of fuel clad flattening is performed using the NRC approved Westinghouse clad flattening model [8]. Calculations performed for current generation stable fuel in pre-pressurized rods show that

predicted clad flattening time exceeds residence times expected for extended burnup fuel management. Typical values of clad flattening times are in excess of 45000 effective full power hours.

2.2.8 Pellet Clad Interaction

The concern that a fuel rod may become more susceptible to pellet cladding interaction (PCI) type fuel failure at extended burnup has been addressed based on Westinghouse and other available PCI data. This concern was raised due to the increase in fission product inventory and increased pellet-clad contact resulting from increased clad creepdown at extended burnup. Based on an evaluation of available PCI data presented in Section 3.2.6, it has been concluded that the fuel rod failure susceptibility due to PCI does not significantly increase with extended burnup.

Two criteria which have been recognized to have a relationship to PCI have also been evaluated. The first criterion is that the uniform clad strain experienced during a transient is limited to one percent. This criterion was addressed in Section 2.2.4, Clad Strain, where it was concluded that transient clad strain, like transient clad stress, is typically most limiting at the end of the second cycle of operation and is not, therefore, a concern for extended burnup operation. The second criterion is that fuel melting should be avoided. This also is not a concern for extended burnup since, as noted in Section 2.2.6, Fuel Temperature, fuel temperatures are maximum at beginning of life. It is therefore concluded that PCI is not a limiting concern for extended burnup operation.

2.3 FUEL ASSEMBLY STRUCTURAL PERFORMANCE

2.3.1 Fuel Rod Fretting

The design basis for clad fretting wear is that fuel rods shall be designed not to fail due to clad fretting wear during Conditions I and II operation.

Westinghouse uses [ ]$^{+}$ percent wall thickness reduction as a general guide in evaluating clad imperfections, including fretting wear marks. The design criteria for clad fretting wear are that clad stress and fatigue limits discussed in Sections 2.2.3 and 2.2.5, respectively, must be met. Fretting on the fuel rod clad surfaces will not have a significant effect on stresses. The wear is local and on the clad outer diameter where stresses are lower than in other clad areas. (a,c)

Fuel rod fretting is affected by both the increased fluence and increased residence time associated with extended burnup. For Inconel grids the increased fluence results in a slight increase in grid spring relaxation and the increased residence time may result in a slight increase in the wear volume. For Zircaloy grids the effect of increased fluence on grid spring relaxation is not very detrimental, because the grid springs are expected to be fully relaxed before the maximum burnup is reached. However, increased wear is expected due to the increased residence time at zero grid spring force.

Evaluation of clad fretting wear is performed on the basis of empirical data taken from operating reactors and fretting wear tests. These data, discussed in Section 3.3.1, demonstrate that fretting wear limits will be met for extended burnup operation for both the Inconel and the Zircaloy grid designs.

2.3.2 Zircaloy Oxidation and Hydriding

2.3.2.1 Zircaloy Oxidation

An acceptable corrosion rate is required to ensure that mechanical strength and behavior of the grids and guide thimble tubes are not reduced due to the effective loss of metal during reactor operation.

Zircaloy structural component corrosion is affected due to the increased residence time associated with extended burnup operation. However, since the temperature of the grids or guide thimble tubes remain relatively constant at the value of the local coolant temperature,

extended burnup does not impact the Zircaloy corrosion rate. The additional oxidation of Zircaloy grids and guide thimble tubes due to the additional residence time associated with extended burnup has been evaluated using an isothermal corrosion rate dependent on the location in the core. Bounding coolant temperatures of [ ]+ (a,c)
at the mid plane and [ ]+ at the top of the core were considered (a,c)
in the evaluation. Results of these bounding evaluations indicate that Zircaloy structural component corrosion levels are acceptable at lead rod average burnup of [ ]+ MWD/MTU. (a,c)

2.3.2.2 Zircaloy Hydriding

In addition to the maximum oxidation concerns, a hydrogen pickup limit of [ ]+ ppm is required for the Zircaloy structural components in (a,c)
order to preclude excessive embrittlement by hydride precipitation. Westinghouse R&D tests have been performed on the effect of hydriding on the corrosion and mechanical properties of Zircaloy grid straps, and data indicates that a hydrogen concentration of up to [ ]+ ppm will (a,c)
not degrade the corrosion or mechanical properties of the Zircaloy 4 grid strap material. Additional data on Zircaloy hydriding found in the open literature (References 10 through 16) support the use of the [ ]+ ppm limit. (a,c)

Zircaloy hydriding is affected by extended burnup due to increased residence time. Conservative calculations of maximum hydrogen pickup after [ ]+ months of operation result in hydrogen concentrations of (a,c)
approximately [ ]+ for the Zircaloy grid straps and (a,c)
guide thimble tubes, respectively. These values are well within the [ ]+ ppm limit. (a,c)

The possibility of hydride reorientation into the less desired thickness direction of the grid straps has been addressed and found to be acceptable. Significant reorientation in the thickness direction could potentially cause a drastic reduction in the apparent ductility and strength properties for even moderate hydrogen levels. The texture of forged Zircaloy sheet orients essentially all hydrides into the rolling

and transverse directions of the sheet. [

]+ (a,c)

2.3.3 Fuel Rod Growth Gap

An initial fuel rod-to-nozzle growth gap must be provided in design to allow for the differential irradiation growth between the fuel rod cladding and the fuel assembly guide thimble tubes. Initial gaps for Westinghouse fuel assembly designs, with typical values slightly in excess of [ ]+% of the fuel rod length, are designed to assure (a,c) acceptable growth gap distributions at lead rod average burnup of [ ]+ MWD/MTU. Extended burnup impacts the fuel rod-to-nozzle (a,c) growth gap primarily due to the increased fluence associated with extended burnup. Increased fluence results in increased irradiation growth of the Zircaloy fuel cladding and guide thimble tubes. The additional growth of the annealed Zircaloy guide thimble tubes is small relative to the growth of the cold worked cladding, resulting in an increased reduction in the rod-to-nozzle gap. The Westinghouse criterion for fuel rod-to-nozzle growth gap potentially allows for a small finite number of gap closures.

As discussed in Section 3.3.3, the initial fuel rod-to-nozzle growth gap is designed based on measured growth gap data as a function of fluence. The distribution of measured gap data is determined as a function of fluence and is statistically combined with the distribution of rod average fluences for an assembly with a lead rod average burnup of [ ]+ MWD/MTU. The initial rod-to-nozzle gap is determined to (a,c) provide acceptable rod-to-nozzle clearance for operation to lead rod burnup of [ ]+ MWD/MTU. (a,c)

2.3.4 Fuel Rod Bow

The effects of fuel rod bow (i.e., dimensional changes) on DNBR and power peaking are included in design analysis as discussed in WCAP-8691 Revision 1[17]. Therefore, specific limits on fuel rod bow are not required. An evaluation of fuel rod bow in Westinghouse fuel assembly designs, including the impact of burnup on rod bow is provided in WCAP-8691, Revision 1.

As noted in WCAP-8691, Revision 1, the amount of fuel rod bowing has been observed to increase with burnup. The resultant rod bow correlations reflect this trend and consequently the magnitude of the rod bow DNBR effect, and the rod bow power peaking factors also increase with burnup. However, by the time the fuel attains an assembly average burnup of 33000 MWD/MTU, it is not capable of achieving limiting peaking factors due to the decrease in fissionable isotopes and the buildup of ission product inventory. This physical burndown effect is greater than the rod bowing effects which would be calculated based on the amount of bow predicted at those burnups. Therefore, for the purpose of evaluating effects of rod bow, 33000 MWD/MTU represents the maximum burnup of concern so that rod bow is not a limiting concern for lead rod average burnup of [ ]$^{+}$ MWD/MTU. This conclusion is supported by (a,c)
data presented in Section 3.3.4.

2.3.5 Assembly Holddown Spring Force

The fuel assembly holddown springs are designed to keep the fuel assemblies in contact with the lower core plate under all Condition I and II events with the exception of the turbine overspeed transient associated with a loss of external load. A turbine overspeed transient following loss-of-load may cause a fuel assembly lift for a short period of time. Such lifting is expected to be infrequent, but all fuel assembly holddown springs will accommodate the added deflection and function normally following the transient.

The fuel assembly holddown springs will be affected by extended burnup in two ways. First, the higher fluence associated with extended burnup will cause additional fuel assembly growth. This growth will cause additional deflection of the holddown springs and will result in an increased holddown force. The second effect is that the higher fluences will cause additional irradiation relaxation of the spring material. Such relaxation will result in a decreased holddown force.

Past analyses of Westinghouse leaf-type holddown spring performance predict a net increase in holddown force (i.e., the increase due to fuel assembly growth more than compensated for the decrease due to relaxation) as the fuel assembly experiences additional irradiation. Fuel assembly length measurements at Zion Unit 1 confirm that fuel assembly lengths are increasing, as expected, with burnup. Also PIE of Westinghouse leaf-type springs at Point Beach Unit 1 confirmed that the spring rates remained within the range specified for as-built, unirradiated holddown springs. It is therefore concluded that fuel assembly holddown spring force is not a limiting concern for extended burnup design since expected assembly growth compensates for the relaxation of the spring as a function of fluence.

2.3.6 Guide Thimble Tube Wear

Fuel assembly guide thimble tube wear has been observed in Westinghouse designed cores. The most severe wear occurs in the top 8 inches of guide thimbles in fuel assemblies located under Rod Cluster Control Assemblies (RCCAs) in the fully withdrawn position. Guide thimble wear due to interaction with RCCAs in the controlling bank or with thimble plugging devices is not significant and does not require evaluation. It has been determined that guide thimble tube wear due to interaction with RCCAs in the fully withdrawn position is primarily the result of a combination of flow-induced rod vibration and mechanical misalignment. This wear phenomenon has been evaluated for all current design variations including 14x14, 15x15, 16x16, 17x17, standard and optimized, 12 foot and 14 foot designs.

Guide thimble tube wear may be impacted by extended burnup operation primarily due to the additional residence time associated with extended burnup. The increased residence time may result in increased fuel assembly duty under a RCCA in the fully withdrawn position which could result in increased guide thimble wear.

Guide thimble tube wear has been evaluated based on the thimble wear data and the guide thimble wear model[18, 19, 20] which are discussed in Section 3.3.5. These evaluations have shown that all Westinghouse fuel designs can operate under an RCCA in the fully withdrawn position for up to 225 weeks (4.3 years). This is equivalent to a minimum of 5 annual cycles (5 years) of reactor operation with an 86 percent capacity factor.

Extended burnup operation to a lead rod average burnup of [ ]+ MWD/MTU will not require greater than 225 weeks residence time under a fully withdrawn RCCA. Therefore, this guide thimble tube wear criteria will be met for extended burnup. (a,c)

2.4 NUCLEAR DESIGN PARAMETERS

In this section the effects of extended burnup on nuclear design parameters are discussed. Fuel burnup is typically increased by increasing the average residence time in the core through a reduced number of feed assemblies for either a fixed or an increased (i.e., 18 month cycle) cycle length. A result of having fuel with higher burnup in the core (higher core average burnup) is an increase in the core fission product inventory. The reduced number of feed assemblies and increased fission product inventory leads to a need for higher enrichment of feed assemblies and causes a larger difference in K_∞ between fresh and burnt assemblies.

In addition to the extended burnup, some other features which improve fuel economics and utilization are also incorporated in the fuel management schemes leading to extended burnup. These include 18-month cycles, low leakage loading patterns and improved burnable absorbers. Presence

of the high reactivity fuel at the core periphery, as in the conventional loading patterns, leads to high neutron leakage. Low leakage loading patterns, with preferential fresh fuel placement inboard, are advantageous for better fuel utilization. Such economically advantageous fuel management schemes further increase the power peaking due to a larger reactivity difference between fresh and high burnup fuel. Burnable absorber rods are used to control the power peaks. For an 18-month cycle, the use of burnable absorbers for power distribution control does not typically constitute an additional constraint because they are also required for moderator temperature coefficient control, at current burnup levels.

The design parameter changes in the extended burnup core are mainly due to (in addition to design features to improve fuel utilization discussed above):

i. harder spectrum - use of burnable absorbers produces a more absorbing and a dryer core. Higher enrichment and larger plutonium and fission product concentration also make the spectrum harder.

ii. larger fraction of fissions in plutonium.

iii. more competition for neutron absorption.

The primary nuclear parameters affected by extended burnup due to the factors mentioned above are:

a. Power Peaking Factors
The use of a low leakage loading pattern, a larger difference in the reactivities of the fresh and burnt fuel, and a harder spectrum combine to produce higher power peaks. To a large extent these peaks can be minimized by using burnable absorbers having a depletion rate which keeps the maximum power slowly decreasing with burnup. Without the use of burnable absorbers, the power gradients in an extended burnup core would be larger

compared to the conventional discharge burnup core. However, when the required burnable absorbers are used, power shape gradients are quite comparable. Therefore, the uncertainties in the power peak predictions would not be affected by extended burnup operations.

b. Effective Delayed Neutron Fraction and Prompt Neutron Lifetime
Changes in these parameters are small. Effective delayed neutron fraction tends to be lower due to the larger fraction of fissions in plutonium, but the change is small due to the shift in power sharing in favor of the fresh fuel. Prompt neutron lifetime decreases slightly as the spectrum becomes harder and the core more opaque.

c. Moderator Temperature Coefficient
For a fixed concentration of soluble boron the moderator temperature coefficient is more negative for extended burnup fuel management. On the other hand, the associated 18-month cycle length soluble boron concentration can be higher to control excess reactivity. Therefore, with the use of an appropriate number of burnable absorbers, which are also required for power peaking control, moderator temperature coefficient can be controlled within the required limits at the beginning of cycle. At the end of cycle, the moderator temperature coefficient is more negative for extended discharge burnup fuel management, but is still within the current limits used for the safety analysis.

d. Control Rod Worth
The larger competition for absorption and harder spectrum in the extended burnup fuel management reduce the control rod worth. The reduction in control rod worth could slightly reduce the shutdown margin but the control rod worth would still be above the required limits for steamline break analysis. The changes would be comparable to those for the normal design variations and would be handled by the normal

safety evaluation methodology[21]. On the other hand, the reduced rod worth would improve safety margin for the other control rod dependent accidents like those due to dropped rod, rod misalignment, stuck rod and rod ejection.

e. Doppler Coefficient

The Doppler coefficient is slightly more negative for extended burnup schemes but is within the acceptable range.

f. Boron Worth

Soluble boron worth would be somewhat smaller for the extended burnup schemes. This is similar to the effect of a transition to an 18-month cycle. Reduced boron worth would increase the required boron concentration during refueling. But the safety implication of this is to improve safety for a boration/deboration accident.

g. Fuel Rod Power Census

Due to the larger difference in reactivity between the burnt and fresh fuel, a larger fraction of the fuel rods operate at a higher power rating. The effect is expected to be comparable to that of a transition to an 18-month cycle, which has been successfully achieved for many plants by Westinghouse.

h. Excore Detector Response

In the extended burnup core the flux levels at the core periphery are lower and the core more opaque. This reduces the flux at the excore detectors. The major reduction comes due to the transition to an 18-month cycle, with a small reduction due to extended burnup.

The major effect of extended burnup is on the power sharing between fresh and burnt assemblies. Design for power peak tailoring requires a little more effort for extended burnup cycles, particularly with neutron efficient loading patterns and improved burnable absorbers having low residual penalty. The effect is qualitatively similar to the transition

to 18-month cycle with these improved features. Westinghouse programs have demonstrated that the loading patterns required for these transitions can be obtained.

Current results indicate that changes in the nuclear parameters which are used as input to safety evaluations (as described in WCAP-9272[21]) are all expected to be within the current range of values used for safety evaluation input. All safety related nuclear parameters are examined on a cycle to cycle and design specific basis according to the methodology presented in WCAP-9272. The impact of changes in nuclear parameters on the safety evaluation is discussed in Section 2.5.

2.5 SAFETY EVALUATION METHODOLOGY

2.5.1 Non-LOCA Transients

This section discusses the impact of extended burnup fuel on methodology and computer codes used for non-LOCA safety evaluation. The reference analysis approach as described in WCAP-9272[21] is fundamental to the reload safety evaluation process. This process determines if a core configuration is bounded by existing safety analyses in order to confirm that applicable safety criteria are satisfied. The methodology, therefore, systematically identifies parameter changes on a cycle-by-cycle basis which may invalidate existing safety analyses and identifies postulated accidents that need to be re-evaluated. Because cycle specific parameters are used, the methodology is applicable to extended burnup cores.

The re-evaluation may be of two types. If the parameter is only slightly out of bounds, or the transient is relatively insensitive to that parameter, a simple quantitative evaluation may be made which conservatively evaluates the magnitude of the effect and explains why the actual analysis of the event does not have to be repeated. Alternatively, should the deviation be large and/or expected to have a

more significant or not easily quantifiable effect on the accident, a re-analysis of the accident is performed. If the accident is reanalyzed, the analysis methods follow standard procedures and will typically employ analytical methods which have been used in previous submittals to the NRC. These methods are those which have been presented in the FSAR, subsequent submittals to the NRC for a specific plant, Reference SARs such as RESAR-3S[22], or reports submitted for NRC approval. No changes in methods, analytical procedures, or computer codes are necessary for extended burnup fuel. The reanalyzed accident must continue to meet the appropriate safety limit for that event in order to be considered to have acceptable results.

The nuclear parameters that are most sensitive to extended burnup and the expected direction of change (with respect to lower burnup fuel) are discussed in Section 2.4. Table 2.1 lists those Non-LOCA transients that are most sensitive to these nuclear parameters. Also listed in Table 2.1 are the key nuclear parameters for each of the transients that are adversely impacted by extended burnup.

The Westinghouse acceptance criterion (as presented in the FSAR, subsequent submittals for a specific plant, Reference SARs such as RESAR-3S[22], or reports submitted for NRC approval) for eleven of the twelve transients listed in Table 2.1 are independent of burnup. For the rod ejection event, the Westinghouse criteria, as discussed in Reference 31, are as follows:

A. Average fuel pellet enthalpy at the hot spot below 225 cal/gm for non-irradiated fuel and 200 cal/gm for irradiated fuel .

B. Average clad temperature at the hot spot below $2700^{\circ}F$ the temperature above which clad embrittlement may be expected.

C. Peak reactor coolant pressure less than that which would cause stresses to exceed the Faulted Condition stress limits.

D. Fuel melting limited to less than the innermost 10 percent of the fuel pellet at the hot spot, even if the average fuel pellet enthalpy at the hot spot is below the limits of criterion A.

Criteria B, C, and D are not burnup dependent. Criterion A may be affected by burnup, and this is currently being investigated by the NRC. The observed burnup dependence of this parameter has occurred within the range of current fuel burnup and therefore this is not an extended burnup issue. The current Westinghouse criterion an average fuel pellet enthalpy is less than the criterion provided in Regulatory Guide 1.77. Pending new requirements in this area, Westinghouse will continue to apply its more conservative criterion.

All of the computer codes used in non-LOCA safety evaluations (LOFTRAN[32], FACTRAN[33], TWINKLE[34], and THINC[35]) are valid for extended burnup operation to lead rod average burnup of [ ]$^{+}$ MWD/MTU. (a,c)

2.5.2 LOCA Evaluation

Sensitivity studies performed by Westinghouse in the past[5,23] have shown that the calculated LOCA peak clad temperature decreases with increasing burnup from the time of highest pellet temperature, which occurs near the beginning of fuel life. For these sensitivity studies a series of analyses were performed which used fuel parameters and operating conditions characteristic of various burnups throughout life. The analysis included the effect of flow blockage, as calculated by the burst and blockage models approved by the NRC for use in Appendix K analyses at that time. The results show that the maximum peak clad temperature during a LOCA occurred using fuel parameters and initial conditions consistent with the time in life which exhibits the highest pellet average temperatures, near the beginning of life. From this stage of fuel burnup, the calculated peak clad temperatures were shown

to decrease throughout life. More recent analyses using the 1981 version of the Westinghouse ECCS Evaluation model (Ref. 24) support this conclusion.

The studies cited here utilized the Appendix K LOCA model and show that current design criteria should be unaffected by extended burnup. In addition, the sensitivity study performed in the past (Ref. 5) showed a relatively large margin to Appendix K LOCA limits with calculated peak clad temperatures during LOCA dropping rapidly throughout fuel life. Thus, minor variations among different fuels or plant designs would not affect the conclusions.

The sensitivity studies performed for the LOCA analysis[5,23] utilized the approved Westinghouse ECCS Evaluation model[25,26,27] applicable at the time. Recent upgrading of the ECCS model[24] should yield similar results since the basic code remained the same. Work performed recently using the 1981 version of the ECCS model[24] for moderately high burnup (20000 MTD/MTU) supports previous sensitivity studies. The ECCS model has sufficient input flexibility to explicitly model burnup dependent parameters. The currently approved model presumes the most limiting time in fuel life as near beginning of life at the time of peak fuel stored energy. Thus, continued use of the approved Westinghouse ECCS model assures that the most limiting fuel conditions have been used.

TABLE 2.1

Non-LOCA Transients Sensitive to Burnup Dependent Nuclear Parameters and the Impacting Parameters

+(a,c)

Transient	Impacting Parameter
Uncontrolled RCCA Bank Withdrawal at Power	
RCCA Misalignment	
Loss of Flow	
Loss of External Load and/or Turbine Trip	
Loss of Normal Feedwater/Loss of Offsite Power	
Excessive Heat Removal Due to Feedwater System Malfunction/Excessive Load Increase	
Accidental Depressurization of the Reactor Coolant System	
Single RCCA Withdrawal at Power	
Accidental Depressurization of the Secondary System/Steamline Rupture	
Main Feedline Rupture	
Single Reactor Coolant Pump Locked Rotor	
Rupture of an RCCA Drive Mechanism Housing (RCCA Ejection)	

*Turbine Runback Plants

2.5.3 Radiological Safety Evaluation

2.5.3.1 Core Inventory

Analysis of the core inventory of fission products has been performed using the ORIGEN[30] code and the ENDF/B-IV data library at region average burnups of 33000 MWD/MTU and at [ ]+ MWD/MTU (which is consistent with extended burnup operation. The results of the analysis, expressed as percentage changes in the core inventory of major isotopes of iodine, krypton, and xenon, are given in Table 2.2. The results are expressed in this manner since the percentage changes are applicable to a variety of core designs and are not sensitive to fuel enrichment nor to size of the core. (a,c)

The analysis shows that the increased burnup changes the core inventory of short-lived nuclides by between +5% and -13%. The effect of increased burnup on the long-lived krypton-85 is a near linear increase in core inventory. Krypton-85 is not a significant contributor to the radiological impact for postulated accidents or normal operation, and the remaining nuclides available for atmospheric release have changed by a small percentage. Therefore, the results presented in safety analysis reports for offsite dose consequences would not be changed significantly as a result of the extended burnup.

2.5.3.2 Gap Release Fractions

Using the ANS 5.4 Standard Fission Product Release Model, an analysis was performed to determine the effect of extended burnup on gap release fractions for the radioactive isotopes of Xenon, Krypton, and Iodine. The ANS 5.4 model provides an analytical method for calculating the release of volatile fission products from the fuel pellets during normal operation. In the analysis, the column of fuel pellets was divided into 7 axial and 10 radial nodes and the irradiation period was divided into a series of burnup (time) increments such that no burnup increment exceeded 2000 MWD/MTU. The temperature and specific power distribution

were assumed to remain fixed throughout each burnup increment. Data for temperature and specific power as a function of burnup was based on a Zion Station high burnup fuel assembly (discharged at about 55000 MWD/MTU) since this fuel has representative temperature and power histories.

The calculated peak gap fractions are given in Table 2.3 where they are compared with the Regulatory Guide 1.25 gap fractions. The results of this analysis show that, for isotopes with half-lives of less than a year, the gap fraction is more sensitive to fuel temperature than to burnup. Thus, as extended burnup is achieved (tending to increase the gap fraction), the lower fuel temperatures associated with the higher burnup level tend to decrease the gap fraction. For these isotopes having half-lives of less than one year, a peak release fraction is calculated to occur at the end of the second fuel cycle with a rod average burnup of [ ]+ MWD/MTU. For extended burnups, the gap release fraction declines from the peak value. (a,c)

For isotopes having half-lives greater than a year, e.g., krypton-85, the gap release fraction does not reach a peak but continues to increase with increasing burnup.

As can be seen on Table 2.3, the gap fractions for radioactive nuclides given in Regulatory Guide 1.25 are conservative for burnups up to [ ]+ MWD/MTU. (a,c)

The total fission gas release fraction, discussed in Section 3.2.2, includes both radioactive nuclides and stable nuclides (e.g. helium) and does not reach a peak but increases with increasing burnup.

2.5.3.3 Iodine Spiking

The prevailing theory on the thermal and hydraulic mechanisms producing the iodine spiking phenomena is the result of independent investigations in both the United States and abroad. The theory describes the probable

source of the spike inventory as cesium iodine salts which are deposited in the inner surfaces of the fuel rod cladding and to a lesser degree on the outer surface of the fuel pellets. A fuel rod with a cladding defect will admit reactor coolant liquid to contact the inner surfaces of the fuel rod only when the local power is below approximately 2 kw/ft. When reactor coolant enters the fuel rod, it will dissolve the cesium iodine salts deposited there. The dissolved cesium and iodine is then free to be transported to the reactor coolant system where it is seen as an iodine spike. A similar hydraulic mechanism occurs during reactor coolant depressurization wherein the trapped gases within the fuel rod above and/or below the defect location will be at a higher pressure than the coolant as the reactor coolant system is depressurized. This creates a driving head to expel the iodine laden water from the fuel rod thus producing another iodine spike. These spikes have become known as power spikes and pressure spikes.

Since the source of iodine is the fuel rod cladding gap activity, any spike activity resulting from fuel rods with extended burnup (i.e., greater than 33000 MWD/MTU region average burnup) would be less than that from defected rods at lower burnups due to the decreasing gap release fraction with increased burnup (Section 2.5.3.2) and the slight change in iodine inventory due to increased burnup (Section 2.5.3.1).

The only effect of higher burnups on iodine spiking would result from the lower linear power of extended burnup fuel rods. If a defected fuel rod were at a high burnup level, the linear power of the fuel rod would be closer to 2 kw/ft. compared to lower burnup fuel rods. Thus, a smaller decrease in reactor power would produce an iodine spike of the power spike variety. This implies two observations:

1) Upon reactor shutdown, the spike contribution from extended burnup fuel rods would be produced earlier in the shutdown transient than that contribution from defected rods at lower burnups, and

2) Iodine spikes would be produced with smaller power variations (e.g., smaller load follow swings) if defected rods were at extended burnup as compared to defected rods at lower burnups.

However, since the use of the iodine spiking phenomena in Safety Analysis Reports and Safety Evaluation Reports is in the area of radiological consequences of accidents, this effect would not affect the results of the analyses. In accident scenarios, a reactor trip occurs thereby resulting in all fuel rods dropping below 2 kw/ft at roughly the same time. In this case, extended burnup rods would not produce an early spike component.

In conclusion, the conservatism in the NRC spiking model is only added to in dealing with iodine spiking considering fuel rod defects in extended burnup fuel rods.

2.5.3.4 Fuel Handling Accident

The dropping of a fuel assembly considers resultant damage to all rods in the dropped fuel assembly plus 50 rods in an impacted fuel assembly. This results in the release of the gap activity for the damaged fuel rods. As indicated in Section 2.5.3.1, the core inventory for most nuclides of concern does not change significantly with burnup. Thus, even using the highly conservative gap release fractions given in Regulatory Guide 1.25, the radiological consequences of the fuel handling accident would not change significantly. However, an analysis has been performed for a fuel handling accident involving fuel assemblies from a discharge region having an average burnup of [ ]+ MWD/MTU, (a,c) which is consistent with extended burnup operation. The resulting releases from the fuel are indicated in Table 2.4 where they are compared with releases from a fuel handling accident involving fuel assemblies from a discharge region having an average burnup of 33000 MWD/MTU.

As stated in Section 2.5.3.2, the gap release fractions for most of the significant nuclides reach a peak (much lower than the Regulatory Guide 1.25 values) and then decrease with continued burnup. Therefore, the fuel handling accident involving fuel assemblies that have experienced extended burnup would, in actuality, have reduced radiological consequences in comparison with fuel at region average discharge burnup of 33000 MWD/MTU.

Another factor in determining the radiological consequences of a fuel handling accident is the capability of the pool of water in the refueling cavity or the spent fuel pit to remove iodine from the gas bubbles released from the damaged fuel rods.

A determination of pool decontamination factor (DF) for iodine is provided in WCAP-7828[28] which considers total volumes of fission gases formed plus helium fill gases in the fuel pins for assembly arrays of 14x14 and 15x15 at burnups up to 42000 MWD/MTU for a lead assembly. The maximum total gas for an assembly at this burnup was calculated at [ ]+ standard cubic feet. (a,c)

An analysis has been performed to extend the information provided in WCAP-7828 to determine the pool DF for extended burnup fuel. The analysis is conservatively based on a fue' assembly consisting of fuel rods which all have a burnup exceeding [ ]+ MWD/MTU. With this (a,c)
burnup, the contained gas is calculated to be [ ]+ standard cubic (a,c)
feet in a 15x15 fuel array. [
]+ Since the experiments (a,c)
conducted in WCAP-7828 were simulations of 15x15 fuel assemblies, the results were extrapolated to consider the new volume of [ ]+ standard cubic feet. (a,c) This extrapolation indicates a pool DF of approximately [ ]+. (a,c) It is concluded that the DF value of 100, which is assumed for a fuel handling accident by guidance of Regulatory Guide 1.25, is not only valid, but conservative, for the extended burnup fuel.

TABLE 2.2

Change in Reactor Core Inventories of Iodines and Noble Gases for Change in Fuel Burnup from 33000 to [ ]+ MWD/MTU (a,c)

(Region Average Discharge)

Nuclide	Percentage Change
Kr-83m	- 7%
Kr-85	+34%
Kr-85m	-10%
Kr-87	-12%
Kr-88	-11%
Kr-89	-13%
Xe-131m	+ 3%
Xe-133	+ 1%
Xe-133m	No Change
Xe-135	- 8%
Xe-135m	+ 5%
Xe-138	- 3%
I-131	+ 3%
I-132	+ 3%
I-133	No Change
I-134	- 1%
I-135	- 1%

2209F:6

TABLE 2.3

Fuel Rod Gap Release Fractions for
Fuel Rod Average Burnups to [ ]+ MWD/MTU (a,c)

Isotope	Regulatory Guide 1.25	Peak Release* Fraction
I-131	0.10	[ ]+ (a,c)
I-132	0.10	
I-133	0.10	
I-134	0.10	
I-135	0.10	
Xe-131m	0.10	
Xe-133m	0.10	
Xe-133	0.10	
Xe-135m	0.10	
Xe-135	0.10	
Xe-138	0.10	
Kr-83m	0.10	
Kr-85m	0.10	
Kr-85	0.30	
Kr-87	0.10	
Kr-88	0.10	
Kr-89	0.10	

*The gap release fractions for all the listed nuclides except Kr-85 reach peak values at the end of the second fuel cycle with a rod average burnup of [ ]+ MWD/MTU. For Kr-85 the release fraction does not peak but increases with fuel burnup. The release fraction of [ ]+ is associated with a rod average burnup of [ ]+ MWD/MTU which is the upper value used in the analysis. For nuclides that have a peak gap release fraction, the release fraction decreases from the peak value as burnup increases. (a,c) (a,c

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TABLE 2.4

Noble Gas and Iodine Inventories Releases as the Result of a Fuel Handling Accident

Radionuclides	Region Average Discharge Burnup: 33000 MWD/MTU*	Region Average Discharge Burnup: [ ]+ MWD/MTU*	(a,c)
Noble Gases	Curies Released		
		[ ]+	(a,c)
Kr-85	2.8×10^{3}		
Xe-131m	6.2×10^{2}		
Xe-133m	1.2×10^{4}		
Xe-133	1.5×10^{5}		
Xe-135m	7.8×10^{-1}		
Xe-135	2.6×10^{2}		
Iodines			
I-130	7.8	[ ]+	(a,c)
I-131	7.1×10^{4}		
I-132	6.0×10^{4}		
I-133	7.5×10^{3}		
I-135	5.1		
	Grams Released		
I-127	1.1×10^{1}	[ ]+	(a,c)
I-129	4.6×10^{1}		

*The values are based on the following assumptions:

a) Gap inventory of 314 fuel rods in discharge region

b) Radial Peaking Factor of 1.65

c) Accident occurs 100 hrs. after shutdown

d) 4.1 w/o U-235

3.0 REVIEW OF CURRENT DATA BASE, MODELS, AND METHODOLOGY

3.1 INTRODUCTION

This section provides evidence that the Westinghouse data base, performance models, and methodology are sufficient to justify design to a lead rod average burnup of [ ]+ MWD/MTU. Due to extensive Westinghouse extended burnup experience in the past, Westinghouse has had available a significant extended burnup data base for several years. Because of the availability of these data, performance models developed over the past several years have already included the observed extended burnup effects. In addition to the measured data provided in this section, comparisons between measured data and predicted results are provided, when appropriate, to confirm the applicability of the models. Current design methods are also discussed in terms of their applicability to extended burnup. (a,c)

The Westinghouse fuel experience in reactors at burnups in excess of 36000 MWD/MTU is summarized in Table 3.1. There are several plants where region sized batches (~1/3 of a core) have achieved average discharge burnups of 36000 to 37000 MWD/MTU. In addition, several smaller batches of 1-8 assemblies have achieved burnups in excess of 40000 MWD/MTU and 4 assemblies have reached lead assembly burnups ~ 55000 MWD/MTU. These lead assemblies are part of a joint EPRI/Westinghouse extended burnup program, and have recently been discharged after 5 cycles of operation in the Zion Unit 1 reactor. The results of examinations of these assemblies through 4 cycles of operation are currently available, and additional data will be available following the 5-cycle assembly examination scheduled for August 1982.

The coolant activity levels at the end of cycle for each of the extended burnup batches, shown in Table 3.1, lie in the range typical of commercial reactors. No correlation has been observed between coolant activity and extended burnup. The Zion extended burnup assemblies were

sipped following their third and fourth cycle of irradiation and found to be leak tight. This demonstrates shows that good fuel integrity is achievable through these burnup levels.

The above experience was obtained at normal operating conditions. Experience under lead rod power conditions is required to provide confidence that no performance limiting phenoma that would be seen under more severe operating conditions are being approached, but not observed at normal operating conditions. Data from extended burnup, high power irradiations performed in the Jose Cabrera and BR-3 reactors address this concern. The burnup levels achieved in these programs are summarized in Table 3.2. The Jose Cabrera program consisted of irradiation of "spiked" rods (~6 wt%) to rod average burnups up to 58000 MWD/MTU [44,45]. One of the significant observations concerning extended burnup from this program was that of increased fission gas release in the extended burnup rods, high power rods.

The BR-3 program consists of irradiation and examination of enriched test rods (~8 wt%) in the BR-3 reactor at lead rod power levels. The fuel rods achieved maximum rod average burnups as high as 61500 MWD/MTU with maximum pellet burnups of 74000 MWD/MTU. Sipping confirmed that the rods were intact upon discharge from the reactor.

The relationship of the power-burnup histories of the normal operation and lead rod data is shown in Figure 3.1. The lower curve in the figure bounds all of the power-burnup histories of Westinghouse power reactor data. Taken as a whole, this data tends to bound normal operating conditions. The curves for the Jose Cabrera and BR-3 programs represent an envelope of the maximum power-burnup histories achieved in these programs (the envelope is not necessarily a single rod, but is typically a combination of several rod histories). The maximum power conditions used for design tend to fall between the normal operation envelope of Figure 3.1, and the Jose Cabrera and BR-3 envelopes. The commercial reactor data addresses concerns dependent on burnup and residence time. The lead rod irradiations provide data on performance reliability at operating conditions exceeding expected levels as well as providing information for modelling and licensing.

Table 3.1

BURNUP EXPERIENCE IN EXCESS OF 36,000 MWD/MTU
IN WESTINGHOUSE COMMERCIALLY OPERATING REACTORS

PLANT	CYCLE EXPOSURE	NO. OF FA	BATCH AVG. BU	MAX. ROD BU	I-131 μCI/GM
Point Beach 2	3	37	36000	42100	0.004
	4	1	43600	45,800	0.001
	4	8	40100	44,200	
	4	4	37300	42,100	
	4	4	36000	38,400	
Point Beach 1	3	2	36400	40000	0.04
Prairie Island 1	3	1	36500	40900	0.0001
	3	1	37700	41000	.015
Prairie Island 2	4	1	44500	49000	0.0008
	3	39	35900	42000	0.001
Zion 1	3	60	36000	42000	0.03
	3	5	36000	37500	0.03
	4	4	46510	49300	0.0015
	5	4	~ 55000	~ 60000	0.011
Zion 2	3	64	36900	42700	0.01

Table 3.2

EXTENDED BURNUP EXPERIENCE
LEAD ROD CONDITIONS

	Number of Rods	Burnup Range (MWD/MTU)
	18	47000 - 51000
Jose Cabrera	10	52000 - 56000
	2	56000 - 58000
High Burnup BR-3 Rods	9	49000 - 61500
		($\sim$74000 peak pellet)

3.2 FUEL ROD PERFORMANCE

3.2.1 Clad Oxidation and Hydriding

3.2.1.1 Clad Oxidation

Significant corrosion data under a variety of operating conditions have been obtained in the Saxton, Zorita, BR-3 and Trojan programs. The Saxton, Zorita, and BR-3 programs produced data under high power conditions to rod average burnups in the range of 30000 to 61500 MWD/MTU. Irradiations under more representative power conditions were performed in the Trojan and Zorita reactors. The Trojan reactor is one of the highest coolant temperature plants operating today and is, therefore, representative of conditions in thermally efficient plants now coming on line. The data from several programs from commercial reactors are summarized in Figure 3.2. As shown in the figure, crud deposition has a significant effect on corrosion behavior. For those cases where crud deposition was nominal ($\sim$ 0.3 mils), the corrosion is observed to be generally consistent with that expected from thermal corrosion. The range of the oxide thickness shown in the figure represents the range of the maximum local corrosion thickness observed as a function of burnup. The maximum corrosion thickness observed under these conditions is approximately [ ]$^{+}$. The data base includes a (a,c)
maximum burnup of $\sim$60000 MWD/MTU. This level of waterside corrosion poses no limitation for extended burnup operation. It is significant that these data are obtained under both lead rod and typical power conditions. Larger oxide thicknesses, observed in situations where higher than normal crud deposition occured, are also shown in Figure 3.2. This observation was particularly true when excessive crud deposition occurred in combination with high power levels. In spite of the increased corrosion thickness for the crudded rods shown in Figure 3.2, there were no adverse performance effects in these rods. The above experience shows that under nominal crud conditions, cladding oxidation is low and generally consistent with that expected from thermal predictions. Observations of increased oxidation have been related to abnormal coolant

chemistry/crud conditions and high power operation; extended burnup or residence time per se did not result in accelerated waterside corrosion. Therefore under normal operating conditions, cladding waterside corrosion will not limit operation below a lead rod burnup limi . of [ ]+ MWD/MTU. (a,c)

3.2.1.2 Clad Hydriding

The hydrogen uptake of Zircaloy-4 is typically low and generally lies between 5 and 20% of theoretical during oxidation [29]. Data has been accumulated from several programs from commercial reactors. Figure 3.3 shows the trend of the range of hydrogen uptake as a function of rod average burnup. The maximum hydrogen uptake observed under these conditions is approximately [ ]+ corresponding to a burnup of 61500 MWD/MTU. This level of hydrogen uptake poses no limitation for extended burnup operation of the fuel cladding. (a,c)

3.2.2 Fission Gas Release

The Westinghouse fission gas release data base is composed of fuel rod test data obtained over a broad range of burnup and power levels. Test rod burnup levels range from several thousand MWD/MTU to greater than 57000 MWD/MTU. Average linear power density ranges from 3.4 to 9.2 KW/FT, and peak linear power levels range from 7.3 to 18.3 KW/FT. Both pressurized and unpressurized rods are included in the data base, and test rod fuel densities span the range of current design fuel.

Fission gas release data obtained from fuel rods taken from the Saxton and Jose Cabrera reactors is summarized in detail in Reference 4, and a subset of the data base composed of data from the Jose Cabrera core is provided in Figure 3.4. Fission gas release data in Figure 3.4 obtained from the Jose Cabrera program for both high and nominal power fuel has shown a distinct power dependency (fuel temperature) on fission gas release. In addition, the observed gas release appears to vary with burnup. The high power rods at approximately 30000 MWD/MTU and in the 35000-40000 MWD/MTU range in Figure 3.4 were operated at high powers for one and two cycles respectively. The nominal power rods in the

35000-40000 MWD/MTU range were operated for three reactor cycles at power levels more typical of nominal reactor operation. The extended burnup rods in the 49000-58000 MWD/MTU range were irradiated for three cycles at high power levels.

As can be seen in Figure 3.4, the high power, 35000-40000 MWD/MTU rods showed markedly higher release than the lower power rods with comparable burnups. This indicates a significant dependency on fuel temperatures with little release being observed in the normal operating range and an increasing amount released as fuel temperatures are raised. In addition, the observed releases of the high power, extended burnup, three cycle rods was significantly higher than the lower burnup, two cycle high power rods even though the power levels in the second and third cycles of irradiation were comparable to the two cycle rods. Although all of the rods in the 49000-58000 MWD/MTU range were irradiated at higher than normal power levels, dependency on power history was also observed within this group. The two rods at approximately 50000 MWD/MTU in Figure 3.4 did not operate at a constant high power level for three cycles as did the other three cycle high power rods but instead had a gradually decreasing power level with time. The lower releases of these rods shows that the high gas releases observed in the extended burnup rods was very sensitive to the operating power level and, therefore, fuel temperature.

The Westinghouse fission gas release model, discussed in detail in Reference 4, is based on data taken from high power rods at burnups of up to ~58000 MWD/MTU. The extended burnup effects on fission gas release have already been included in the NRC approved Westinghouse fuel performance code [4]. A comparison of measured versus predicted release fraction as a function of burnup is provided in Figure 3.5. This figure illustrates the good agreement between the model predictions and measurements particularly for high power rods at extended burnup. A comparison of measured versus predicted release fraction for normal power rods which are more representative of typical PWR rods at extended burnup is provided in Figure 3.6. This figure indicates that the

current Westinghouse fission gas release model is conservative, predicting generally higher release fractions than measured for normal power rods. It is concluded from these comparisons that the current Westinghouse fission gas release model conservatively predicts release fractions in normal power fuel rods irradiated to lead rod average burnup of [ ]+ MWD/MTU. (a,c)

Additional extended burnup fission gas release data is anticipated from Zion extended burnup test rods operated through 5 cycles of irradiation to burnup of approximately [ ]+ MWD/MTU. These rods represent typical extended burnup rods operated at normal PWR rod power densities. Additional, fission gas release data for extended burnup, high power density rods operated in the BR-3 reactor to burnup in excess of 61000 MWD/MTU, and from the Jose Cabrera reactor to burnup of ~58000 MWD/MTU will also be available soon. These sources of additional data (discussed in Section 5.0) are expected to provide additional confirmation of the existing data base. (a,c)

Data presented in this section addresses the total fission gas release including both stable and radioactive isotopes. The release function of the radioactive nuclides as a function of burnup is discussed in Section 2.5.3.2.

3.2.3 Fuel Rod Profilometry

Cladding profilometry data provides a measure of outward cladding deformation and has been measured in extended burnup programs. The Zorita program provided profilometry data for pressurized and non-pressurized fuel irradiated at high and nominal power conditions to rod average burnups of 58,000 MWD/MTU. Profilometry data was obtained after each of the first four cycles of irradiation at Zion with maximum rod average burnup of ~49,000 MWD/MTU. A significant difference was observed in the profilometry behavior between the high and nominal power conditions.

In the Zorita program, the high power rods showed uniform pellet cladding contact and cladding outward deflection as shown in Figure 3.7. In both the pressurized and non-pressurized rods, this "creepout" at EOC-3

represented approximately 50% of the cladding diameter change at EOC-1 where creepdown was maximum. The creepout was attributed to fuel swelling since calculations and rod puncture data showed the fuel rod pressures were less than system pressure throughout operation. For those rods, ridging was observed on all rods indicating pellet-clad contact.

The behavior of the nominal power Zorita rods and the Zion rods are summarized in Figure 3.8. The nominal power prepressurized Zion and Zorita rods displayed similar trends of continued creepdown with increasing burnups although slight diameter increases were noted at some isolated points. Light ridging was observed on all rods and localized pellet-cladding contact was evident. The difference in creepdown for the pressurized rods in Figure 3.8 is attributed to differences in initial prepressure levels and core system pressure. The observation of continued creepdown indicated that no general fuel swelling occurred and that the rod internal pressure remained below system pressure.

Although minor diameter increases were observed in the non-pressurized rods as shown in Figure 3.8, no major effect of fuel swelling was observed. The amount of diameter increase observed between Cycles 2 and 3 was very small (approximately 0.03%, within measurement uncertainty) and was clearly less than the diameter increases observed in the high power fuel (Figure 3.7) at the higher burnups.

Based on the above profilometry data, no significant fuel swelling effects have been observed for rods irradiated at typical power levels. Further, for the prepressurized rods typical of current design, continued creepdown was observed through four cycles of iradiation (49000 MWD/MTU rod average burnup). Some local clad "creepout" was observed, however, for the fuel operated at high power levels. This data indicates that for fuel operated at power levels typical of those projected for extended burnup application, no significant effects from fuel swelling are expected. However, higher power operation can lead to an increase in fuel swelling at such burnups. The combination of high power Zorita data and the Zion data will enable this effect to be adequately addressed in design.

3.2.4 Fuel Swelling and Densification

The Westinghouse fuel swelling and densification model is based on an extensive data base composed of both measured post-irradiation fuel density an fuel stack heigh data. Data with burnups of greater than 53000 MWD/MTU are included in the data base and were used to develop the Westinghouse fuel swelling and densification models. The data and model are discussed in detail in Reference 4 and 36.

The fuel densification process is essentially complete at current fuel discharge burnups. For extended burnup operation the swelling model is of primary interest. The swelling rate used in the Westinghouse fuel performance code is based on both fuel density and fuel stack length measured data is discussed in the above references. The data base upon which the swelling model was developed included density data at extended burnup (48000 MWD/MTU). This data base is considered representative of extended burnup operation, no mechanism has been identified which would cause the swelling rate to change at extended burnup. A fuel swelling rate of [ ]$^{+}$ $\Delta V/V$ per fission per cm^3 was determined (a,c)
based on both the density data and fuel stack length data [4] and is used in Westinghouse fuel performance calculations. [

].$^{+}$ Comparison (a,c)
with data found in the open literature show that the swelling rate of
[ ]$^{+}$$\Delta V/V$ per fission per cm^3 is consistent with data (a,b,c)
for similar type fuel.[38, 39, 40, 41]

3.2.5 Clad Flattening

Fuel clad flattening is evaluated using the NRC approved Westinghouse clad flattening model.[8] In previous generation fuel, clad flattening occurred following the formation of gaps in the fuel stack due to fuel densification. Stress induced irradiation creep of the cladding

then resulted in collapse of the cladding into the unsupported gap region of the rod. For current generation stable fuel in prepressurized fuel rods, no instances of clad flattening have been observed, even in fuel irradiated through four cycles of operation in Zion Unit 1. This is consistent with predicted clad flattening time obtained using the Westinghouse clad flattening model. For typical current generation fuel, predicted clad flattening time is typically in excess of 45000 effective full power hours.

The conservatism of the clad flattening model was demonstrated in Reference 8 by comparing the observed clad flattening frequency in previous generation fuel with the predicted flattening frequency. In all cases, the Westinghouse model was shown to conservatively predict flattening with respect to the data. For current generation fuel, experience with fuel operated through 4 cycles of irradiation has resulted in no observations of clad flattening, thus confirming model predictions.

Further confirmation of the model is anticipated following the examination of 4 Zion Unit 1 assemblies in July, 1982 which have been irradiated through 5 cycles and have achieved lead rod burnup of ~[ ]$^{+}$ (a,c)
MWD/MTU. No clad flattening is anticipated.

3.2.6 Pellet Clad Interaction

Westinghouse and other available PCI data has been analyzed in Reference 9 to determine the burnup dependence of PCI for burnup up to ~30000 MWD/MTU. Analysis of the available data showed that the burnup dependence of PCI susceptibility tends to saturate after approximately 15000 MWD/MTU. The critical power change (ΔP) required to exceed the failure threshold was found to decrease strongly with burnup to 15000 MWD/MTU. For test rods in excess of 15000 MWD/MTU the critical power (ΔP) did not change significantly.

Additional data on PCI failure at burnup in excess of 30 GWD/MTU has been obtained through Westinghouse participation in the International SUPER-RAMP Project. The scope of the SUPER-RAMP Project is an experimental investigation of PCI performance under power ramp conditions for

commercial type LWR test fuel rods irradiated to extended burnup. Test rod burnups range from 29 to 45 GWD/MTU. The results of the program to date for PWR fuel rods are shown in Figure 3.9. These rods were preconditioned at [ ]+ KW/FT prior to ramping at rates ranging from [(b,c)
]+ W/cm/min. No evidence of increased PCI susceptibility is (b,c)
noted in the data in Figure 3.9. It is also observed that in many cases rods at extended burnup did not fail even when the failure threshold at 15000 MWD/MTU was exceeded. It is therefore concluded that the available PCI data does not indicate increased PCI susceptibility at extended burnup.

3.3 FUEL ASSEMBLY STRUCTURE PERFORMANCE

3.3.1 Fuel Rod Fretting

The adequacy of the current Westinghouse Inconel grid design has been verified through extensive operating experience which has resulted in no observed incidents of fuel rod fretting wear. This data base includes assemblies which have experienced four cycles of operation to an assembly burnup of approximately 47000 MWD/MTU in the Zion Unit 1 extended burnup program. Further examination of these same assemblies, burned for an additional cycle to an assembly burnup of approximately 55000 MWD/MTU (with a lead rod average burnup of approximately 60000 MWD/MTU) will provide additional confirmation that fretting wear is not a concern for lead rod average burnup of [ ]+ MWD/MTU in assemblies with (a,c)
Inconel grids.

The data base for assemblies with Zircaloy grids is composed of data from four 17 x 17 OFA demonstration assemblies which have completed two cycles of irradiation. Examination of removable rods after two cycles revealed [
]+. (b,c)

Prior to the insertion of the demonstration assemblies, a test program was conducted on an assembly with all Zircaloy grid cells presized to simulate operation with gaps between the cladding and grid springs. Results of this test program were extrapolated to show that the Zircaloy grid assembly design was adequate through [ ]+ of this gapped operation. [(a,c)

]+ of (a,c)

allowed operation with gaps based on the test program combine to demonstrate that fuel rod fretting wear in assemblies with Zircaloy grids will not be a limiting concern through at least [six annual]+ cycles of operation. This is sufficient to allow operation to lead rod average burnup of [ ]+ MWD/MTU. (a,c)

The Zircaloy gridded demonstration assemblies are now in their third cycle of operation, and a fourth cycle is currently planned. It is expected that future examinations of these assemblies will provide additional confirmation that fretting wear in assemblies with Ziracloy grids is not a concern for extended burnup.

3.3.2 Zircaloy Hydriding

Hydrogen pickup data for Zircaloy guide thimble tubes after one cycle of operation in the Point Beach Unit 1 reactor indicates a maximum hydrogen concentration of 83 ppm. A conservative extrapolation of this data for an assumed residence time consistent with operation to lead rod average burnup of [ ]+ results in a value of [ ]+ ppm, which is well within the design criteria. (a,c)

Planned hot cell programs for 3 and 5 cycle fuel assembly skeletons from the Zion Unit 1 reactor are expected to provide additional confirmation of Zircaloy hydrogen levels for extended burnup assemblies.

3.3.3 Fuel Rod Growth Gap

The reduction in the fuel rod-to-nozzle gap as a function of burnup (fluence) is primarily due to the differential Zircaloy growth of the cold worked fuel cladding and the annealed fuel assembly guide thimble tubes. Irradiation induced creep of the guide thimble tubes also affects fuel assembly growth and therefore growth gap to a lesser extent. Creep occurs as a result of the stresses in guide thimbles due to holddown spring forces acting to compress the guide thimbles and fuel rod expansion forces acting through the grid springs tending to lengthen the guide thimbles. The balance between these two effects changes with burnup as grid springs relax and holddown forces change. As a result, the creep contribution to fuel assembly growth can be either positive or negative, and can change with residence time.

Significant extended burnup fuel rod-to-nozzle gap data are available through on-site examinations of Westinghouse extended burnup fuel. In references 46 and 47, rod-to-nozzle growth gap data are provided for Zion Unit 1 assemblies through four cycles of operation to an assembly average burnup of 46500 MWD/MTU (lead rod average burnup of approximately 49000 MWD/MTU). The range of exposure for these data includes rods exposed to a fluence near 9.0×10^{21} nvt (E<1.0 Mev). Evaluation of the data shows that the rod-to-nozzle gap behavior is best fit as a [ ]$^{+}$ function of rod average fluence. (a,

The relationship of measured rod-to-nozzle gap as a function of fluence is used in design to establish the initial rod-to-nozzle gap. Based on the data, a probability function of rod average fluence has been developed which accounts for scatter in the data base. Since not all rods in an assembly are exposed to the same fluence, the gap closure relationship is statistically combined with the expected distribution of rod average fluence for an assembly with lead rod average burnup of [ ]$^{+}$ MWD/MTU to determine the percent gap closure for design to (a,c the target lead rod average burnup. The initial gap is designed to provide an acceptble gap distribution for assemblies which have a lead rod average burnup of [ ]$^{+}$ MWD/MTU. (a,c

Additional data from the Zion Unit 1 assemblies after five cycles of operation will be available in August 1982. These assemblies are at 55000 MWD/MTU assembly average burnup with a lead rod average burnup of approximately 60000 MWD/MTU. The data is expected to confirm the trends exhibited in the existing data base.

3.3.4 Fuel Rod Bow

Westinghouse has carried out rod bow evaluations in numerous PIE programs during refueling shutdowns. As a result, an extensive body of rod bow data has been accumulated on Westinghouse fuel including 14x14, 15x15 and 17x17 designs [17]. In many cases repeat measurements were made on the same fuel assemblies in the same channels in order to follow the progression of bow with increasing burnup.

The most comprehensive and accurate rod bow data have been obtained in the Zion 15x15 program, in the 17x17 demonstration assemblies in the Surry reactors, and the recent Trojan program where 17x17 fuel assemblies have completed up to 3 cycles with burnups up to 37,000 MWD/MTU. Figure 3.10 summarizes the Westinghouse rod bow performance. The 95th percentile worst span closure, a conservative bounding design parameter, is plotted against burnup. It is seen that in all cases the rod bow is well below the design curve. Of greatest significance for extended burnup, rod bow did not increase significantly at the higher burnup levels. There has been no observation in any of the Westinghouse studies of excessive increases in rod bow at extended burnups that would invalidate the general trend seen in Figure 3.10

Examination of extended burnup fuel assemblies at Zion during successive cycles of operation revealed that one rod was observed to have bowed to near contact (>90% closure) with its neighbor rods in one or two grid spans early in life and operated in that condition for at least two cycles. Close examination of the bowed rod and its neighbors revealed no adverse effects of prolonged operation in contact. There was no evidence of excessive local heating or corrosion at the closure location

or of secondary effects elsewhere in the fuel rods, and leak testing after 4 cycles of operation indicated the assembly to be sound. The above data shows that rod bow is remaining relatively stable and well below licensing criteria at extended burnup.

3.3.5 Guide Thimble Tube Wear

Westinghouse has conducted measurement programs to assess guide thimble tube wear in 14 x 14 and in 17 x 17 fuel assemblies. The first observation of guide thimble wear was in hot cell examinations at a Point Beach Unit 1 fuel assembly. Abnormal wear was noted in two out of seven guide tubes, with maximum wall thinning of 45% of the wall thickness. Later, on-site measurements were performed on 49 thimbles selected from 14 x 14 assemblies irradiated in Point Beach Unit 1 and 2 reactors. Eighteen guide tubes showed no wear, and of those with measured wear, the examinations revealed a maximum wear depth of 65% of the wall thickness. Maximum wall thinning was noted near the top eight inches of the guide tube, corresponding to the insertion length of the fully withdrawn control rod. D-Loop test data on a 17 x 17 design fuel assembly showed that maximum guide tube wear at the first bulge point was only 0.4 miles, and it was concluded that this amount of wear will not affect fuel assembly performance even if it were to occur in all guide thimbles. Confirmatory qualitative assessments of guide thimble tube wear in 17 x 17 assemblies irradiated in the Salem Unit 1 reactor were carried out on a total of six region 1 assemblies after two cycles of irradiation. A visual examination was performed using a Westinghouse designed borescope and camera system, and no evidence of through-wear holes or other excessive wear was noted.

In response to the NRC's assessment of the susceptibility and impact of guide thimble wear in Westinghouse plants, a comprehensive quantitative evaluation of the wear measurements on 14x14 Westinghouse fuel assemblies from Point Beach Units 1 and 2 was conducted, and an analytical guide thimble wear model was developed to predict the wear magnitude for 15x15, 16x16 and 17x17 fuel designs. Based on these analyses, it was

concluded that in the worst case, no perforation would occur for at least 225 weeks of continuous fuel assembly operation under a parked RCCA location (equivalent to 5 cycles of reactor operation) and that the integrity of the guide thimble tube would be maintained during normal operation, accident conditions, and nonoperational loading conditions throughout the normal life time of a fuel assembly. Details of the guide thimble wear model development and results have been presented to the NRC in References 18, 19, and 20.

The restriction of 225 week residence time under a Rod Cluster Control Assembly (RCCA) in the fully withdrawn position does not pose a limitation on the extended burnup design of Westinghouse fuel.

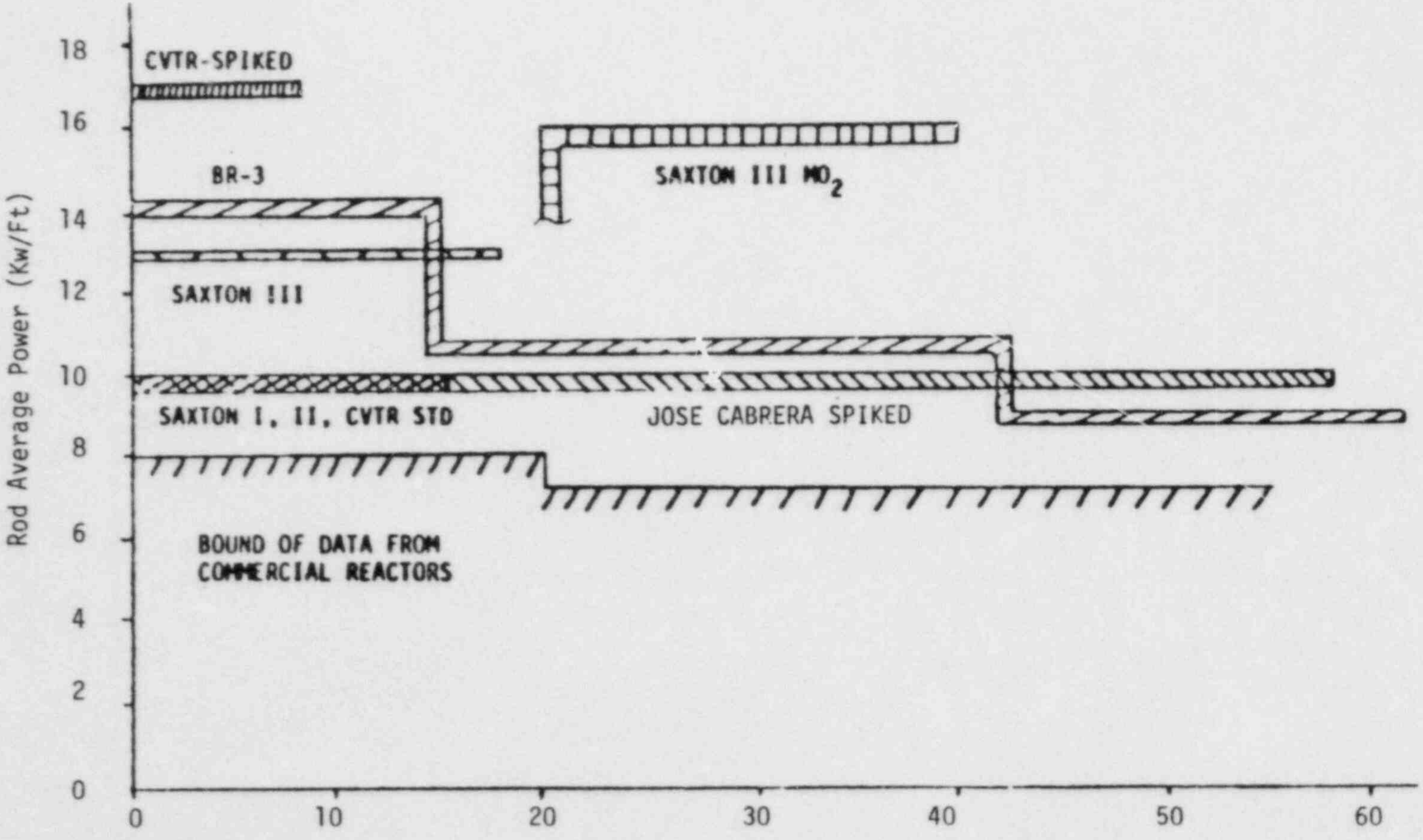


Figure 3.1
Summary of Westinghouse Experience

Figure 3.2

Cladding Oxide Thickness Data As A Function Of Burnup
(From Commercial Reactors)

+ (a,b,c)

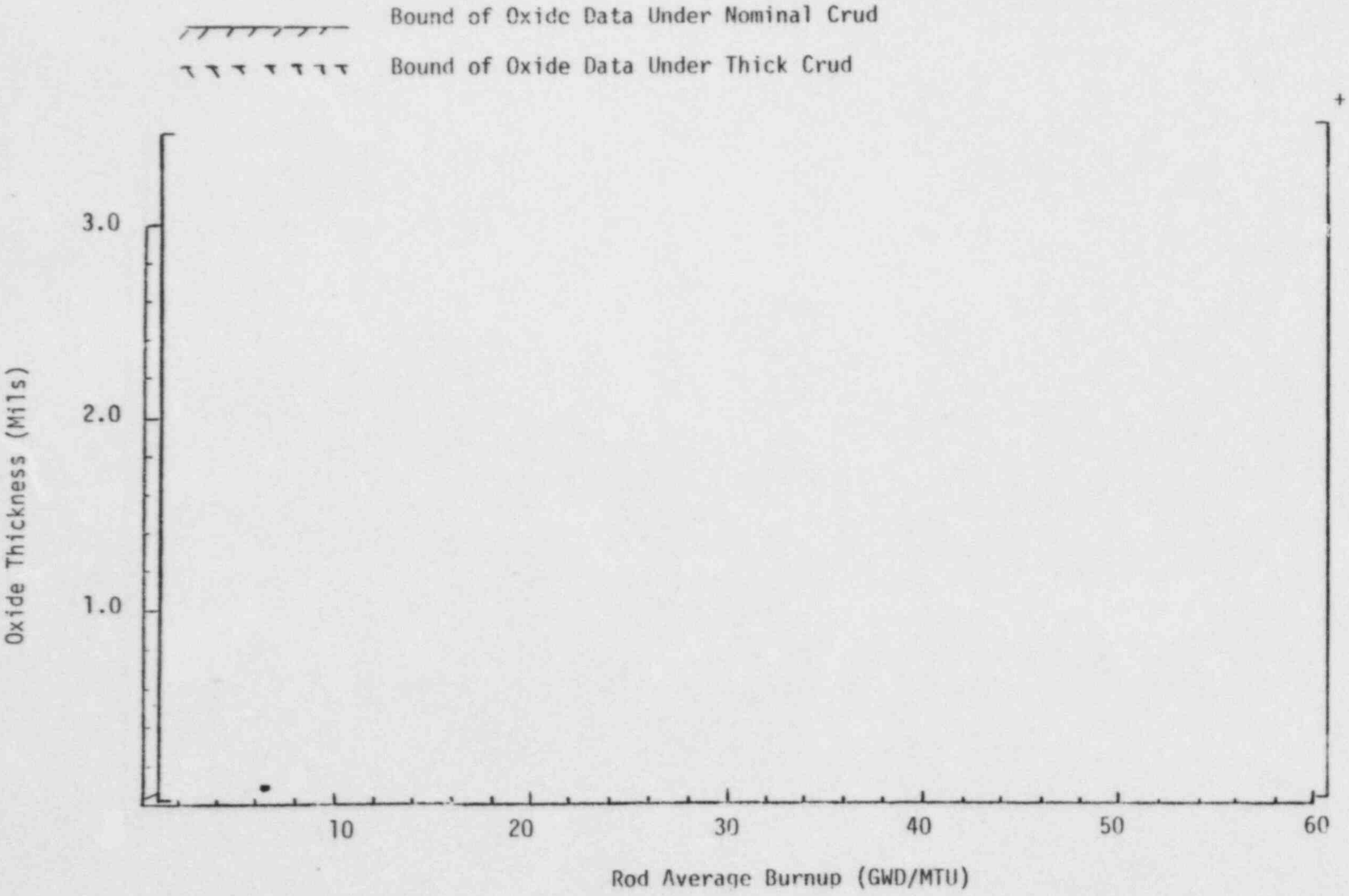
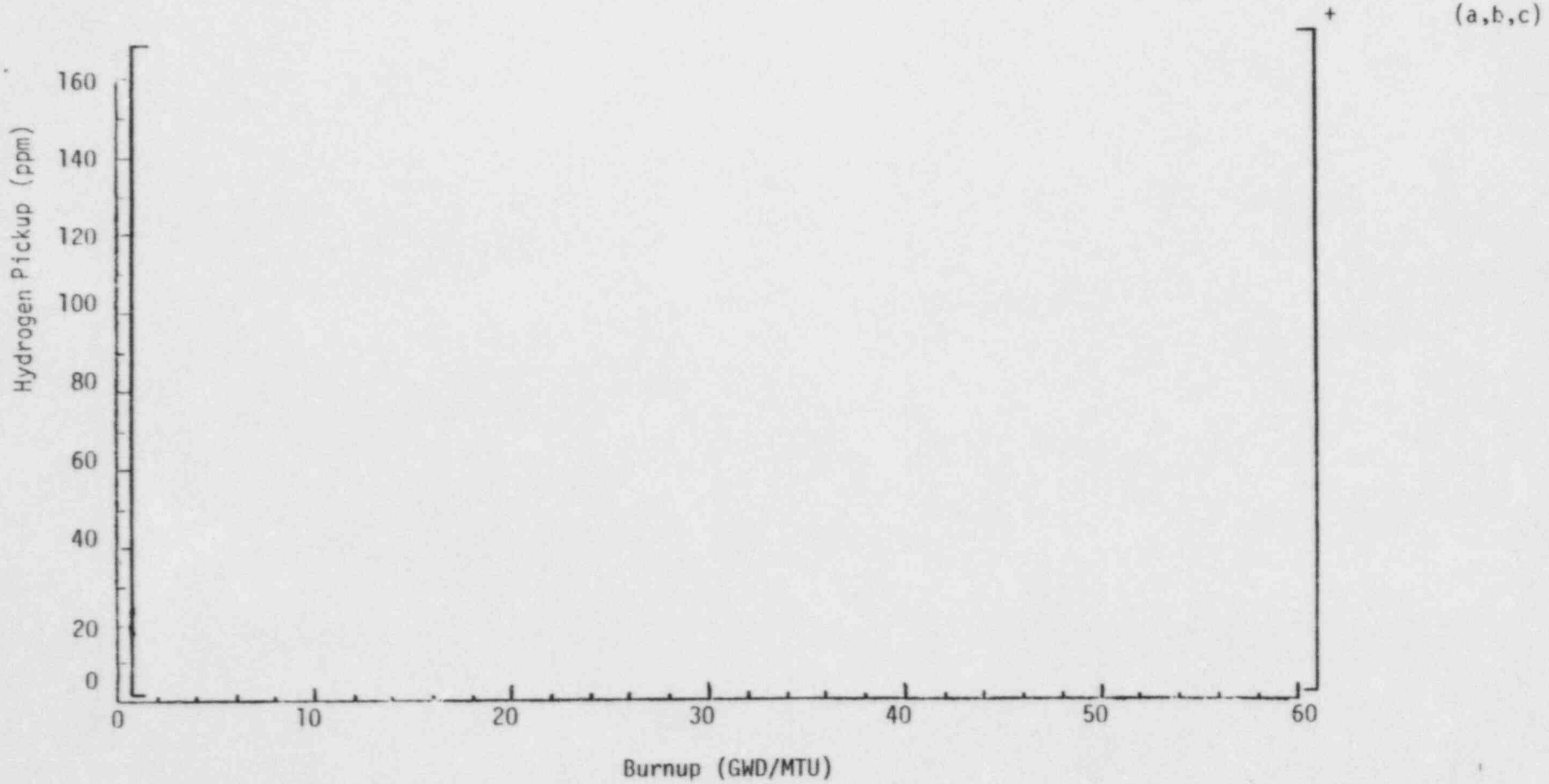


Figure 3.3

Cladding Hydrogen Pickup Data As A Function Of Burnup
(From Commercial Reactors)



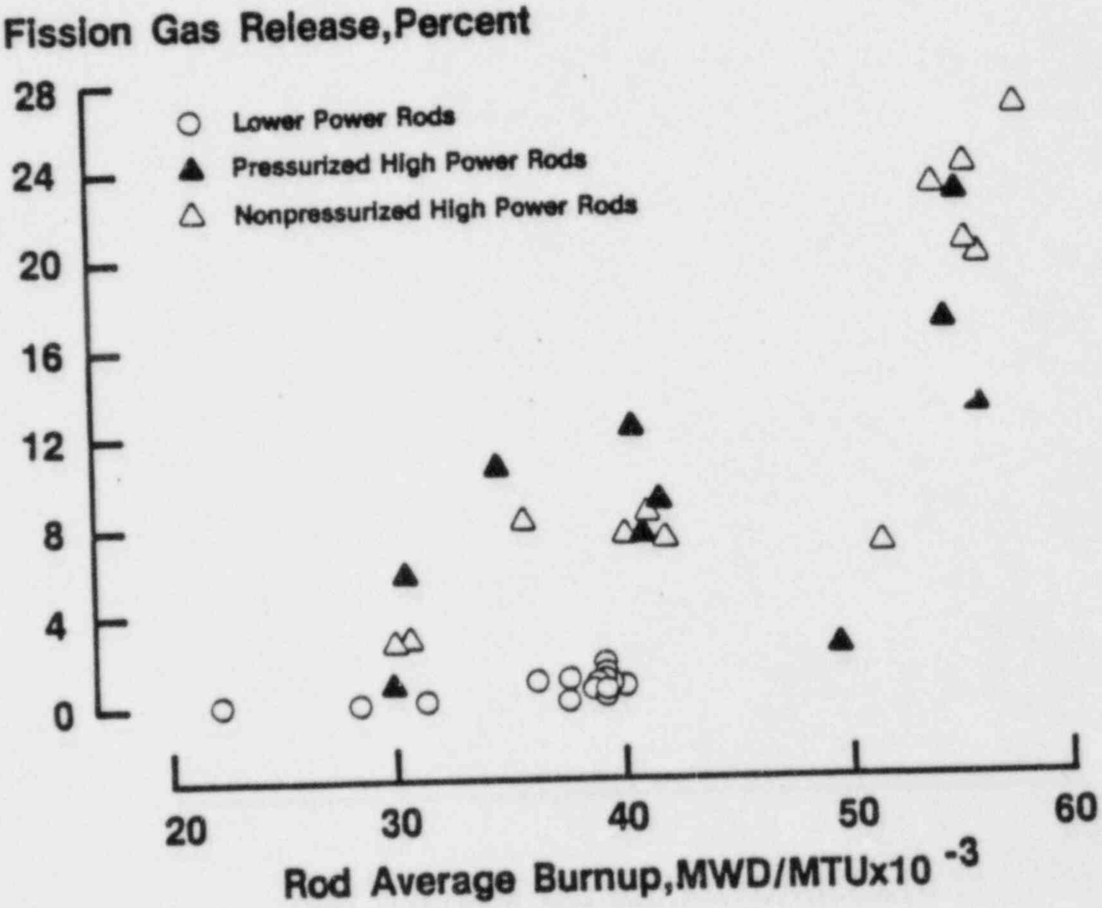


Figure 3.4
Jose Cabrera Fission Gas Release Data

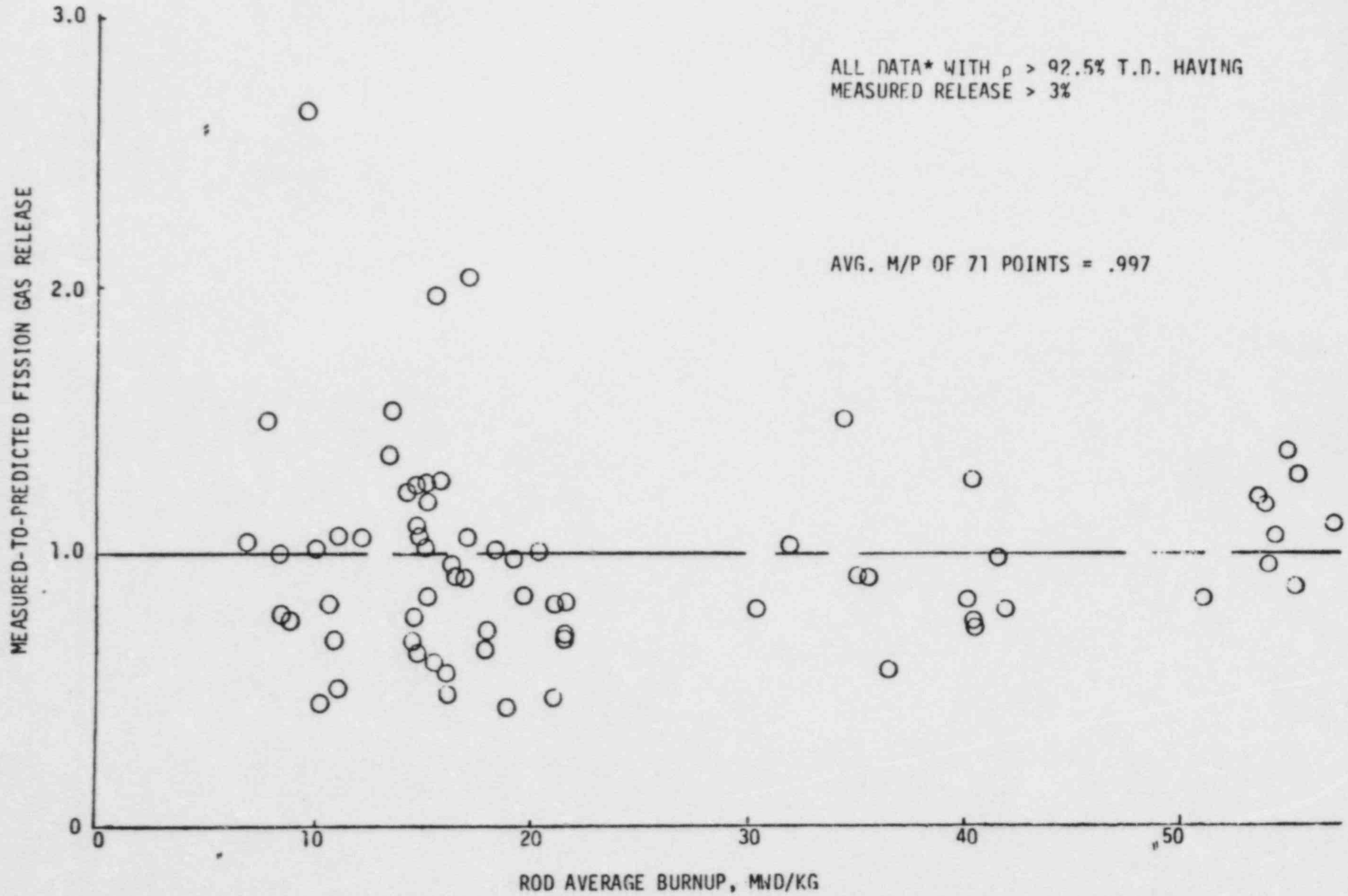
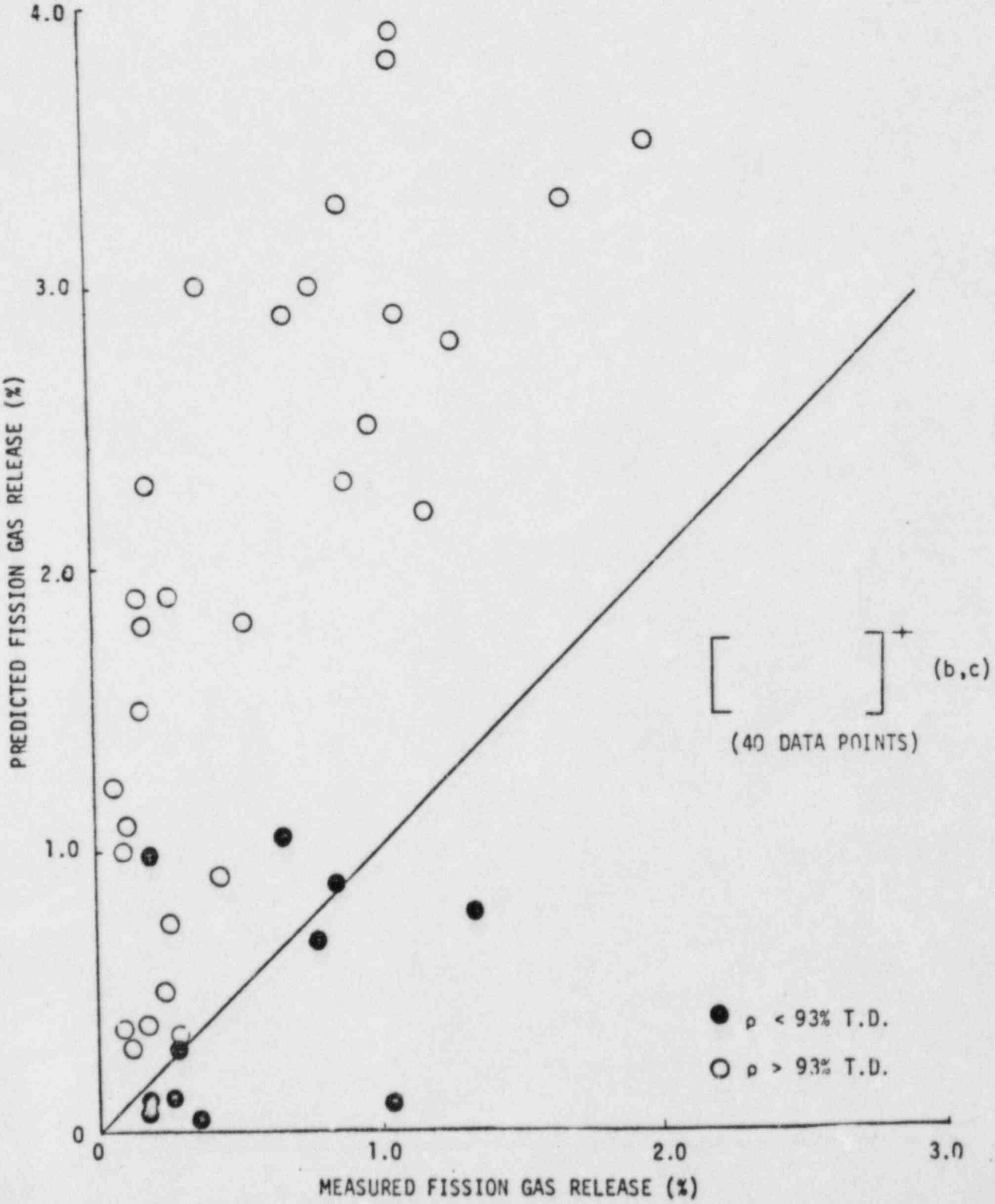


Figure 3.5
Measured/Predicted Fission Gas Release Versus Burnup

Figure 3.6

MEASURED VS. PREDICTED FISSION GAS RELEASE
LOW TEMPERATURE RODS



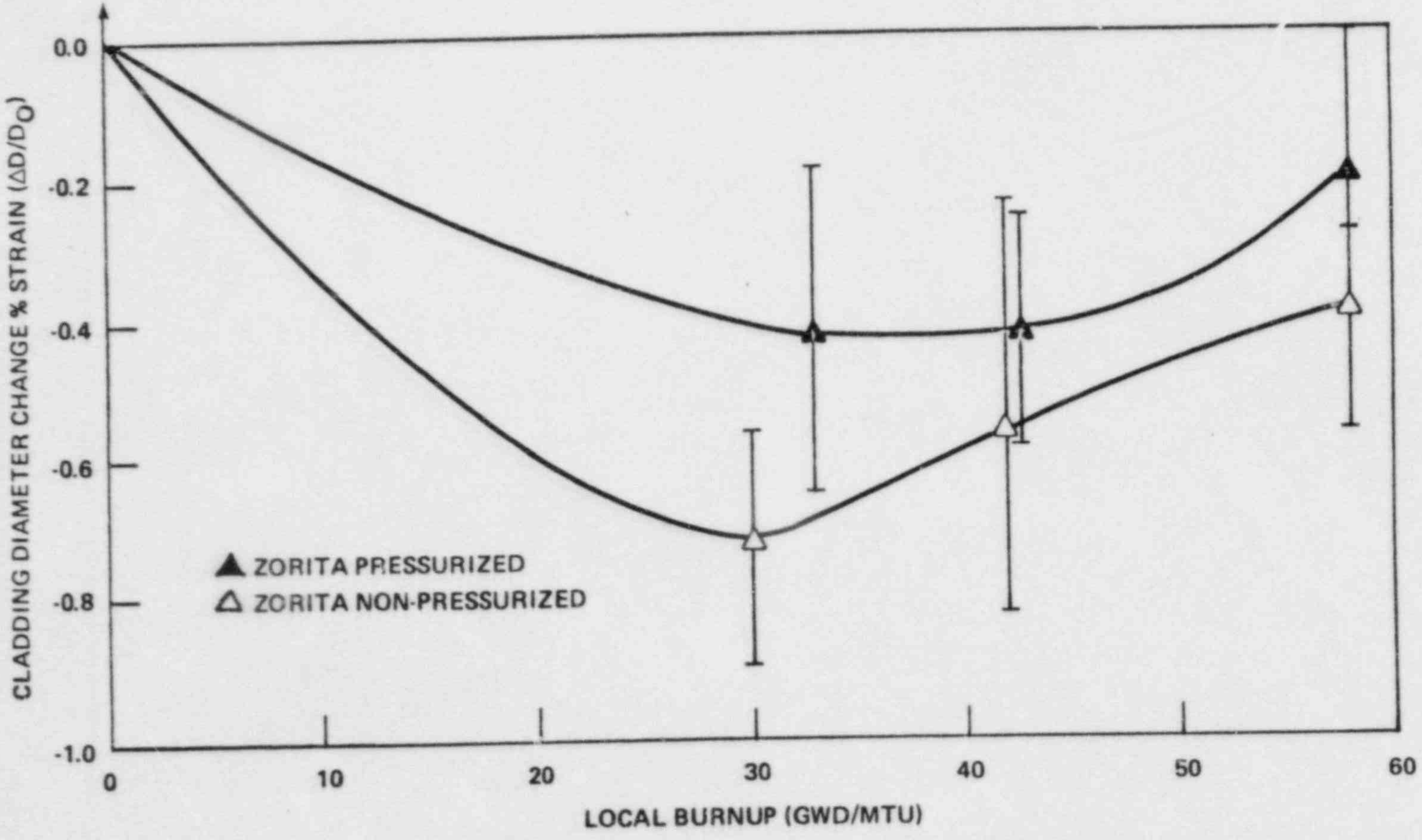


Figure 3.7
Cladding Diameter Change For High Power Rods

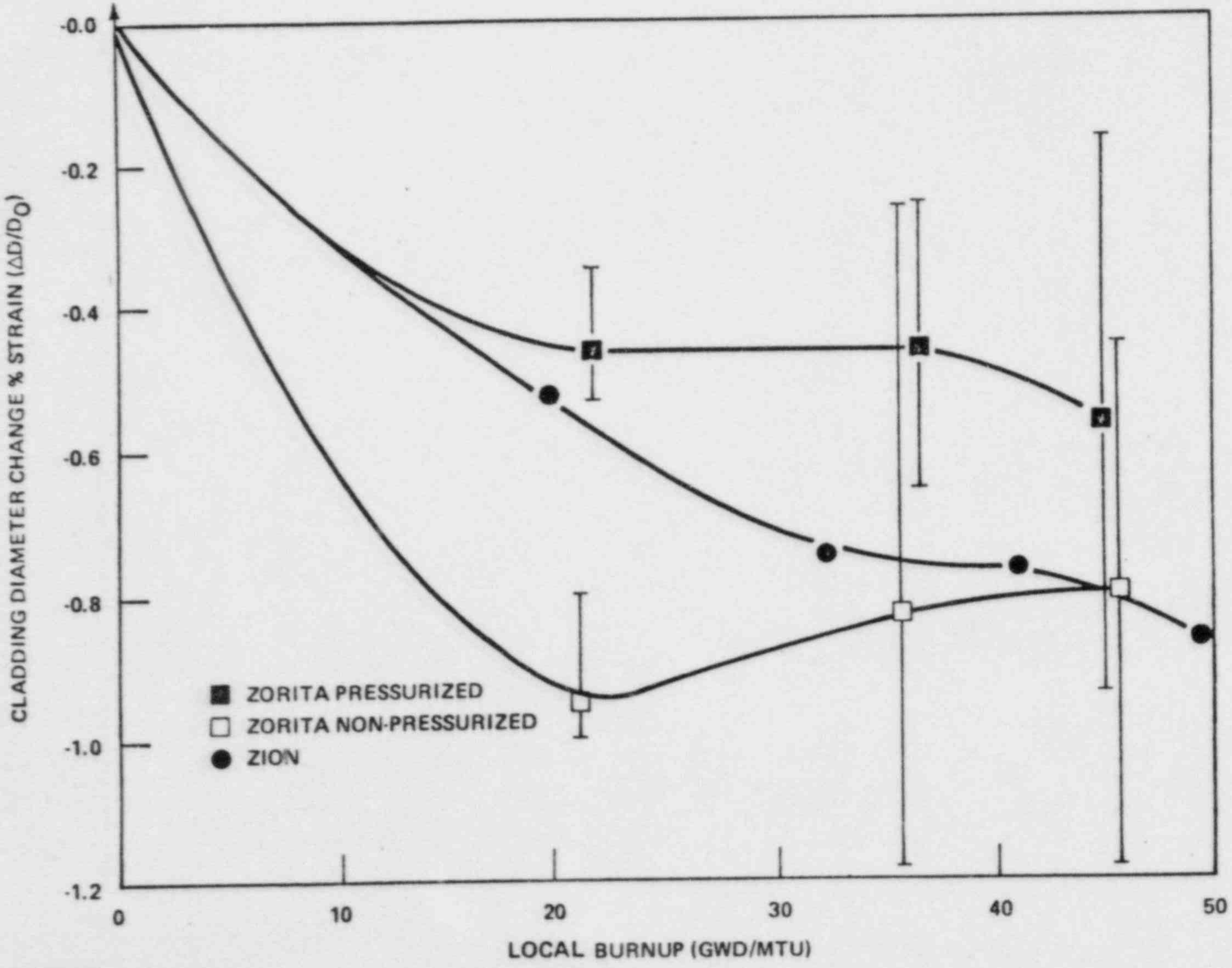
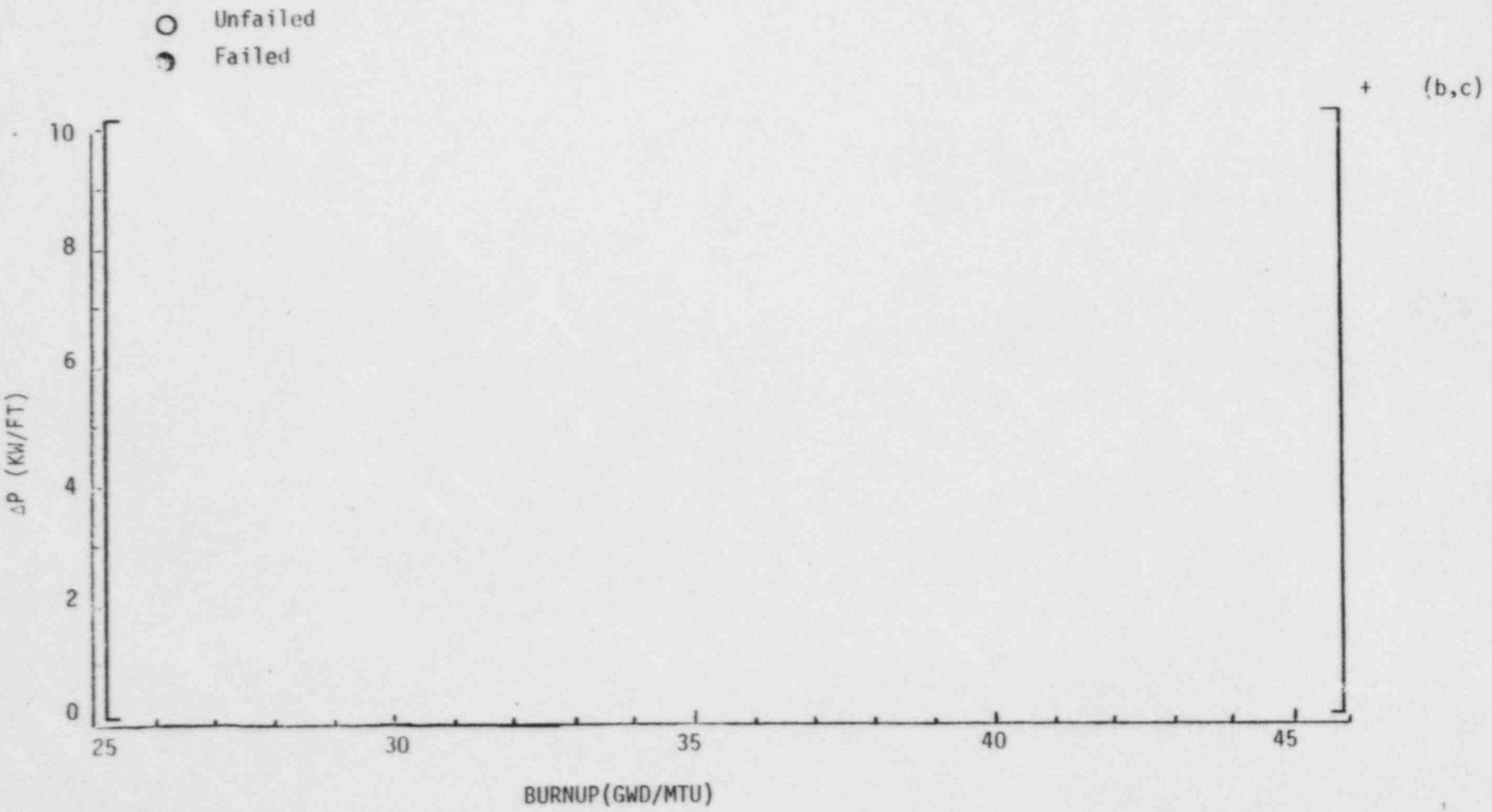


Figure 3.8
Cladding Diameter Change For Normal Power Rods

Figure 3.9

SUMMARY OF POWER RAMP TEST DATA
AT BURNUPS IN EXCESS OF 30 GWD/MTU



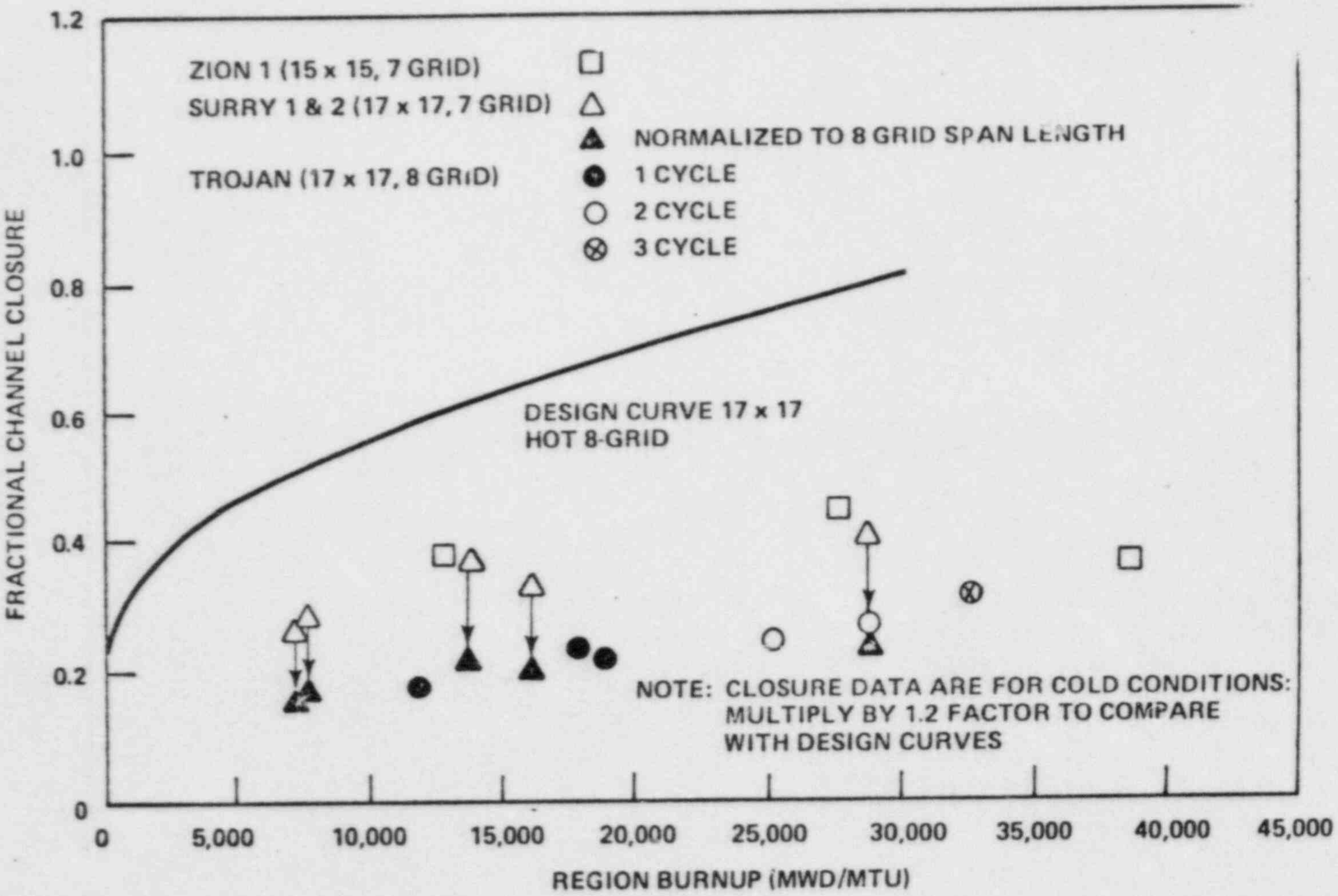


Figure 3.10
Worst Span Channel Closure At 95th Percentile Level vs. Region Burnup

4.0 CURRENT AND PLANNED EXPERIMENTAL PROGRAMS

4.1 INTRODUCTION

Westinghouse extended burnup demonstration programs have been in effect for several years and have included both commercial reactor experience and extended burnup test rod experience. The results from these measurement programs, discussed in Section 3, form the basis for the Westinghouse extended burnup design methodology and models which have been presented herein.

Currently Westinghouse is involved in several ongoing experimental programs, including Joint Industry programs as well as Westinghouse commercial and test rod programs, to further broaden the extended burnup data base. It is anticipated that data from the current and future programs discussed in this section will result in the identification of additional design margin, and will therefore be used to support future model and methodology improvements.

4.2 WESTINGHOUSE DEMONSTRATION PROGRAM

Westinghouse is currently conducting several extended burnup demonstration programs with DOE and the electric utility industry. Current programs in progress include irradiaton and testing of 15x15 assemblies to extended burnups ($\geq$4 cycles) in the Zion Unit 1 reactor, irradiation and testing of 9 high power test rods in the BR-3 test reactor, and additional testing of extended burnup fuel rods previously irradiated in the Jose Cabreara reactor. The current status and future scope of each of these programs will be discussed in detail.

In the Zion extended burnup demonstration program, four 15x15 assemblies have been irradiated through five cycles of operation and have achieved ~55000 MWD/MTU assembly average burnup, corresponding to a lead rod burnup of ~60000 MWD/MTU. These assemblies have been discharged from the core and are scheduled for leak testing and onsite non-destructive examinations in August, 1982.

On site examinations will provide extended burnup data on fuel rod and fuel assembly growth, rod bow, grid spring relaxation, fuel rod profilometry, and fuel stack length. The overall integrity of the extended burnup fuel assemblies will be assessed using high and low magnification TV visual examinations.

In addition to the onsite inspection at Zion, a two phase hot cell program is planned. In Phase I, a highly precharacterized 3 cycle assembly will be examined in the hot cell. In the second phase of the hot cell program, a five cycle, precharacterized, removeable rod assembly with ~55000 MWD/MTU burnup will be examined. The aspects of fuel performance to be addressed in this program include:

1) Fuel structural changes
2) Fuel swelling and fission gas release
3) Clad corrosion and hydriding
4) Corrosion and hydriding in guide thimble tubes
5) Guide thimble tube wear

It is anticipated that the data obtained from these examinations will identify margin in performance models, particularly in the area of fission gas release where it is expected that the Zion data for normal power rods at extended burnup will verify the conservatism of the Westinghouse fission gas release model which is based on extended burnup, high power gas release data.

In the BR-3 irradiation program, nine highly characterized 15x15 fuel rods were irradiated at lead rod power conditions to rod average burnups in excess of 60000 MWD/MTU, with peak pellet burnup exceeding 73000 MWD/MTU. These rods were examined on site, and after being confirmed to be leak tight were shipped to the CEN hot cell according to a joint DOE/Westinghouse/SCK-CEN contract. Hot cell examinations were performed in 1981 and early 1982 with the primary focus of the examinations being the collection of fission gas release and waterside corrosion data.

Fuel rod profilometry and gamma scan data were also obtained. Evaluation of the data will be performed during the remainder of 1982, and it is expected that the data will confirm the adequacy of the current data bases in these areas.

In a third hot cell program, ten spiked enrichment extended burnup Jose Cabrera fuel rods (approximately 50000 MWD/MTU) are being examined in 1982 at the Windscale hot cell facility. Additional fission gas release and clad waterside corrosion data will be obtained from these rods.

4.3 JOINT INDUSTRY PROGRAM

Several cooperative industry programs are currently in progress which are addressing specific areas of concern for increased burnup operation. These programs include the International SUPER-RAMP Project, the Battelle High Burnup Effects Program, and the TRIBULATION program. These programs will provide data in the areas of burnup dependence of PCI, extended burnup fission gas release, and the effects of transients on subsequent extended burnup operation.

The International SUPER-RAMP Project is a cooperative industry program designed to study the effects of increased burnup (>30000 MWD/MTU) on the PCI threshold of LWR fuel rods. Both PWR and BWR fuel rods are tested in the program. PWR fuel rods which were ramp tested included standard Westinghouse fuel rods. The ramp testing was performed during 1981 and early 1982, and data analysis is continuing throughout 1982. Results from the project confirm that PCI failure susceptibility is not increased with extended burnups.

The Battelle High Burnup Effects Program is a jointly sponsored international program with the purpose of investigating extended burnup effects in Zircaloy clad UO_2 LWR fuel. The program is divided into three major tasks as follows:

Task 1 - Evaluation of existing data (complete)
Task 2 - Fission gas sampling of existing rods (in progress)
Task 3 - Parameter effects study.

Under the second task, fission gas sampling and other PIE work will be performed on 45 rods with burnups ranging from 20000 to 52000 MWD/MTU. These data will supplement existing Westinghouse fission gas release data at extended burnup.

Task 3 is designed to provide well characterized data on the effects of fuel temperature, burnup, power history, and different fuel characteristics on fission product behavior. Forty rods, designed for this task, will be irradicated in the BR-3 reactor with specified power histories. Rod burnups will range from 31000 to 73000 MWD/MTU. Following irradiation, fission gas sampling and other PIE work will be performed on these rods. It is anticipated that Task 3 will provide well characterized data for the development of an extended burnup fission gas release correlation with reduced conservatisms.

The TRIBULATION program is a jointly sponsored international program with the purpose of establishing the effects of a Condition II overpower transient on the subsequent normal operation performance of PWR fuel. This will be done by comparing the performance of fuel which has been ramped to fuel which has not been ramped. Power ramps will occur at approximately 30000 MWD/MTU burnup, and subsequent burnups will be approximately 60000 MWD/MTU. Following irradiation, both destructive and non-destructive testing will be performed.

5.0 CONCLUSIONS

The following conclusions have been reached on the basis of information presented in this report:

1) Westinghouse is designing a lead rod average burnup of [ ]+ MWD/MTU. No fuel performance limitations have been identified which would preclude the fuel Westinghouse design from operating to lead rod burnups of [ ]+ MWD/MTU. (a,c) (a,c)
2) Current design and safety evaluation criteria are applicable for design to lead rod average burnup of [ ]+ MWD/MTU, therefore no modification to the current criteria is required. (a,c)

3) The effects of increased region average discharge burnup on core physics parameters are typically evaluated each cycle according to approved methodology. No modifications are required in the current evaluation methodology for design to lead rod average burnup of [ ]+ MWD/MTU. (a,c)

4) The effects of the extended burnup to lead rod burnup of [ ]+ MWD/MTU on radiological doses have been evaluated. It is concluded that the effects of extended burnup are not limiting. (a,c)
5) The Westinghouse fuel performance data base is comprehensive. It includes data from both commercial and test reactor fuel which has operated to lead rod burnups in excess of [ ]+ MWD/MTU, and includes both high power rods and rods operated at normal commercial PWR power levels. It is concluded that the current data base sufficiently bounds expected extended burnup operation to assure that the efforts of such operation have been adequately assessed. (a,c)
6) Current Westinghouse fuel performance models are based on data which include extended burnup fuel and have been shown to conservatively model the parameters of concern for extended burnup operation. It is therefore concluded that these models are adequate design tools which can be used to assure that all fuel design criteria are met in fuel designed to operate to lead rod average burnups to [ ]+ MWD/MTU. (a,c)

7) Current Westinghouse design and safety evaluation methodologies address all of the areas affected by extended burnup operation, and therefore no modifications or additions are required to justify design to extended burnup.

8) Additional data to be obtained from the on-going extended burnup experimental programs discussed in Section 4 are expected to identify additional conservatism in Westinghouse performance models and thus identify additional margin in extended burnup design.

All of the data provided in this report support the overall conclusion that Westinghouse design and safety evaluation criteria and methodology adequately address all pertinent areas of concern for design to a lead rod burnup of [ ]+ MWD/MTU, and that the current Westinghouse data base at extended burnup is sufficient to confirm that extended burnup effects are adequately treated in current Westinghouse performance models. Westinghouse fuel can achieve a lead rod average burnup of [ ]+ MWD/MTU and fuel is currently being designed to this target burnup. (a,c) (a,c)

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SECTION E

NRC LETTER FROM C. O. THOMAS,
REQUEST NUMBER 1
FOR ADDITIONAL INFORMATION ON
WCAP-10125 (P)
DATED MARCH 31, 1983
TO WESTINGHOUSE, E. P. RAHE

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

MAR 31 1983

Mr. R. P. Rahe, Manager
Nuclear Technology Division
Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Dear Mr. Rahe:

Subject: Request Number 1 for Additional Information on WCAP-10125(P)

We are currently reviewing the Westinghouse Licensing Topical Report WCAP-10125(P) entitled "Extended-Burnup Evaluation of Westinghouse Fuel".

The initial review reveals the need for the additional information indicated in the enclosure. In order to complete this review within the current scheduled time, responses to all questions should be received by NRC by May 23, 1983. Please advise D. H. Moran on 301-492-9777 if this date is unacceptable.

Sincerely,

Cecil O. Thomas

Cecil O. Thomas, Chief
Standardization & Special
Projects Branch
Division of Licensing

Enclosure:
As stated

"The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P. L. 96-511."

ENCLOSURE

FEB 17 1983

Request for Additional Information
For Extended Burnup Fuel Operation

The subjects to be covered in topical reports covering extended burnup were discussed at a series of meetings with all the fuel vendors which took place in early 1981 and were further discussed in a letter to each fuel vendor dated June 2, 1981.

The Accident Evaluation Branch (AEB) has concluded from its review so far that the subject of radiological consequences of accidents requires reiteration of the depth of coverage expected in the topical reports.

For each reload application involving extended burnups, AEB would review the effects of factors that are normally considered burnup-related and of other factors (e.g., peaking factors) caused by the extended burnup on all Standard Review Plan, Chapter 15 accidents. The following factors would be reviewed and, therefore, should be covered in the topical reports:

a. The methods by which the limiting rod (or rods) are to be determined, considering the fraction of the rods' inventory in the pellet-clad gap and peak linear heat generation rate. There must be provision for suitable conservatism in one or both factors, consistent with the usual practice for safety analyses. Documentation should be provided for any codes that are used. The equations used to determine the "gap fraction" of volatile radionuclides in failed rods should also be provided.

b. Justification, based on design of the plants, the plants' operating procedures, or other factors, for the continued use of noble gases and iodines as surrogates for the full range of fission products that are produced, considering the changes in the total inventory and mix of radioisotopes in the fuel resulting from extended burnup compared to design basis accidents involving TID 14844 assumptions. If continued use is not justified, provide descriptions of methods to be used to consider the full range of fission products in safety analyses.

c. A description of the calculation of decontamination factors for the fuel handling accidents (inside and outside of containment) considering the potential for increased pressure in the fuel rods should be provided.

d. A justification for the iodine spiking behavior modelled in accidents such as the steam generator tube rupture accident considering the "gap fraction," internal rod pressures, linear heat generation rates, and other factors should be provided. Alternatively, justification for the continued use of the model described in SRP Section 15.6.3 may be provided.

SECTION F

WESTINGHOUSE LETTER FROM E. P. RAHE,
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION ON
"EXTENDED BURNUP EVALUATION OF WESTINGHOUSE FUEL,"
WCAP-10125, (PROPRIETARY), NS-EPR-2803,
DATED AUGUST 3, 1983
TO NRC, C. O. THOMAS
(RESPONSE TO NRC REQUEST NUMBER 1)

Westinghouse Electric Corporation
Water Reactor Divisions
Nuclear Technology Division
Box 355
Pittsburgh Pennsylvania 15230

August 3, 1983
NS-EPR-2803

Dr. Cecil O. Thomas, Chief
Special Projects Branch
Division of Project Management
U.S. Nuclear Regulatory Commission
Phillips Building
7920 Norfolk Avenue
Bethesda, Maryland 20014

SUBJECT: Response to Request for Additional Information on "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125, (Proprietary)

Dear Dr. Thomas:

Enclosed are:

Twenty-five (25) copies of the Westinghouse response to your request for additional information (dated 3/31/83) on "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125, (Proprietary).

Also enclosed are:

One (1) copy of Application for Withholding (Non-Proprietary).

One (1) copy of original Affidavit (Non-Proprietary).

As stated in your letter, the attached additional information is needed to complete the NRC's review of this topical report.

This submittal contains proprietary information of Westinghouse Electric Corporation. In conformance with the requirements of 10CFR2.790, as amended, of the Commission's regulations, we are enclosing with this submittal an application for withholding from public disclosure and an affidavit. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission.

Correspondence with respect to the affidavit or the application for withholding should reference AW-83-64 and should be addressed to R. A. Wiesemann, Manager of Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P.O. Box 355, Pittsburgh, Pennsylvania 15230.

Very truly yours,

E. P. Rahe, Jr., Manager
Nuclear Safety Department

JWM/kk
Enclosures

ATTACHMENT

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION ON WCAP-10125

QUESTIONS

For each reload application involving extended burnups, AEB would review the effects of factors that are normally considered burnup-related and of other factors (e.g., peaking factors) caused by the extended burnup on all Standard Review Plan, Chapter 15 accidents. The following factors would be reviewed and, therefore, should be covered in the topical reports:

Question a. The methods by which the limiting rod (or rods) are to be determined, considering the fraction of the rods' inventory in the pellet-clad gap and peak linear heat generation rate. There must be provision for suitable conservatism in one or both factors, consistent with the usual practice for safety analyses. Documentation should be provided for any codes that are used. The equations used to determine the "gap fraction" of volatile radionuclides in failed rods should also be provided.

Response

The limiting rod, from an accident evaluation standpoint, would be the one containing the greatest inventory of volatile and gaseous radionuclides in the rod gap and thus available for release in the event the cladding is breached.

In order to provide suitably conservative results the Westinghouse fuel performance code (PAD)* was run assuming a worst case lead rod (one that maximizes gas release) for the plant with the highest core average linear heat generation rate (6.70 kw/ft). The fuel temperature versus burnup history obtained from the Westinghouse fuel performance code was input to the ANS 5.4 Standard Fission Product Release Model to determine the gap release fractions

for the noble gases and iodines as a function of burnup. The peak values calculated for the gap release fractions are listed in Table 2.3 of WCAP-10125.

With the exception of Kr-85, these peak gap release fractions are significantly smaller than the value of 0.1 given in Regulatory Guide 1.25. For Kr-85 the peak gap release fraction, while close to the value of 0.3 given in Regulatory Guide 1.25, is smaller at[ ].+ For accident evaluations, the values of gap release fractions are taken to be as specified in the Regulatory Guide thus providing additional conservatism. (a,c)

It is noted that there are times when the linear heat generation rate for a rod may exceed the power production levels discussed above. However, these would be transient in nature and would not have sufficient impact to cause gap release fractions to exceed the values specified in Regulatory Guide 1.25.

* J. V. Miller (Editor), "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations," WCAP-8720, October, 1976 (Proprietary) and WCAP-8785, October, 1976 (Non-Proprietary).

<u>Question b.</u> Justification, based on design of the plants, the plants' operating procedures, or other factors, for the continued use of noble gases and iodines as surrogates for the full range of fission products that are produced, considering the changes in the total inventory and mix of radioisotopes in the fuel resulting from extended burnup compared to design basis accidents involving TID 14844 assumptions. If continued use is not justified, provide descriptions of methods to be used to consider the full range of fission products in safety analyses.

<u>Response</u>

The consideration of noble gases and iodine as the airborne radioactive releases, is owed to the fact they are gaseous (xenon and krypton) or volatile (iodine) and thus readily transported through the atmospheric pathway. The

remaining radionuclides are neither gaseous nor volatile or, if they are, are not significant in inventory and/or radiological effects when compared with the noble gases and iodines.

Extended burnup of fuel does not change types of fission products generated but does result in some changes in the reactor core inventory of fission products. As indicated in Table 2.2 of WCAP-10125, the inventory of a particular fission product does not necessarily increase, and in many cases actually decreases, with extended fuel burnup.

Question c. A description of the calculation of decontamination factors for the fuel handling accidents (inside and outside of containment) considering the potential for increased pressure in the fuel rods should be provided.

Response

The method used to calculate the pool decontamination factor (DF) for iodine in the instance of a postulated fuel handling accident is to extrapolate the test data in Table 3-5 of WCAP-7828, obtaining values for average bubble diameter and bubble rise time for gas stored at[ ]+psig (the fuel pin internal pressure (a,c)
based on a burnup of[ ]+MWD/MTU). These extrapolated values are then (a,c)
inserted into equation 3-3 of WCAP-7828 which is solved for DF.

Question d. A justification for the iodine spiking behavior modeled in accidents such as the steam generator tube rupture accident considering the "gap fraction." internal rod pressures, linear heat generation rates, and other factors should be provided. Alternatively, justification for the continued use of the model described in SRP Section 15.6.3 may be provided.

Response

The gap fractions for iodines and the linear heat generation rates are reduced with extended fuel burnup beyond 33,000 MWD/MTU. The effects of these burnup induced changes on iodine spiking are discussed in Section 2.5.3 of WCAP-10125.

While fuel pin internal pressures do increase with extended burnup, this does not influence the iodine spiking phenomenon. Pressure buildup only occurs in fuel pins containing no cladding defects and these pins do not contribute to the iodine spike.

SECTION G

NRC LETTER FROM C. O. THOMAS,
REQUEST NO. TWO FOR
ADDITIONAL INFORMATION ON WCAP-10125(P),
DATED FEBRUARY 1, 1984
TO WESTINGHOUSE, E. P. RAHE

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

FEB 1 1984

Mr. E. P. Rahe, Manager
Nuclear Technology Division
Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Dear Mr. Rahe:

Subject: Request Number Two for Additional Information on WCAP-10125(P)

We are currently reviewing the Westinghouse Licensing Topical Report entitled "Extended-Burnup Evaluation of Westinghouse Fuel."

The initial review reveals the need for the additional information indicated in the enclosure. In order to complete these reviews within the currently scheduled time, responses to all questions should be received by NRC by February 24, 1984. Please advise D. H. Moran at (301) 492-9784 if you are unable to meet this date.

Sincerely,

Cecil O. Thomas

Cecil O. Thomas, Chief
Standardization & Special
Projects Branch
Division of Licensing

Enclosure:
As stated

QUESTIONS

WESTINGHOUSE HIGH BURNUP TOPICAL REPORT WCAP-10125

1. Several programs presented in WCAP-10125 were projected at later dates to provided additional data at extended burnups. Please provide those data in the areas of cladding corrosion, crud thickness, fission gas release, fuel swelling, rod bow, fretting wear, pellet-cladding interaction, axial fuel gaps, grid growth, fuel rod and assembly growth, and guide tube wear obtained since the publication of the subject document, i.e., during calendar years 1982 and 1983, that are applicable to Westinghouse (W) fuel. Because many of the above phenomena are design dependent, please identify the design to which the data apply. Also, these phenomena can be divided into categories of local fuel behavior (e.g., gas release), fuel rod (e.g., rod bow) and assembly (e.g., assembly growth) dependent phenomena; consequently, each data set from these categories should be referred to in terms of local peak burnup, rod average burnup, and assembly average burnup, respectively.

2. The burnout of extended burnup fuel results in lower attainable peak linear heat ratings at these extended burnups. Has a burnup dependent linear heat rating been assumed in the safety analysis of extended burnup fuel and, if so, how is the burnup dependency incorporated into plant operating limits for this fuel? Also, how are power histories derived for calculated end-of-life internal rod pressures?

3. No analysis is given of the radiological impact of extended burnup on the rod ejection accident. What effect does extended burnup have on the radiological impact of the rod ejection accident? Because some of the radioactive isotopes calculated to be released in the fuel handling accident have gone up and others stayed the same with extended burnup, it is not clear that the fuel handling accident with extended burnup fuel is bounded by the standard analysis. What effect does extended burnup have on the radiological impact of the fuel handling accident?

4. What fraction of the rod-to-nozzle gap is expected to remain for the lead burnup rod at extended burnups? How does this compare to the standard deviation of the data?

5. The analyses for transient clad strain, using an earlier version of W fuel performance code (Ref. 1) has indicated that conditions are most limiting during the second cycle of operation and that cladding strains are maintained below 1 percent. Is the second cycle of operation still limiting and less than 1 percent for cladding strain using the revised PAD code with NRC restrictions on compressive and tensile creep (Refs. 2 and 3)?

References:

1. J. V. Miller (Editor), "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations," WCAP-8720, October 1976 (Proprietary) and WCAP-8785, October, 1976 (Non-Proprietary).

2. Letter from J. F. Stolz (NRC) to T. M. Anderson (W) dated February 9, 1979.

3. Letter from H. Bernard (NRC) to E. P. Rahe (W) dated July 20, 1982, "Acceptance for Referencing of Licensing Topical Report WCAP-8720, Addendum 1."

SECTION H

WESTINGHOUSE LETTER FROM E. P. RAHE,
RESPONSE TO REQUEST NUMBER 2 FOR
ADDITIONAL INFORMATION ON WCAP-10125,
"EXTENDED BURNUP EVALUATION OF WESTINGHOUSE FUEL"
(PROPRIETARY), NS-EPR-2917,
DATED JUNE 11, 1984
TO NRC, C. O. THOMAS

Westinghouse Electric Corporation
Water Reactor Divisions
Box 355
Pittsburgh Pennsylvania 15230

NS-EPR-2917

June 11, 1984

Dr. Cecil O. Thomas, Chief
Standardization and Special Projects Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Response to Request Number 2 for Additional Information on WCAP-10125, "Extended Burnup Evaluation of Westinghouse Fuel" (Proprietary)

Ref.: Letter dated February 1, 1984 from C. O. Thomas (NRC) to E. P. Rahe, Jr. (Westinghouse)

Attention: Laurence E. Phillips, Acting Chief
Core Performance Branch

Dear Dr. Thomas:

Enclosed are:

1. Twenty-two (22) copies of the Westinghouse Response to Request Number 2 for Additional Information on WCAP-10125, "Extended Burnup Evaluation of Westinghouse Fuel" (Proprietary).

2. One (1) copy of an Application for Withholding Proprietary Information From Public Disclosure (Non-Proprietary).

Please note that additional analyses are currently being performed to augment the response to question No. 2. This information will be provided by July 31, 1984. Please refer any additional questions or concerns regarding the content of these responses to Douglas G. Bevard (412/374-5597) of my staff.

The enclosed material is submitted for your information and is to be treated as proprietary information of Westinghouse Electric Corporation. The information will be separately resubmitted in whole in conformance with the requirements of 120CFR2.790 should it be employed as part of a license application or other action identified in 10CFR2.790(a).

Correspondence with respect to the application for withholding should reference AW-84-48 and should be addressed to R. A. Wiesemann, Manager of Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P.O. Box 355, Pittsburgh, Pennsylvania 15230.

Very truly yours,

E. P. Rahe, Jr., Manager
Nuclear Safety Department

MDB/kk
Enclosures

cc: D. Moran
M. Dunenfeld

Westinghouse Electric Corporation

Water Reactor Divisions

Box 355
Pittsburgh Pennsylvania 15230

June 11, 1984

AW-84-48

Mr. Cecil O. Thomas, Chief
Standardization and Special Projects Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Response to Request Number 2 for Additional Information on WCAP-10125, "Extended Burnup Evaluation of Westinghouse Fuel" (Proprietary)

Ref.: Westinghouse Letter No. NS-EPR-2917, Rahe to Thomas, dated June 11, 1984.

Dear Dr. Thomas:

The enclosed material transmitted by the reference letter contains information proprietary to the Westinghouse Electric Corporation.

The material is not intended to be employed as part of a license application or other action identified in 10CFR2.790(a). It will be separately submitted with an Application for Withholding Proprietary Information from Public Disclosure accompanied by an Affidavit meeting the requirements of 10CFR2.790(b) prior to such use.

Accordingly, we request the material be treated as proprietary information within the provisions of 10CFR9.5(4), "Freedom of Information Act Regulations." If there is a need to make public disclosure of the material prior to a separate Westinghouse submittal for docket in accordance with the provisions of 10CFR2.790(a), please notify Westinghouse prior to making a disclosure determination.

Correspondence with respect to the proprietary aspects of this submittal should reference AW-84-48 and should be addressed to the undersigned.

Very truly yours,

Robert A. Wiesemann

R. A. Wiesemann, Manager
Regulatory and Legislative Affairs

/kk

cc: E. C. Shomaker, Esq.
Office of the Executive Legal Director, NRC

QUESTION 1:

Several programs presented in WCAP-10125 were projected at later dates to provide additional data at extended burnups. Please provide those data in the areas of cladding corrosion, crud thickness, fission gas release, fuel swelling, rod bow, fretting wear, pellet-cladding interaction, axial fuel gaps, grid growth, fuel rod and assembly growth, and guide tube wear obtained since the publication of the subject document, i.e., during calendar years 1982 and 1983, that are applicable to Westinghouse (W) fuel. Because many of the above phenomena are design dependent, please identify the design to which the data apply. Also, these phenomena can be divided into categories of local fuel behavior (e.g., gas release), fuel rod (e.g., rod bow) and assembly (e.g., assembly growth) dependent phenomena; consequently, each data set from these categories should be referred to in terms of local peak burnup, rod average burnup, and assembly average burnup, respectively.

RESPONSE:

Please see the attached report, "Update of the Extended Burnup Database for Westinghouse Fuel."

UPDATE OF THE
EXTENDED BURNUP DATABASE
FOR WESTINGHOUSE FUEL

May, 1984

Edited by
P. J. Larouere

Contributors:

C. A. Bly
B. S. Burns
E. E. DeMario
S. Fyfitch
P. J. Kersting
P. J. Larouere
W. R. Smalley
R. A. Weiner
H. W. Wilson

Approved: RS Miller
R. S. Miller, Manager
Performance Analysis

WESTINGHOUSE ELECTRIC CORPORATION
Nuclear Energy Systems
P. O. Box 355
Pittsburgh, Pennsylvania 15230

1350L:6/840604

TABLE OF CONTENTS

1.0 INTRODUCTION

The Westinghouse extended burnup topical report [1] issued in July, 1982, showed that applicable Westinghouse design criteria, performance models and methodology are sufficient for Westinghouse to design and operate fuel to a target lead rod average burnup of [ ]$^{+}$ MWD/MTU. Section 4.0 of that report described several programs which were to provide confirmatory data at extended burnup. (a,c)

The purpose of this report is to summarize the additional information which has since become available, such as that from the Zion extended burnup demonstration program [6,7] and the Surry 17X17 extended burnup demonstration program [10,11]. The Zion program irradiated 4 assemblies to lead assembly burnups of 54800 MWD/MTU. WCAP-10125 presented results through four cycles of operation; information from the fifth irradiation cycle and the hot cell examination program are presented here. In the Surry 17X17 demonstration program, one 17X17 assembly (standard fuel design, with Inconel grids) was irradiated for four cycles of operation, achieving an assembly average burnup of 42434 MWD/MTU. Data from both on-site and hot-cell examinations of this assembly are included here. In addition, analyses of data from other non-commercial programs (such as the BR-3 irradiation program[12] and the International SUPER-RAMP Project) in progress when WCAP-10125 was issued have been completed. Where applicable, the results of these analyses have also been included.

This additional data is summarized in the following paragraphs, which are modeled on Section 3 of WCAP-10125. The emphasis is on information which has been acquired since the beginning of 1982; however, limited amounts of data available prior to 1982 but not presented in WCAP-10125 have also been added where appropriate (e.g., on fuel rod and assembly growth) to provide a comparison with the extended burnup data. Where possible, figures from WCAP-10125 are reproduced here with the new information added. This should simplify comparisons with the previously reported data.

2.0 FUEL ROD PERFORMANCE

2.1 Clad Oxidation and Hydriding

Information on clad oxidation, including crud effects, and hydriding of fuel rods which was obtained after WCAP-10125 was issued may be found in References 7, 11 and 12.

2.1.1 Clad Oxidation

Additional corrosion data have been acquired for fuel rods irradiated in the Zion high burnup demonstration program, the BR-3 irradiation program, and the Surry 17X17 extended burnup demonstration program. The data in Figure 3.2 of WCAP-10125 have been reproduced here as Figure 1, and the results of the metallographic examinations of the fuel rods from these additional programs have been added. The curve showing the bound of oxide data under nominal crud has also been modified slightly to bound all of the new data. (The curve required modification to fit only one point: all other data points representing nominal crud situations were bound by the curve as shown in WCAP-10125.) The data base represented in this figure includes a maximum rod average burnup of ~61500 MWD/MTU.

Where crud deposition was nominal (< ~0.3 mils), the magnitude of waterside corrosion was fairly consistent with the previous results. The Zion and Surry data show a trend of increasing (but still very moderate) oxidation levels with burnup. From the lower curve in Figure 1, the maximum corrosion thickness expected at burnups of [ ]$^{+}$ MWD/MTU is approximately [(a,c
]$^{+}$ when crud deposition is nominal. This level of waterside (a,c
corrosion poses no limitation for extended burnup operation.

Data for the BR-3 irradiated rods included four measurements which showed anomalously high corrosion films at peak power locations. These (locally) large oxide films were associated with tenacious crud deposits, which had a maximum thickness of ~0.75 mils, or 19 microns. The oxide films under these thick crud deposits were 1 1/2 to 2 times as thick as the surrounding oxide film, where only nominal crud deposition was observed. These larger oxide thicknesses for the BR-3 rods all fell well within the previously presented (in WCAP-10125) curve for oxide data under thick crud (Figure 1); in fact, 3 of the 4 values also fall under the curve for nominal crud deposition. Despite the increased oxide levels and very high power levels and burnups, these rods showed no adverse performance effects, which is consistent with the information presented previously in WCAP-10125. The additional crud and corrosion data presented here are, in general, consistent with the data from other plants previously reported in WCAP-10125.

2.1.2 Clad Hydriding

Data on the hydrogen uptake of Zircaloy-4 cladding have been acquired for fuel rods irradiated under the Zion extended burnup demonstration program, the BR-3 irradiation program, and the Surry 17X17 extended burnup demonstration program. These new data continue to show the hydrogen uptake of Zircaloy-4 to generally lie between 5 and 20% of theoretical during oxidation, as reported in WCAP-10125.

Figure 3.3 of WCAP-10125 has been reproduced here as Figure 2, with the new hydrogen uptake versus rod average burnup data added. The BR-3 data have also been included for comparison, even though these data are from fuel rods with power histories significantly higher than those typical for commercial reactors. The arbitrary data-bounding curve has also been redrawn to include all of the new data. (Only one new data point fell outside the old curve; however, it should be noted that this point still represented <20% of the theoretical uptake.) The maximum hydrogen uptake represented by this curve is approximately [ ]+ (corresponding to a rod average burnup of [ ]+ MWD/MTU), a level which poses no limitation for the extended burnup operation of fuel cladding.

2.2 Fission Gas Release

Information on fission gas release not previously included in WCAP-10125 was obtained from References 7, 11, and 12.

The fission gas release data from the Zion extended burnup demonstration program, the BR-3 irradiation program, and the Surry 17X17 demonstration program are presented in Figure 3. The data from the Zorita (Jose Cabrera) core previously presented in Figure 3.4 of WCAP-10125 are also included for comparison.

The data from the BR-3 fuel rods, like the Zorita data previously presented, represent high power (>9 kw/ft) operating levels. Both BR-3 and Zorita spiked rods display relatively high fractional gas release (4-36%) and a dependency on fuel burnup. The measured fission gas releases for the BR-3 rods were also consistently lower than the calculated values, which offers further indication that the Westinghouse fission gas model conservatively predicts fission gas release fractions.

The data from both the Zion and Surry programs represent low-power, low-temperature UO_2 operation. Both sets of fuel rods also had consistently low gas releases: Surry data, for rod average burnups ranging from 28700 to 44900 MWD/MTU, were always $\leq$0.60%; and Zion data, for rod average burnups ranging from 19000 to ~59400 MWD/MTU, were usually <2%. While the gas release was low, a slight burnup dependency was observed for the Surry data, although the Zion data did not show the fission gas release to increase significantly with increasing burnup.

Fission gas release predictions for the Zion and Surry fuel rods are generally significantly higher than the measured fission gas release, as shown in Figure 4. (Figure 4 corresponds to Figure 3.6 in WCAP-10125, with additional points shown for the Zion and Surry rods.) This confirms that the Westinghouse fission gas release model gives conservative predictions for high burnup fission gas release for commercial reactor fuel operating conditions.

2.3 Fuel Swelling and Densification

A typographical error has been detected in Section 3.2.4 of WCAP-10125.[1] The effective swelling rate, which is currently shown as [ ]$^+$ (a $\Delta V/V$ per fission per cm^3 should read [ ]$^+$ $\Delta V/V$ per fission (a per cm^3. It is this latter value which is used in the Westinghouse fuel performance code and which was found to be consistent with data in the open literature.

Information on fuel swelling and densification not previously included in WCAP-10125 was obtained from References 7, 11 and 12. Preliminary analysis of data from the Zion extended burnup demonstration program indicates that, for stable high burnup fuel (rod average burnup of ~48000 MWD/MTU) operated under typical power conditions for a PWR, the total fuel swelling rate, including both fission product swelling and densification effects, is only about [ ]$^+$ the value [(a,b]$^+$ used in the Westinghouse fuel performance code. This means that the current Westinghouse model will conservatively predict fuel swelling behavior. Analysis of the Surry 17X17 extended burnup demonstration examination data likewise indicates that the information presented in WCAP-10125 will yield conservative fuel swelling results. Finally, analysis of the rods irradiated in the BR-3 reactor yielded an overall swelling rate of [ ]$^+$ times the value used in the fuel performance code, once again (a,b showing the WCAP-10125 value to be conservative.

In summary, the swelling and densification data which have become available since WCAP-10125 was issued support the information contained therein, and actually indicate that the swelling and densification model used by Westinghouse for design is conservative.

2.4 Clad Flattening

Information on axial fuel column gaps and clad flattening not previously presented in WCAP-10125 was obtained from References 7, 11 and 12.

No clad flattening was noted in any of the three programs providing additional data on axial fuel column gaps. Gamma activity scanning of fuel rods from the Zion high burnup demonstration program showed very little axial gap formation, with all gaps ≤0.030 inch. The gamma activity scans of all five rods from the BR-3 irradiation program showed an absence of significant axial gaps in all rods. Pellet interface gaps were detectable only in the extreme top and bottom 5 to 10 inches of the BR-3 fuel rods, disappearing in the higher power middle region of the fuel columns. Scanning of fuel rods from the Surry 17X17 extended burnup demonstration program also revealed no significant gaps in the fuel columns

These observations support the position stated in WCAP-10125 that the Westinghouse clad flattening model is conservative, and clad flattening is not anticipated for lead rod average burnups of up to [ ]$^{+}$ MWD/MTU. (a,c)

2.5 Pellet Clad Interaction

Additional PCI data for burnup in excess of 30 GWD/MTU have been obtained from a Westinghouse Power Cycling and Ramp Test Program[15] and the Petten High Burnup PWR Ramp Test Program[16]. Figure 5 is a reproduction of Figure 3.9 of WCAP-10125 with the new data added.

The Westinghouse test rods were identical to standard 17X17 fuel rods except for length, and achieved burnups of ~33 GWD/MTU. The rodlets in the Petten Program were not Westinghouse fuel rods; comparison of dimensions indicates that the pellet and clad diameters are roughly comparable to Westinghouse 14X14/15X15 fuel rods. The Petten rodlets achieved final burnups of ~31 to 46 GWD/MTU.

The additional data are generally consistent with the trends identified in Reference 9 of WCAP-10125. The failed and non-failed rods did not show any trend with burnup. In fact, there are several cases where rods did not fail at extended burnup (~45 GWD/MTU). The additional data, therefore, confirm the conclusion stated in WCAP-10125 that the available PCI data base does not indicate increased PCI susceptibility at extended burnup.

3.0 FUEL ASSEMBLY STRUCTURE PERFORMANCE

3.1 Fuel Rod Fretting

Assemblies using the Westinghouse Inconel grid design have been irradiated for five cycles (assembly average burnup of 55400 MWD/MTU) in the Zion extended burnup demonstration program[6,7] and for four cycles in the Surry 17X17 extended burnup demonstration program.[10,11] Detailed visual examinations have indicated no evidence of clad fretting (mechanical wear at the contact areas between the fuel rod and grid dimples and springs due to vibration), indicating that sufficient grid spring forces remain even after extended burnup to preclude rod-grid fretting. These results were anticipated, based on previous examinations of one- and two-cycle fuel rods in assemblies with Inconel grids. Therefore it is concluded that fretting wear is not a concern for a lead rod average burnup of [ ]+ MWD/MTU in assemblies with Inconel grids. (a,c)

Four 14X14 OFA demonstration assemblies with Zircaloy grids have completed two cycles of irradiation in the Point Beach Unit 2 reactor, three of which are currently in their third cycle of operation. Four 17X17 OFA assemblies with Zircaloy grids have now completed four cycles of irradiation: two assemblies irradiated in the Farley Unit 1 reactor reached an assembly average burnup of 39000 MWD/MTU, and two assemblies irradiated in the Salem Unit 1 reactor achieved an assembly average burnup of 35500 MWD/MTU. Two additional 17X17 assemblies are currently in their third irradiation cycle in the Beaver Valley Unit 1 reactor.

Fretting-type failures were observed at the bottom (Inconel) grid location of several rods in one of the 14X14 assemblies. The cause of these failures was traced to non-standard installation of the rods in the assembly during fabrication, rather than to the rod or grid design. It was concluded that the failed rods were unique to one assembly and have no generic implication to the OFA design. The remaining OFA assemblies (both 14X14 and 17X17) have shown no indication of fretting wear, indicating that the Zircaloy grid spring forces continue to be sufficient to preclude fretting.

To date, the data from these demonstration assemblies support the conclusion stated in WCAP-10125 that the design of the Zircaloy grids is sufficient such that fretting wear in assemblies with Zircaloy grids is not a concern for extended burnup.

3.2 Zircaloy Hydriding

Data have been obtained from the Zion extended burnup demonstration program [Reference 7] on hydrogen pickup for Zircaloy guide thimble tubes after 3 cycles of operation. The maximum guide thimble tube hydrogen concentration was [ ]$^{+}$ (beginning-of-life hydrogen content of [ ]$^{+}$ plus (a
[ ]$^{+}$ hydrogen uptake), and the concentration measured was usually less (a
than [ ]$^{+}$. (a

These values are lower than the one-cycle hydrogen concentration observed for Point Beach, indicating that the extrapolated value given in WCAP-10125 for a residence time consistent with a lead rod average burnup of [ ]$^{+}$ MWD/MTU (a
is conservative. As this conservative value extrapolated from the Point Beach data was well within the design criterion, Zircaloy hydriding of guide thimble tubes will not limit operation below a lead rod burnup limit of [ ]$^{+}$ (a
MWD/MTU.

3.3 Assembly Growth

Data on fuel assembly growth at high burnups was obtained from the Zion extended burnup demonstration program [References 6 and 7] and the Surry 17X17 extended burnup demonstration program [Reference 10]. These data are plotted in Figure 6 as a function of fast fluence (E>1 Mev). The two lines representing growth of 15X15 and 17X17 assemblies are derived from data available from Westinghouse commercial reactors. The 15X15 line represents ~50 data points (including the Zion data shown), and the 17X17 line represents ~200 data points. It should be noted that most of the 17X17 data are for fluences less than [ ]$^{+}$, and that the line shown (b
at higher fluence values is an extrapolation.

As can be seen, the fuel assembly growth increases as exposure (fluence) increases. It is important to note, however, that the highest value for assembly growth [ ]$^{+}$ shown on the y-axis of Figure 6 represents (a,c only half of the minimum amount of assembly growth permitted by Westinghouse's current design criteria.

It is concluded that fuel assembly growth will therefore not limit operation below a lead rod burnup of [
]$^{+}$. (a,c

3.4 Fuel Rod Growth and Rod-to-Nozzle Gap

Fuel rod growth data from the Zion extended burnup demonstration program [References 2-6], the Surry 17X17 extended burnup demonstration program [References 8-9] and the Trojan reactor [Reference 13] are plotted in Figure 7 as a function of fast fluence (E > 1 Mev). Adamson's [14] correlation for the growth rate of unfueled Zircaloy is also shown on Figure 7 for comparison.

For the Zion extended burnup fuel, rod growth averaged ~0.81 percent after five irradiation cycles, with values ranging from ~0.51 to ~1.2 percent. The Surry and Trojan data are slightly lower, with Surry averaging 0.57 percent rod growth (range: 0.43 to 0.68 percent) after four cycles, and Trojan averaging 0.52 percent rod growth (range: 0.30 to 0.79 percent) after three cycles of operation.

In general, the data show fuel rod growth progressing in a systematic manner with increasing burnup, although at a somewhat lower growth rate than reported by Adamson. Neither growth saturation nor increased rod growth rates are observed with increased burnup.

Rod-to-nozzle gap data are now available for Zion Unit 1 assemblies through five cycles of operation, to an assembly average burnup of 54800 MWD/MTU. The range of exposure for these data includes rods exposed to a fluence near 1.1×10^{22} nvt (E>1.0 Mev). The rod-to-nozzle gaps for the extended burnup assemblies ranged from 0 to 1.04 inches at the top and 0 to 0.403 inches at the bottom. The average total gap remaining after five irradiation cycles was 0.694 inches. Although a significant number (77 percent) of the fuel rods were in contact with the bottom nozzle, an adequate rod-to-nozzle gap existed at the top to accommodate continued rod growth (with the exception of one fuel rod which exhibited anomalously high growth). Statistical evaluation of the Zion data shows that for lead fuel rods with burnups of [ ]$^{+}$ MWD/MTU, (a
an average of [ ]$^{+}$ percent (95 percent confidence level) of the (a,
initial gap will remain.

The Surry 17X17 demonstration fuel assembly showed a total rod-to-nozzle gap range of 0.82 to 1.19 inches after four cycles of operation, with an average total gap of 1.00 inch. None of the Surry rods exhibited contact with either the top or bottom nozzles, and adequate rod-to-nozzle gap exists to accommodate continued rod growth.

The above data indicate that the initial rod-to-nozzle gap can be adequately designed to provide an acceptable gap distribution for assemblies which have a lead rod average burnup of [ ]$^{+}$ MWD/MTU. (a,

3.5 Fuel Rod Bow

Additional data on fuel rod bow have been obtained from the Zion extended burnup demonstration program [Reference 6]. Figure 3.10 of WCAP-10125, which shows the 95th percentile worst span channel closure versus burnup, is reproduced here as Figure 8 with the additional data point added. It can be seen that the Zion data point for the fifth irradiation cycle falls well below the design curve used for licensing.

Only one rod in the four extended burnup assemblies was observed to have made contact with neighboring rods; this rod had bowed to near contact (>90% closure) early in life. Leak testing after 5 cycles of operation showed all four extended burnup assemblies to be defect-free.

Rod bow data have also been recorded as part of the Surry 17X17 extended burnup demonstration program. The Surry 2 rod bow data after four cycles of exposure are consistent with earlier bow data from Surry Units 1 and 2. The maximum channel closure observed was 75 percent, and there were only two channel closures of 50 percent or more. When the worst span 95th percentile closure data are plotted as a function of burnup, the Surry 1 and 2 17x17 data fall within a narrow band well below the design curve shown in Figure 8.[10]

These additional data confirm the conclusion in WCAP-10125 that rod bow is remaining relatively stable and well below the licensing criteria at extended burnup.

3.6 Guide Thimble Tube Wear

Information on guide thimble tube wear not previously included in WCAP-10125 was obtained from Reference 7.

Components of the skeleton of one (15X15) assembly were subjected to hot cell examination as part of the Zion extended burnup demonstration program. The subject assembly had been irradiated for three cycles (during two of which it operated in RCCA core locations), followed by four years in the spent fuel storage pool.

Visual examination of the thimble tubes revealed that they were intact and in excellent condition, with no evidence of any deformation or other unusual conditions. Non-destructive eddy current scans performed on all twenty thimble tubes revealed no wall thinning in excess of 10 percent (minimum level of detection) in any of the tubes. In particular, locations just below the top grid (where vibrational wear would most likely occur due to the position of parked control rods in the assembly during operation) were closely reviewed for indications of wall thinning or other anomalies, but none were found.

These observations are consistent with those reported previously in WCAP-10125.[1,17]

4.0 SUMMARY

This report includes data from both commercial and test reactor fuel which has operated to lead rod burnups up to 61500 MWD/MTU. The information in this report supports conclusions previously stated in WCAP-10125, "Extended Burnup Evaluation of Westinghouse Fuel," that current Westinghouse fuel performance models conservatively model the parameters of concern for extended burnup operation. No performance limitations have been identified which would preclude the design of Westinghouse fuel to a lead rod average burnup of [ ]+ MWD/MTU. (a,c

It is therefore the conclusion of this report that, as stated in WCAP-10125, Westinghouse design methods and safety analyses are valid for designing fuel to achieve a lead rod average burnup of [ ]+ MWD/MTU. (a,c)

5.0 REFERENCES

[1] P. J. Kersting (editor), "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125, July, 1982 (Proprietary).

[2] H. H. Crain, et. al., "Interim Report Zion Unit 1 Cycle 1 Fuel Performance," WCAP-8837, December, 1976 (Non-Proprietary).

[3] E. J. Tarby, et. al., "Interim Report Zion Unit 1 Cycle 2 Fuel Performance," WCAP-9255, January, 1979 (Non-Proprietary).

[4] H. H. Crain, et. al., "Interim Report Zion Unit 1 Cycle 3 Fuel Performance," WCAP-9533, September, 1979 (Non-Proprietary).

[5] J. B. Melehan, et. al., "Interim Report Zion Unit 2 Cycle 4 Fuel Performance," WCAP-9845, March, 1981 (Non-Proprietary).

[6] J. A. Kuszyk, et. al., "Interim Report Zion Unit 1 Cycle 6 Fuel Performance," WCAP-10280, May 1983 (Non-Proprietary).

[7] M. G. Balfour, et. al., "Zion High Burnup Fuel Hot Cell Examination Program, Phase I" WCAP-10473, February, 1984 (Non-Proprietary).

[8] J. DeStefano, et. al., "Interim Report Surry Unit 2 End-of-Cycle 2 Onsite Fuel Examination of 17X17 Demonstration Assemblies After One Cycle of Exposure," WCAP-8873, January, 1978 (Non-Proprietary).

[9] J. DeStefano, et. al., "Interim Report Surry Unit 1 End-of-Cycle 3 Onsite Fuel Examination of 17X17 Demonstration Assemblies After Two Cycles of Exposure," WCAP-9139, June, 1978 (Non-Proprietary).

[10] W. Arbiter, "Surry Unit 2 End-of-Cycle 5 On-Site Examination of 17X17 Demonstration Fuel Assembly RD-2 After Four Cycles of Exposure," WCAP-10317, April, 1984 (Non-Proprietary).

[11] J. A. Kuszyk, et. al., "Hot Cell Examination of Surry Three- and Four-Cycle 17X17 Demonstration Fuel," WCAP-10514, April, 1984 (Non-Proprietary).

[12] M. G. Balfour, et. al., "BR-3 High Burnup Rod Hot Cell Program," WCAP-10238, November, 1982 (Non-Proprietary).

[13] J. B. Melehan, et. al., "Interim Report Trojan Cycle 3 Fuel Performance," WCAP-9963, December, 1981 (Non-Proprietary).

[14] R. B. Adamson, "Irradiation Growth of Zircaloy," NEDO-1236, July, 1976.

[15] P. J. Sipush, "Westinghouse Evaluation Report 'A' Power Cycling and Ramping of Westinghouse Rods Irradiated in the BR-3 Reactor," WCAP-10366, August, 1983 (Proprietary).

[16] J. C. LaVake and M. Gaertner, "High Burnup PWR Ramp Test Program Fourth Semi-Annual Progress Report for the Period April 1, 1982 to September 30, 1982," DOE/ET/34030-6, April, 1983.

[17] J. Skaritka (editor), "Salem Unit 1 17x17 Fuel Assembly Guide Thimble Tube Wear Examination Report," January, 1982, attachment to letter from R. H. Leasburg (W) to H. R. Denton (NRC), "Virginia Electric and Power Company North Anna Power Station Unit 2 License Condition 2.C.(15).(a)," February, 1982.

Figure 1

Cladding Oxide Thickness Data As A Function Of Burnup
(From Commercial Reactors)

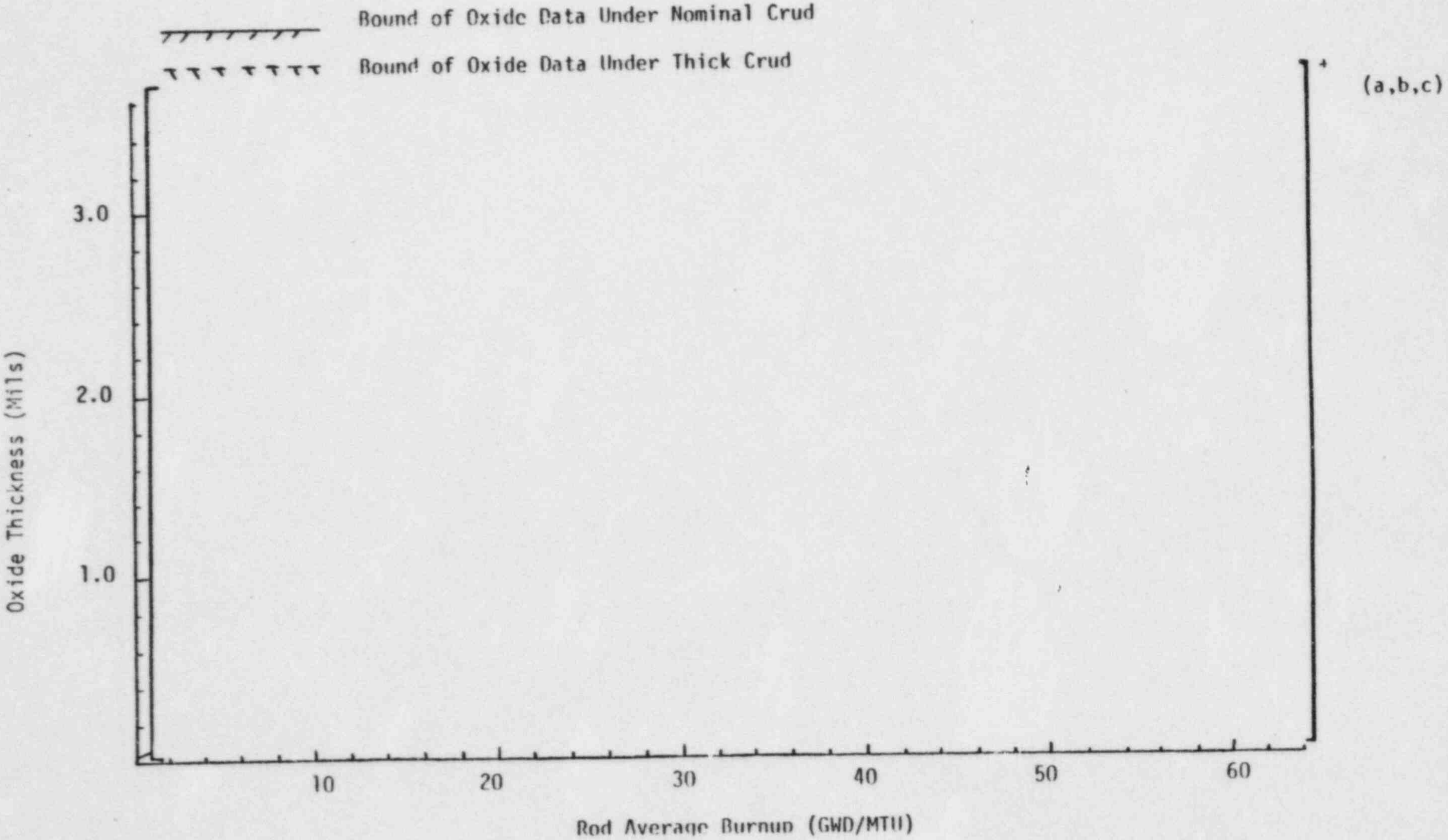
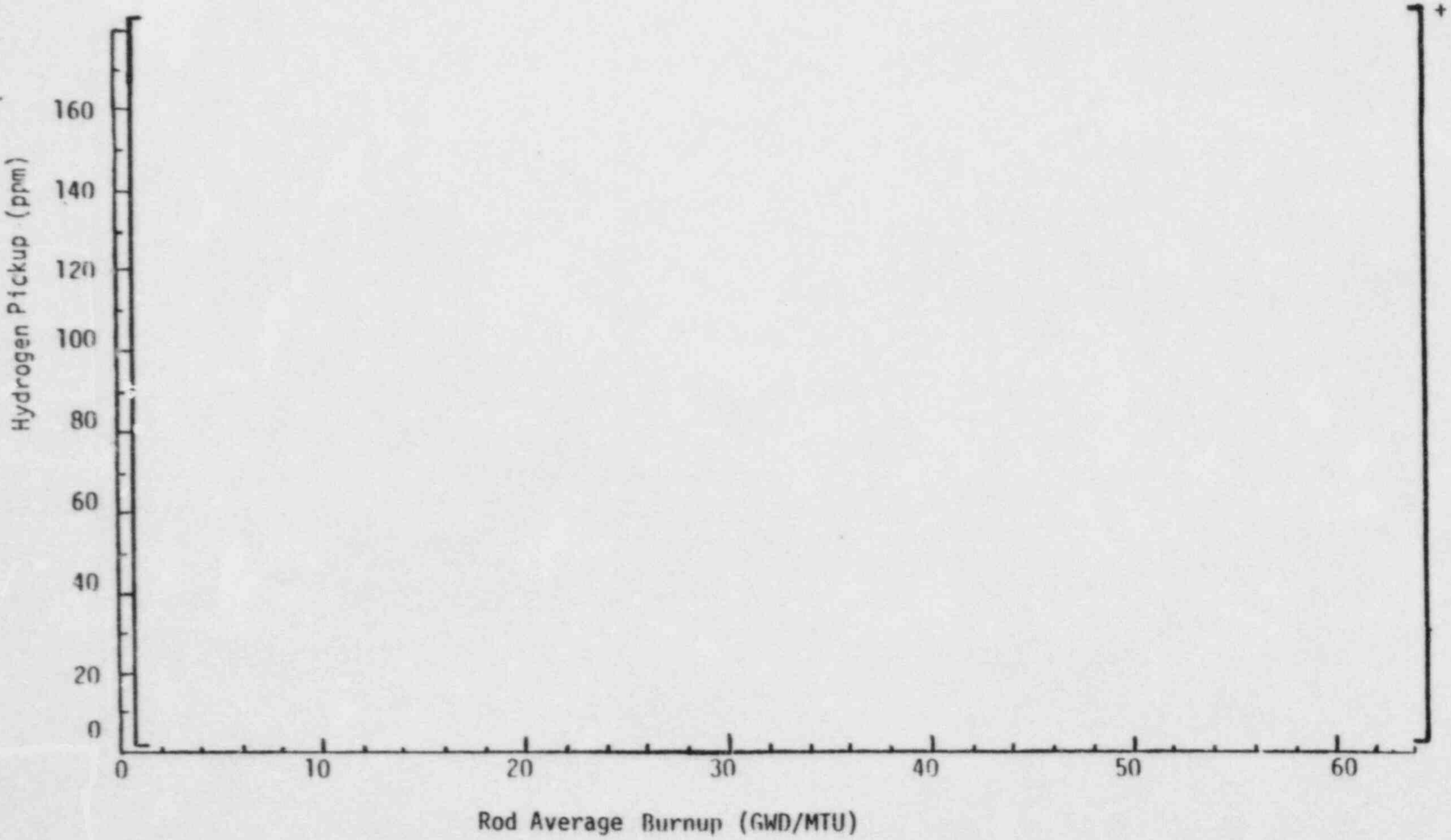


Figure 2

Cladding Hydrogen Pickup Data As A Function Of Burnup
(From Commercial Reactors)



+ (a,b,c)

Figure 3

High Burnup Fission Gas Release Data

(b,c)

Figure 4

MEASURED VS. PREDICTED FISSION GAS RELEASE
LOW TEMPERATURE RODS

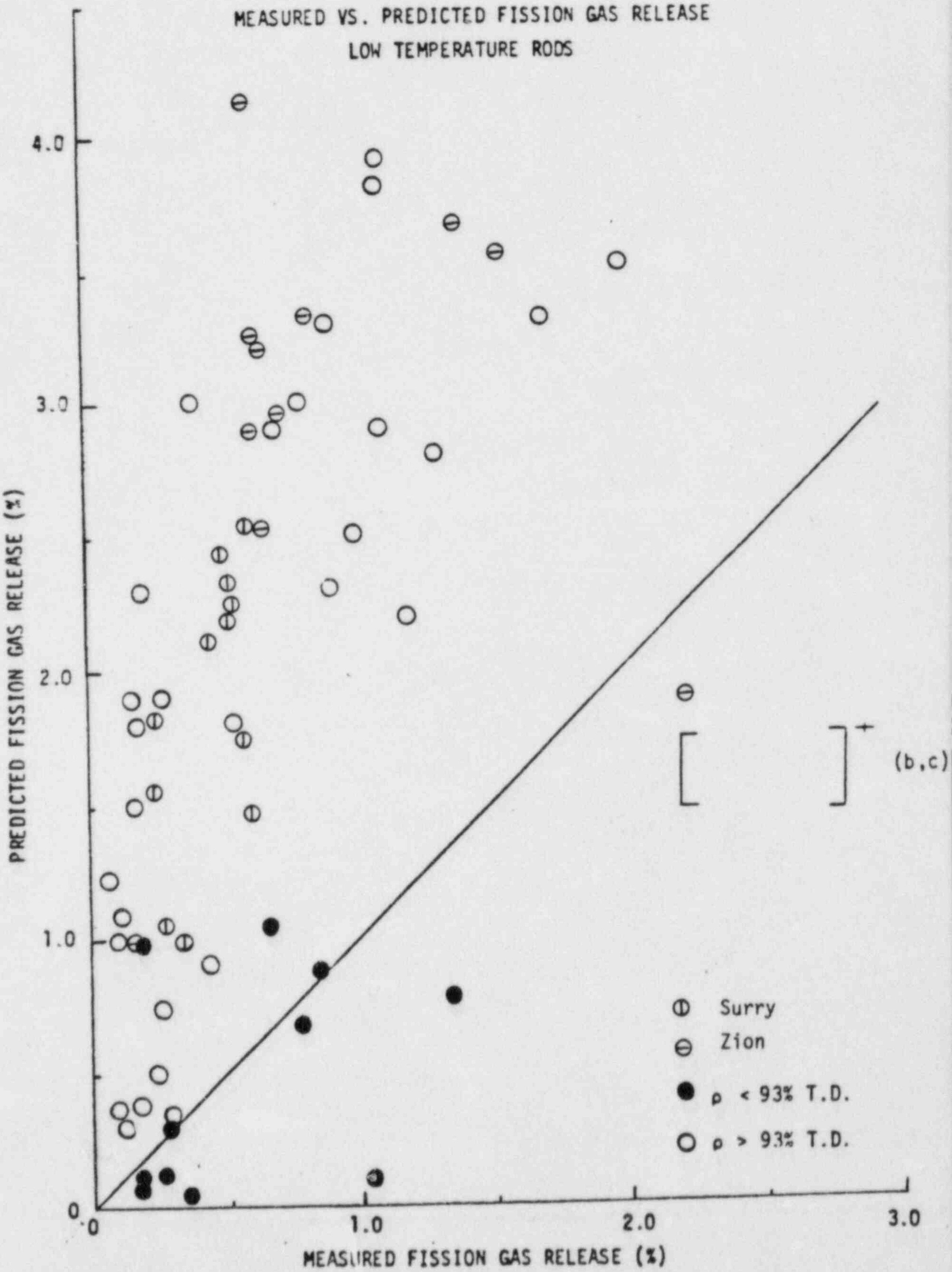
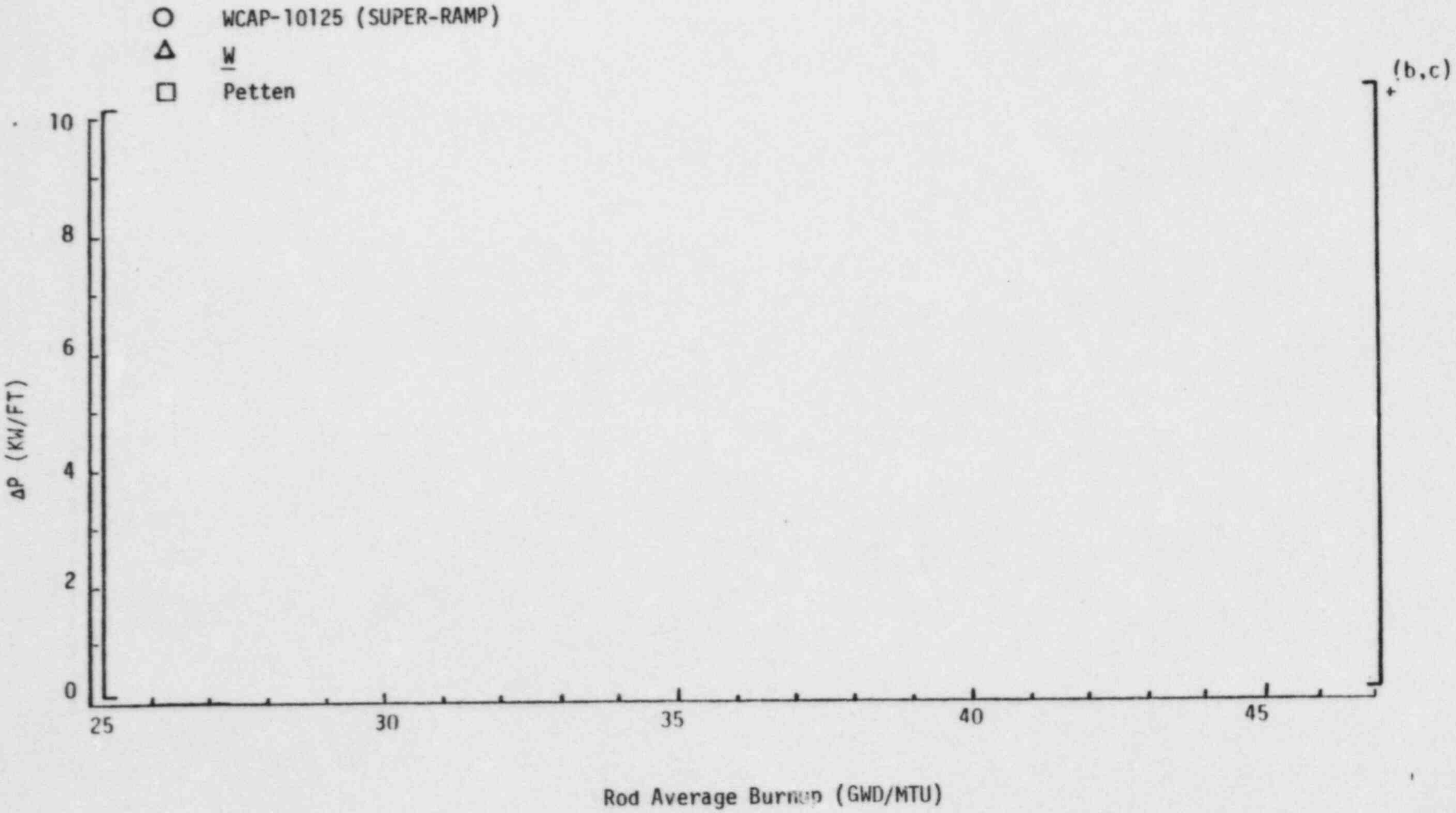


Figure 5

SUMMARY OF POWER RAMP TEST DATA
AT BURNUPS IN EXCESS OF 30 GWD/MTU



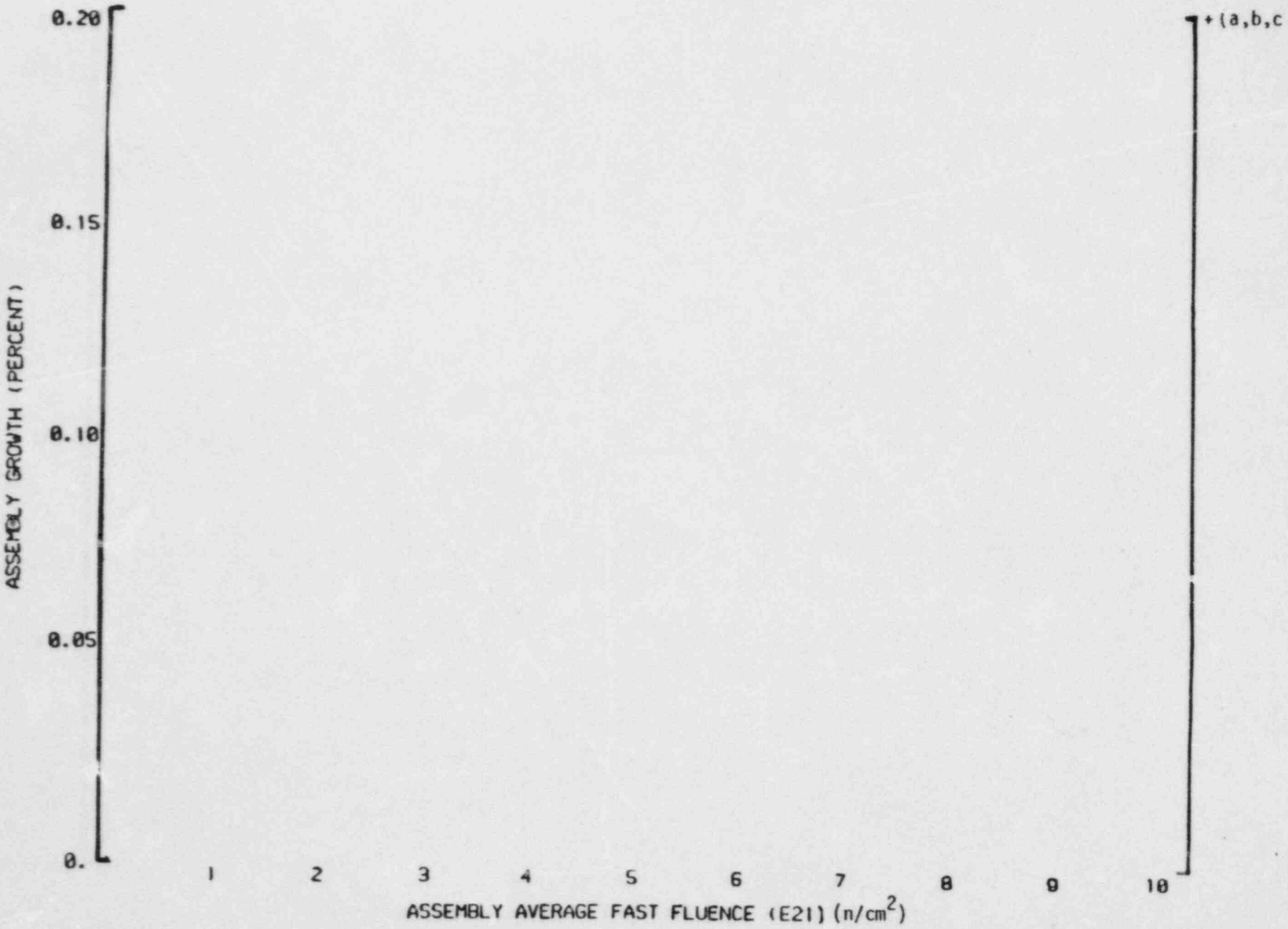


Figure 6 Assembly Growth Variation with Fluence

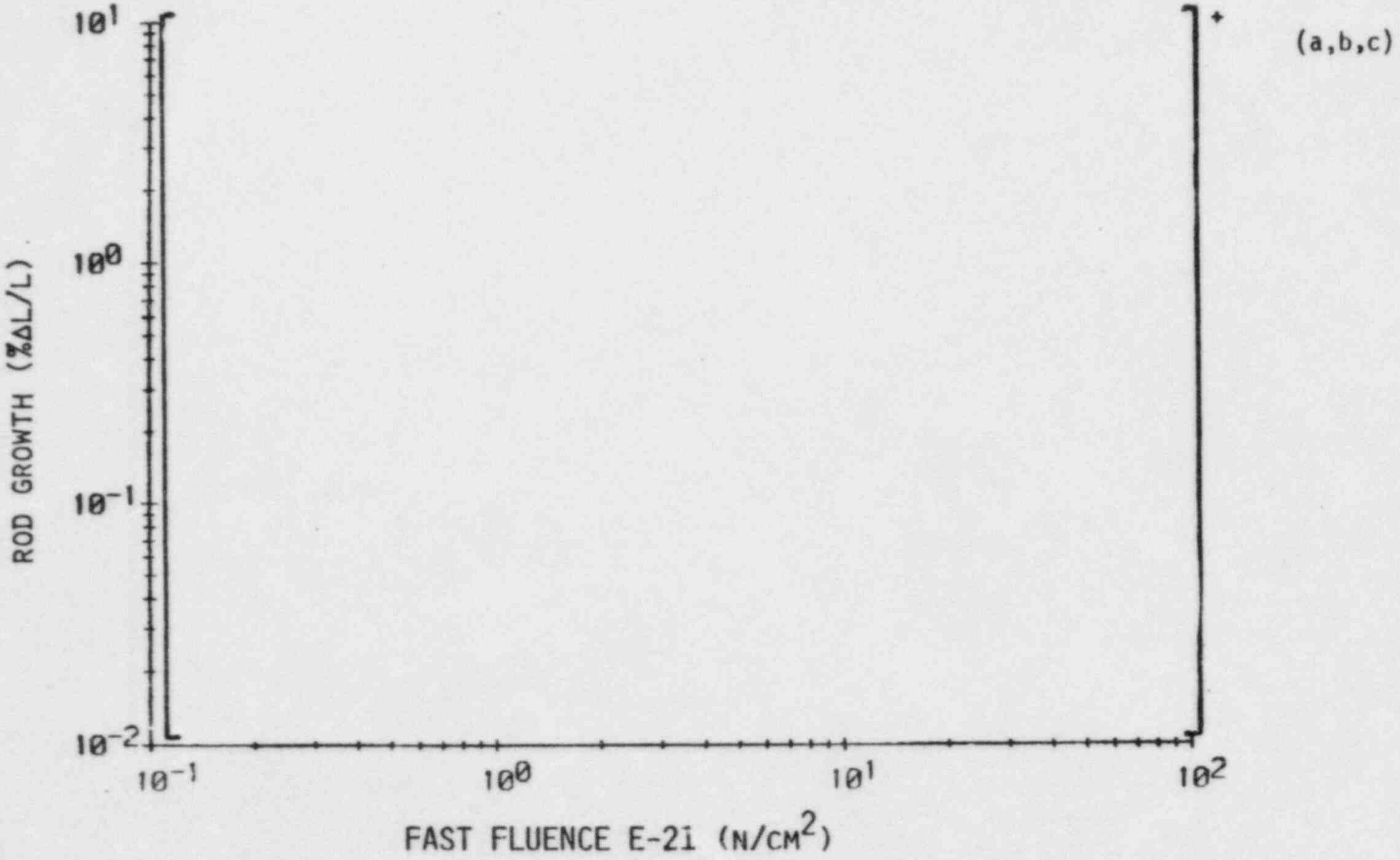


Figure 7
Fuel Rod Growth Variation with Fluence

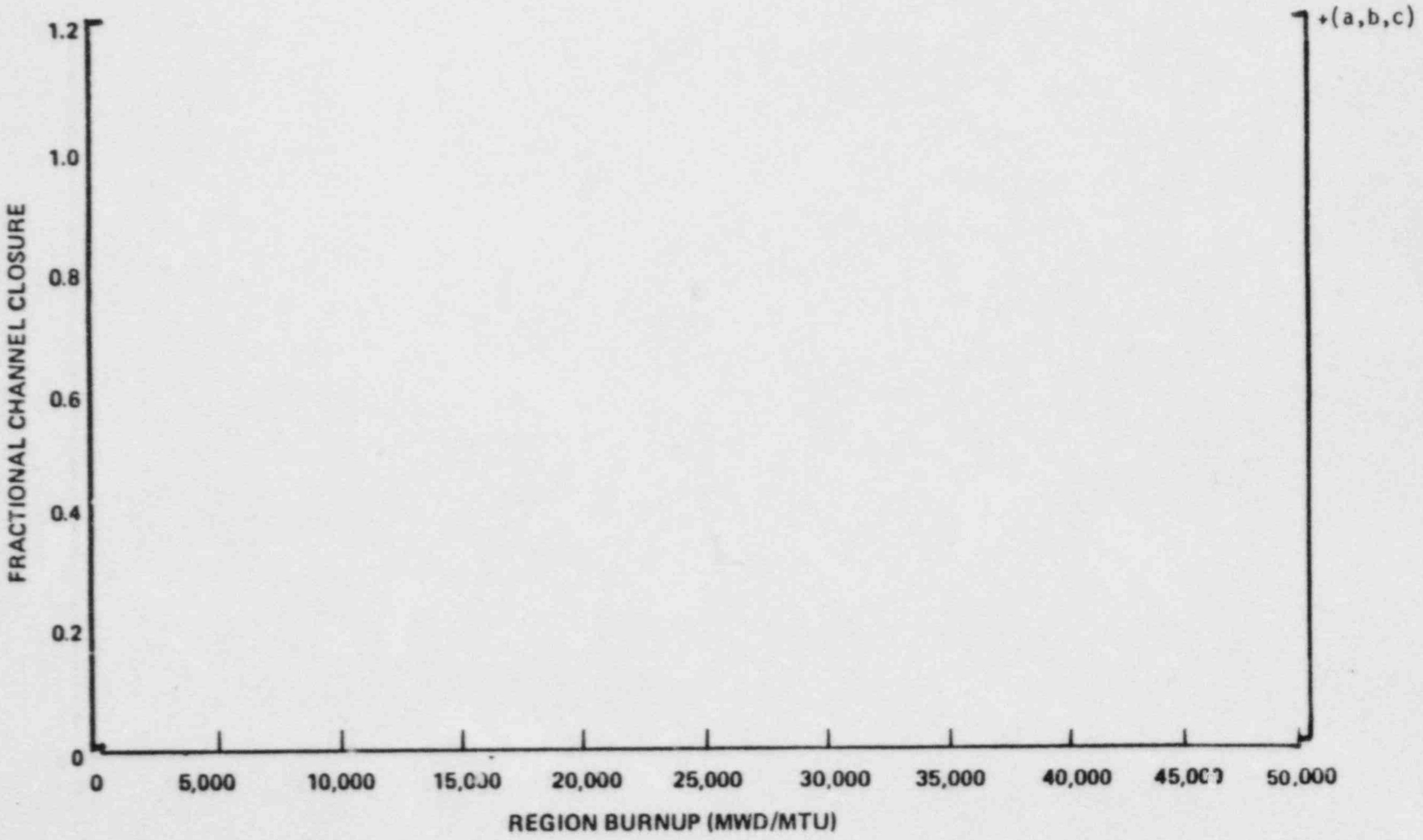


Figure 8
Worst Span Channel Closure At 95th Percentile Level vs. Region Burnup

QUESTION 2:

The burnout of extended burnup fuel results in lower attainable peak linear heat ratings at these extended burnups. Has a burnup dependent linear heat rating been assumed in the safety analysis of extended burnup fuel and, if so, how is the burnup dependency incorporated into plant operating limits for this fuel? Also, how are power histories derived for calculated end-of-life internal rod pressures?

RESPONSE:

A series of analyses were performed for 17x17 standard and OFA and 15x15 standard and OFA fuel types using fuel parameters from the standard PAD thermal model characteristic of various burnups throughout life to a maximum of [ ]+ MWD/MTU. The LOCA analyses utilized the Westinghouse 1981 (a,c)
version of the ECCS Evaluation Model. A constant peak linear heat rating consistent with beginning of life was used for each burnup out to [ ]+ (a,c)
MWD/MTU; no burnup dependent linear heat ratings were used. All analyses were performed on plants representative of those found to have the greatest sensitivity to the changes in fuel parameters due to extended burnup. The results support the conclusion stated in WCAP-10125, Section 2.5.2, that for the 1981 Evaluation Model using fuel parameters from the standard PAD thermal model that the "maximum peak clad temperature during a LOCA occurred using fuel parameters and initial conditions consistent with the time in life which exhibits the highest pellet average temperatures, near the beginning of life."

Three fuel types (14x14 standard and OFA and 16x16) are not enveloped by the work performed to date due to the lack of sufficient time to perform specific analyses. In addition, 15x15 OFA steam cooling type plants require further evaluation. Further analyses will be performed on these fuel types to evaluate their compliance with the above conclusion. The impact of extended burnup on the Evaluation Model with BART will also be evaluated as part of these studies. The results of these analyses will be provided to the NRC reviewers by the end of July.

With respect to the second part of Question 2, lead rod power histories (i.e., those power histories which result in maximum power and/or maximum burnup) based on multi-cycle fuel management calculations consistent with standard fuel rod design practice are used to calculate end-of-life rod internal pressures.

QUESTION 3:

No analysis is given of the radiological impact of extended burnup on the rod ejection accident. What effect does extended burnup have on the radiological impact of the rod ejection accident? Because some of the radioactive isotopes calculated to be released in the fuel handling accident have gone up and other stayed the same with extended burnup, it is not clear that the fuel handling accident with extended burnup fuel is bounded by the standard analysis. What effect does extended burnup have on the radiological impact of the fuel handling accident?

RESPONSE:

Rod Ejection Accident

As discussed in Section 2.5.1 of WCAP-10125, the acceptance criteria for the rod ejection accident remain unaffected by the extended burnup considerations. Thus the extent of fuel failure as a result of this accident is also unaffected. However, because of the somewhat different core inventories when operating with extended burnup fuel (see Table 2.2 in WCAP-10125), there is a minor impact on the radiological consequences in that the thyroid dose increases very slightly (less than 2 percent) and the whole body dose decreases by about 9 percent.

Fuel Handling Accident

An analysis was performed for a fuel handling accident involving fuel assemblies from a discharge region having an average burnup of [ ]+ MWD/MTU, which is consistent with extended burnup operation. The resulting releases are indicated in Table 2.4 of WCAP-10125, where they are compared with releases from a fuel handling accident involving fuel assemblies from a discharge region having an average burnup of 33000 MWD/MTU. In both cases the releases are based on the highly conservative gap release fractions given in Regulatory Guide 1.25. Based on the releases listed in Table 2.4, the effect of extended burnup on the radiological impact of a fuel handling accident is to increase the thyroid dose by 4 percent. The whole body dose is not affected. (a,c)

While the above shows a minor increase in the radiological consequences of a fuel handling accident involving extended burnup fuel, in actuality a significant decrease in the radiological consequences is expected. This is due to the fact that the actual gap release fractions for the isotopes of interest (with the exception of Kr-85) are lower at extended burnups. This is discussed in Section 2.5.3.2 of WCAP-10125.

QUESTION 4:

What fraction of the rod-to-nozzle gap is expected to remain for the lead burnup rod at extended burnups? How does this compare to the standard deviation of the data?

RESPONSE:

Analysis of the total gap data from the four assemblies in the Zion extended burnup demonstration program indicates that fuel rods with burnups of up to
[ ]+ MWD/MTU will, on the average, have [ ]+ percent of the (a,c)
initial total rod-to-nozzle gap remaining at end of life. The standard
deviation on the total gap is [ ]+ percent, which results in upper and (a,c)
lower 95 percent confidence level bounds of approximately [ ]+ and [ ]+ (a,c)
percent, respectively.

QUESTION 5:

The analyses for transient clad strain, using an earlier version of W fuel performance code (Ref. 1) have indicated that conditions are most limiting during the second cycle of operation and that cladding strains are maintained below 1 percent. Is the second cycle of operation still limiting and less than 1 percent for cladding strain using the revised PAD code with NRC restrictions on compressive and tensile creep (Refs. 2 and 3)?

RESPONSE:

Analyses performed with the current version of the Westinghouse fuel performance code, PAD 3.3[1], still indicate that, with respect to transient clad strain, conditions are most limiting during the first or second cycle of operation. Westinghouse continues to require that cladding strains be maintained below the 1 percent limit.

[1] Miller, J. V. (editor), "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations," WCAP-8720, October, 1976 (Proprietary) and WCAP-8785, October, 1976 (Non-Proprietary).

SECTION I

WESTINGHOUSE LETTER FROM E. P. RAHE,
ADDITIONAL INFORMATION IN RESPONSE TO REQUEST NUMBER 2
FOR ADDITIONAL INFORMATION ON WCAP-10125,
"EXTENDED BURNUP EVALUATION OF WESTINGHOUSE FUEL"
(PROPRIETARY), NS-EPR-2960,
DATED OCTOBER 31, 1984
TO NRC, C. O THOMAS

Westinghouse
Electric Corporation

Water Reactor
Divisions

Box 355
Pittsburgh Pennsylvania 15230

NS-EPR-2960
October 31, 1984

Dr. Cecil O. Thomas, Chief
Standardization and Special Projects Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Additional Information in Response to Request Number 2 for Additional Information on WCAP-10125, "Extended Burnup Evaluation of Westinghouse Fuel" (Proprietary)

Reference: Letter NS-EPR-2917 dated June 11, 1984 from E. P. Rahe, Jr. (W) to C. O. Thomas (NRC)

Attention: M. Dunenfeld, Core Performance Branch

Dear Dr. Thomas:

Enclosed are:

1. Twenty-two (22) copies of the Westinghouse Additional Information in Response to Request Number 2 for Additional Information on WCAP-10125, "Extended Burnup Evaluation of Westinghouse Fuel" (Proprietary).

2. One (1) copy of an Application for Withholding Proprietary Information From Public Disclosure, AW-84-88 (Non-Proprietary).

Consistent with the commitment made in the referenced letter, the enclosed information augments the initial response to question 2, and completes the Westinghouse response to "Request Number 2". Please refer any additional questions or concerns regarding the content of this response to Douglas G. Bevard of my staff.

The enclosed material is submitted for your information and is to be treated as proprietary information of Westinghouse Electric Corporation. The information will be separately resubmitted in whole in conformance with the requirements of 10CFR2.790 should it be employed as part of a license application or other action identified in 10CFR2.790(a).

Correspondence with respect to the application for withholding should reference AW-84-88 and should be addressed to R. A. Wiesemann, Manager of Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P.O. Box 355, Pittsburgh, Pennsylvania 15230.

Very truly yours,

E. P. Rahe, Jr., Manager
Nuclear Safety Department

MDB/kk
Enclosures

cc: D. Moran

Westinghouse Electric Corporation
Water Reactor Divisions
Box 355
Pittsburgh Pennsylvania 15230

October 31, 1984

AW-84-88

Dr. Cecil O. Thomas, Chief
Standardization and Special Projects Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Additional Information in Response to Request Number 2 for Additional Information on WCAP-10125, "Extended Burnup Evaluation of Westinghouse Fuel" (Proprietary)

Ref.: Westinghouse Letter No. NS-EPR-2960, Rahe to Thomas, dated October 31, 1984

Dear Dr. Thomas:

The enclosed material transmitted by the reference letter contains information proprietary to the Westinghouse Electric Corporation.

The material is not intended to be employed as part of a license application or other action identified in 10CFR2.790(a). It will be separately submitted with an Application for Withholding Proprietary Information from Public Disclosure accompanied by an Affidavit meeting the requirements of 10CFR2.790(b) prior to such use.

Accordingly, we request the material be treated as proprietary information within the provisions of 10CFR9.5(4), "Freedom of Information Act Regulations." If there is a need to make public disclosure of the material prior to a separate Westinghouse submittal for docket in accordance with the provisions of 10CFR2.790(a), please notify Westinghouse prior to making a disclosure determination.

Correspondence with respect to the proprietary aspects of this submittal should reference AW-84-88 and should be addressed to the undersigned.

Very truly yours,

R. A. Wiesemann, Manager
Regulatory and Legislative Affairs

/kk
cc: E. C. Shomaker, Esq.
Office of the Executive Legal Director, NRC

Question 2

The burnout of extended burnup fuel results in lower attainable peak linear heat ratings at these extended burnups. Has a burnup dependent linear heat rating been assumed in the safety analysis of extended burnup fuel and, if so, how is the burnup dependency incorporated into plant operating limits for this fuel? Also, how are power histories derived for calculated end-of-life internal rod pressures?

Response

This response supersedes the partial response to this question included in our June 11, 1984 submittal. (Reference NS-EPR-2917, letter from E. P. Rahe, Jr. to C. O. Thomas).

Heat generation rates in PWR fuel experience a "burndown" due to a decrease in a concentration of fissionable isotopes and the buildup of fission product inventory. This physical phenomenon has been an inherent consideration in defining limiting safety analysis conditions, including verification of the beginning-of-life LOCA worst case condition. In addition, the evaluation of actual "burndown" on a cycle specific basis is a valid element in vendor and licensee review of safety related considerations.

A series of representative analyses were performed for all Westinghouse standard and optimized fuel assembly designs using fuel parameters from the standard PAD thermal safety model characteristic of various burnups throughout life to a lead rod average burnup of [ ] (a,c) to confirm that BOL is the most limiting time in life for LOCA analyses. The extended burnup LOCA analyses utilized ECCS Evaluation Model versions under which the representative plants are currently licensed. Analyses were performed on plants representative of those found to have the greatest sensitivity to the changes in fuel parameters due to extended burnup. The results of these analyses have verified the conclusion stated in WCAP-10125, Section 2.5.2, that:

> "the maximum peak clad temperature during a LOCA occurred using fuel parameters and initial conditions consistent with the time in life which exhibits the highest pellet average temperatures, near the beginning of life."

In addition, analyses were performed which encompass all Westinghouse plant configurations approved for applications with the BART Evaluation Model which also verified the above conclusion for burnups to [ ]. For Westinghouse two loop plants, fuel burnup analyses support (a,c) the above conclusion with the LOCA Evaluation Model for which they were licensed as of September 1984.

It should also be noted that the phenomenon of "burndown" is not controlled by any particular plant operating limit (e.g. axial offset operating band, control rod insertion). The level of "burndown" is inherent in the nature of the core design and physical behavior and thus no change to plant operating limits is required to include this on a conservative basis.

With respect to the second part of Question 2, lead rod power histories (i.e., those power histories which result in maximum power and/or maximum burnup) based on multi-cycle fuel management calculations consistent with standard fuel rod design practice are used to calculate end-of-life rod internal pressures.

SECTION J

WESTINGHOUSE LETTER FROM E. P. RAHE,
RESPONSE TO INFORMAL NRC QUESTIONS ON WCAP-10125,
"EXTENDED BURNUP EVALUATION OF WESTINGHOUSE FUEL,"
JULY 1982, NS-NRC-2992,
DATED JANUARY 7, 1985
TO NRC, C. O. THOMAS

Westinghouse Electric Corporation
Water Reactor Divisions

Box 355
Pittsburgh Pennsylvania 15230

January 7, 1985
NS-NRC-85-2992

Dr. Cecil O. Thomas, Branch Chief
Standardization and Special Projects Branch
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: Response to Informal NRC Questions on WCAP-10125, "Extended Burnup Evaluation of Westinghouse Fuel", July 1982

REFERENCE: Westinghouse Letter No. NS-NRC-84-2980, Rahe to Dunenfeld, dated November 26, 1984

Dear Dr. Thomas:

In accordance with the referenced letter, enclosed are:

1. One (1) copy (attachment 1) of the response (Proprietary) to the questions on the Extended Burnup Topical Report.

2. One (1) copy (attachment 2) of the response (Non-Proprietary) to the questions on the Extended Burnup Topical Report.

3. One (1) copy of an Application for Withholding, AW-85-004 (Non-Proprietary).

4. One (1) copy of an original Affidavit, AW-76-21 (Non-Proprietary).

This submittal contains proprietary information of Westinghouse Electric Corporation. In conformance with the requirements of 10CFR2.790, as amended, of the Commission's regulations, we are enclosing with this submittal an application for withholding proprietary information from public disclosure and an affidavit. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission.

Correspondence with respect to the affidavit or application for withholding should reference AW-85-004 and should be addressed to R. A. Wiesemann, Manager of Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230.

Very truly yours,

E. P. Rahe, Jr., Manager
Nuclear Safety Department

GWH/kk
Attachments

Westinghouse Electric Corporation
Water Reactor Divisions
Box 355
Pittsburgh Pennsylvania 15230

January 7, 1985
AW-85-004

Dr. Cecil O. Thomas, Chief
Standardization and Special Projects Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Response to Informal NRC Questions on WCAP-10125, "Extended Burnup Evaluation of Westinghouse Fuel", July 1982

Ref.: Westinghouse Letter No. NS-NRC-85-2992, Rahe to Thomas, dated January 7, 1985

Dear Dr. Thomas:

This application for withholding is submitted by Westinghouse Electric Corporation ("Westinghouse") pursuant to the provisions of paragraph (b) (1) of Section 2.790 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is of the same technical type as that proprietary material previously submitted with affidavit, AW-76-21, signed by the owner of the proprietary information, Westinghouse Electric Corporation.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-85-004 and should be addressed to the undersigned.

Very truly yours,

Robert A. Wiesemann

Robert A. Wiesemann, Manager
Regulatory & Legislative Affairs

/kk
cc: E. C. Shomaker, Esq.
Office of the Executive Legal Director, NRC

PROPRIETARY INFORMATION NOTICE

TRANSMITTED HEREWITH ARE PROPRIETARY AND/OR NON-PROPRIETARY VERSIONS OF DOCUMENTS FURNISHED TO THE NRC IN CONNECTION WITH REQUESTS FOR GENERIC AND/OR PLANT SPECIFIC REVIEW AND APPROVAL.

IN ORDER TO CONFORM TO THE REQUIREMENTS OF 10CFR2.790 OF THE COMMISSION'S REGULATIONS CONCERNING THE PROTECTION OF PROPRIETARY INFORMATION SO SUBMITTED TO THE NRC, THE INFORMATION WHICH IS PROPRIETARY IN THE PROPRIETARY VERSIONS IS CONTAINED WITHIN BRACKETS AND WHERE THE PROPRIETARY INFORMATION HAS BEEN DELETED IN THE NON-PROPRIETARY VERSIONS ONLY THE BRACKETS REMAIN, THE INFORMATION THAT WAS CONTAINED WITHIN THE BRACKETS IN THE PROPRIETARY VERSIONS HAVING BEEN DELETED. THE JUSTIFICATION FOR CLAIMING THE INFORMATION SO DESIGNATED AS PROPRIETARY IS INDICATED IN BOTH VERSIONS BY MEANS OF LOWER CASE LETTERS (a) THROUGH (g) CONTAINED WITHIN PARENTHESES LOCATED AS A SUPERSCRIPT IMMEDIATELY FOLLOWING THE BRACKETS ENCLOSING EACH ITEM OF INFORMATION BEING IDENTIFIED AS PROPRIETARY OR IN THE MARGIN OPPOSITE SUCH INFORMATION. THESE LOWER CASE LETTERS REFER TO THE TYPES OF INFORMATION WESTINGHOUSE CUSTOMARILY HOLDS IN CONFIDENCE IDENTIFIED IN SECTIONS (4)(ii)(a) through (4)(ii)(g) OF THE AFFIDAVIT ACCOMPANYING THIS TRANSMITTAL PURSUANT TO 10CFR2.790(b)(1).

AW-76-21

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Robert A. Wiesemann, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Corporation ("Westinghouse") and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Robert A. Wiesemann

Robert A. Wiesemann, Manager
Licensing Programs

Sworn to and subscribed
before me this 9th day
of [illegible] 1976.

[illegible]

Notary Public

(1) I am Manager, Licensing Programs, in the Pressurized Water Reactor Systems Division, of Westinghouse Electric Corporation and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing or rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Water Reactor Divisions.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse Nuclear Energy Systems in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(g) It is not the property of Westinghouse, but must be treated as proprietary by Westinghouse according to agreements with the owner.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition in those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information is not available in public sources to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in the attachment to Westinghouse letter number NS-CE-1100, Eicheldinger to Vassallo, dated June 11, 1976, concerning THINC II/THINC IV code comparisons. The letter and attachment are being submitted in response to the NRC request at the May 28, 1976 NRC/Westinghouse meeting.

This information enables Westinghouse to:

(a) Justify the Westinghouse design correlations.

(b) Assist its customers to obtain licenses.

(c) Obtain preliminary design approvals.

(d) Meet warranties.

(e) Provide greater flexibility to customers assuring them of safe and reliable operation.

(f) Reduce plant and fuel costs.

(g) Optimize performance while maintaining high level of fuel integrity.

Further, the information gained from the THINC IV development program is of commercial value and is sold for considerable sums of money as follows:

(a) Westinghouse sells the use of this information to foreign licensees.

(b) Westinghouse uses the information to perform and justify analyses which are sold to customers.

(c) Westinghouse sells testing services based upon the experience gained and the test equipment and methods developed.

Public disclosure of this information concerning THINC II/THINC IV code comparisons is likely to cause substantial harm to the competitive position of Westinghouse because competitors could utilize this information to assess and justify their own designs without commensurate expense.

The comparisons performed and their evaluation represent a considerable amount of highly qualified development effort. This work was contingent upon a THINC IV development program which has been underway during the past six years. Altogether, a substantial amount of money and effort has been expended

by Westinghouse which could only be duplicated by a competitor if he were to invest similar sums of money and provided he had the appropriate talent available.

Further the deponent sayeth not.

ATTACHMENT 2

RESPONSE TO INFORMAL NRC QUESTION
ON POWER HISTORIES AND FISSION GAS
RELEASE UNCERTAINTIES USED IN EOL
ROD INTERNAL PRESSURE CALCULATIONS

NON-PROPRIETARY VERSION

WESTINGHOUSE ELECTRIC CORPORATION

Response to Informal NRC Question on Power Histories and Fission Gas Release Uncertainties used in EOL Rod Internal Pressure Calculations

1. Provide sample rod power history data and associated rod internal pressure versus burnup curves for the analysis of EOL rod internal pressure.

Response

As described during the telecon on 8/17/84, the power histories used in design represent the power duty experienced by individual rods which are followed through each cycle of operation until discharge. To define a set of sample power histories for the rod internal pressure analysis, it was agreed that the power histories to be provided would represent the following cases selected from a single fuel region for core designs which have peak rod powers near the radial peaking factor design limit:

1) Peak rod power occurring in the first cycle
2) The rod with the peak rod power in the second cycle
3) The rod with the peak power in the third cycle

Sample power histories from a core which is representative of anticipated high burnup fuel operation were selected according to the above criteria. The four cases selected are:

1-2) Peak rod power occurring in the first cycle - these represent the maximum power rods in the core (Figures 1 and 2)
3) Rod power history from the same region having the highest power in the second cycle. (Figure 3)
4) Rod power history from the same region having the highest power in the third cycle. (Figure 4)

Rod internal pressure calculations were performed for each of these cases using the PAD code (WCAP-8720)with and without uncertainties on the fission gas release model. These PAD cases were run with the standard 17X17 fuel design configuration, and the resulting plots of rod internal pressure versus rod average burnup are provided in Figures 5 through 8.

In addition to these cases it was requested in the telecon that we consider the hypothetical case where the maximum rod power experienced during the second cycle of operation is assumed to be equal to the design radial peaking factor limit. In order to evaluate this condition, the power history for cycle 2 for case 3 above was multiplied by the factor required to result in a maximum power level of 1.435. This factor was applied for every time interval in cycle 2 so that the time varying behavior was preserved. The resulting power history is shown in Figure 9, and the rod internal pressures versus burnup for this case are shown in Figure 10.

2. Provide a comparison of measured fission gas release data with the PAD predicted fission gas release including uncertainties.

Response

The Westinghouse fission gas release model (WCAP-8720) predictions with uncertainties as used in design are compared with fission gas release data in Figures 11 through 13. Figure 11 shows the comparison with all the data. Due to the large number of data points with measured fission gas releases below 3%, comparisons are also shown for measured fission gas release greater than 3%, Figure 12, and for measured fission gas release less than 3%, Figure 13. The data used in the comparison are restricted to rods typical of the current Westinghouse pressurized design. All the data used in the model development are shown in the figures, along with data from the Zion, Surry and BR3 irradiation programs which have become available since the PAD3.3 fission gas release model was reviewed and approved by the NRC. As can be seen from the figures, the model with fission gas release uncertainties included conservatively bounds the fission gas release data.

FIGURE 1

+(a,b,c)

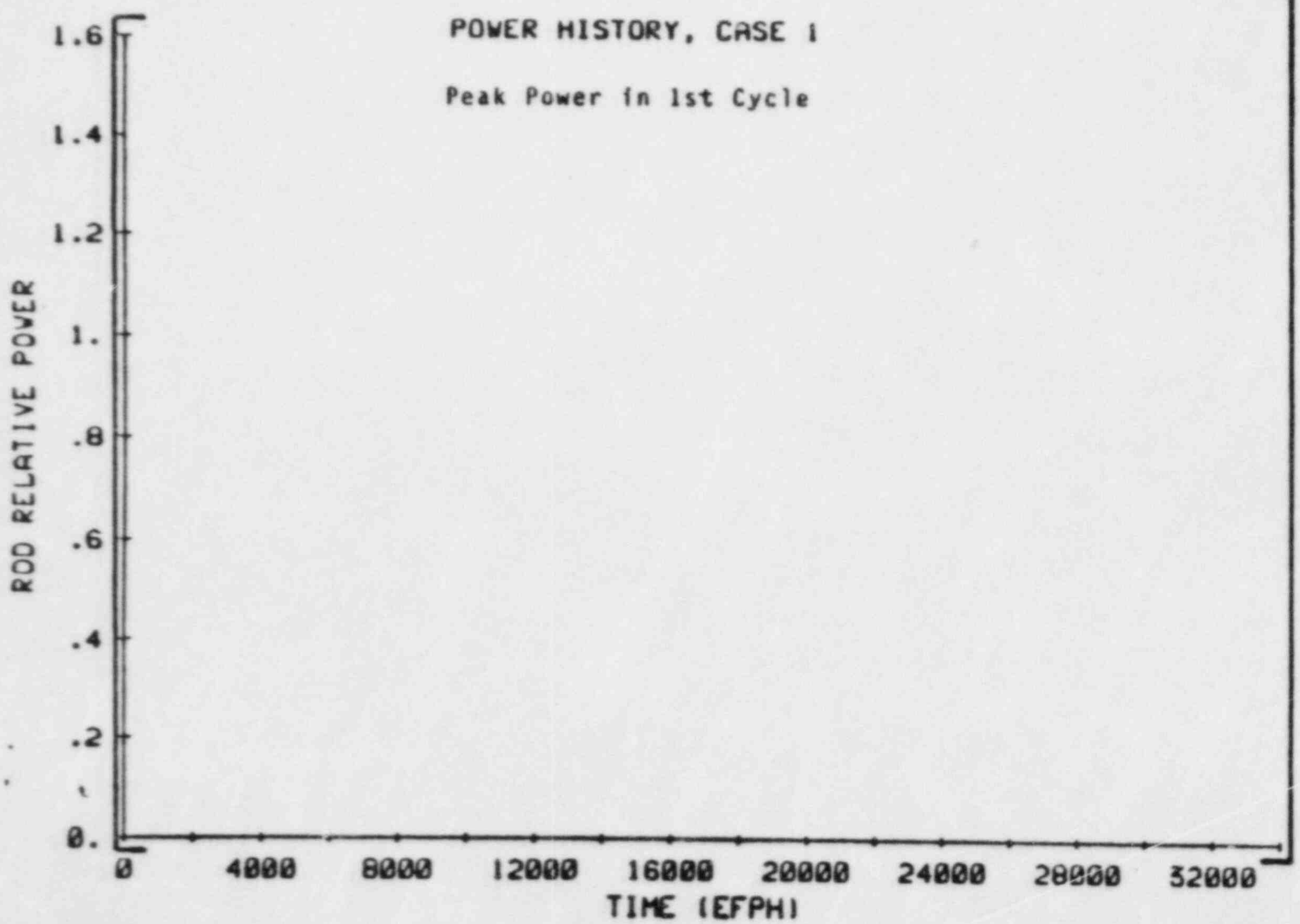

POWER HISTORY, CASE 1
Peak Power in 1st Cycle
ROD RELATIVE POWER
1.6
1.4
1.2
1.
.8
.6
.4
.2
0.
0
4000
8000
12000
16000
20000
24000
28000
32000
TIME (EFPH)

FIGURE 2

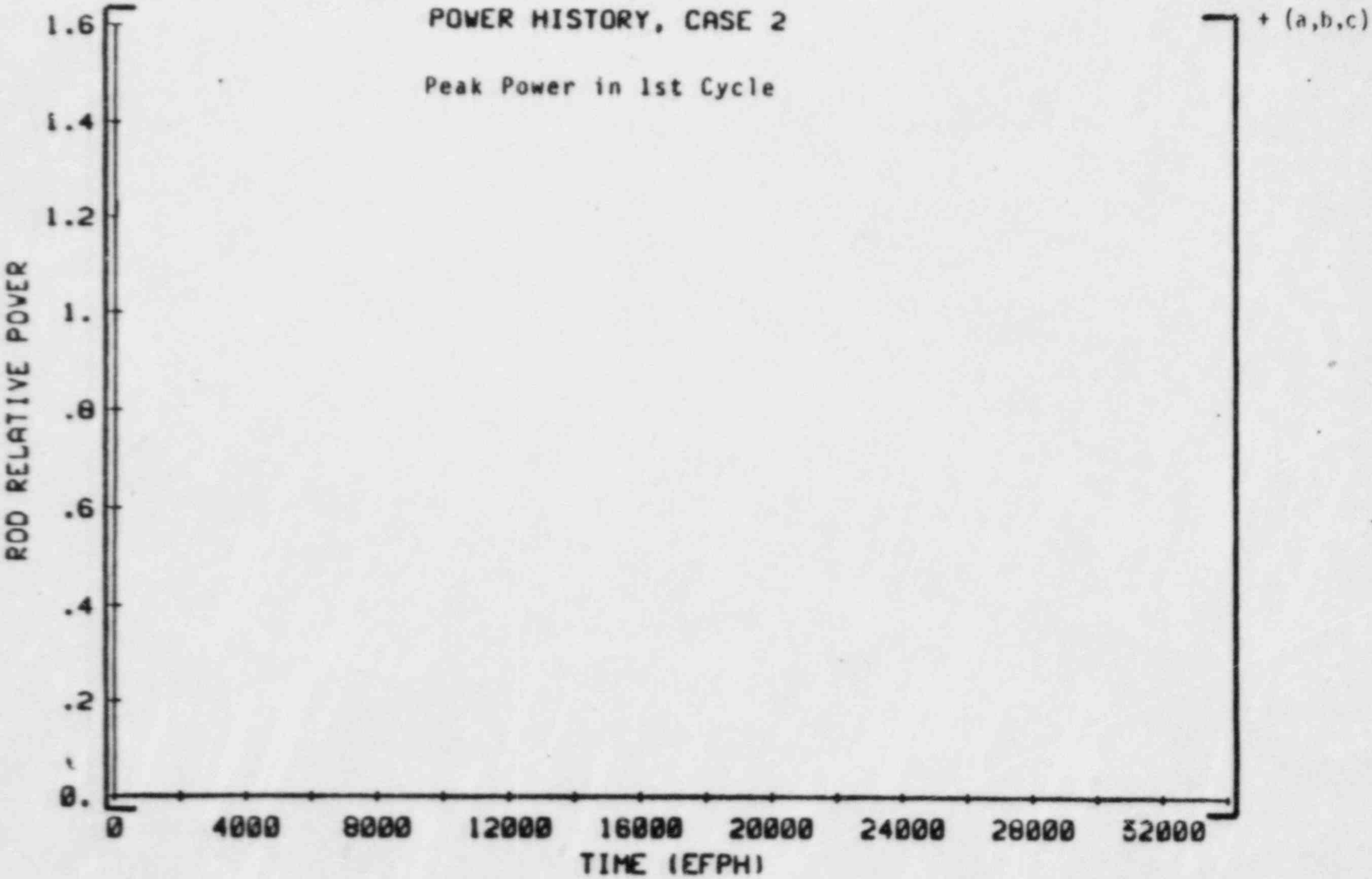

POWER HISTORY, CASE 2
Peak Power in 1st Cycle
+ (a,b,c)
ROD RELATIVE POWER
1.6
1.4
1.2
1.
.8
.6
.4
.2
0.
0
4000
8000
12000
16000
20000
24000
28000
32000
TIME (EFPH)

FIGURE 3

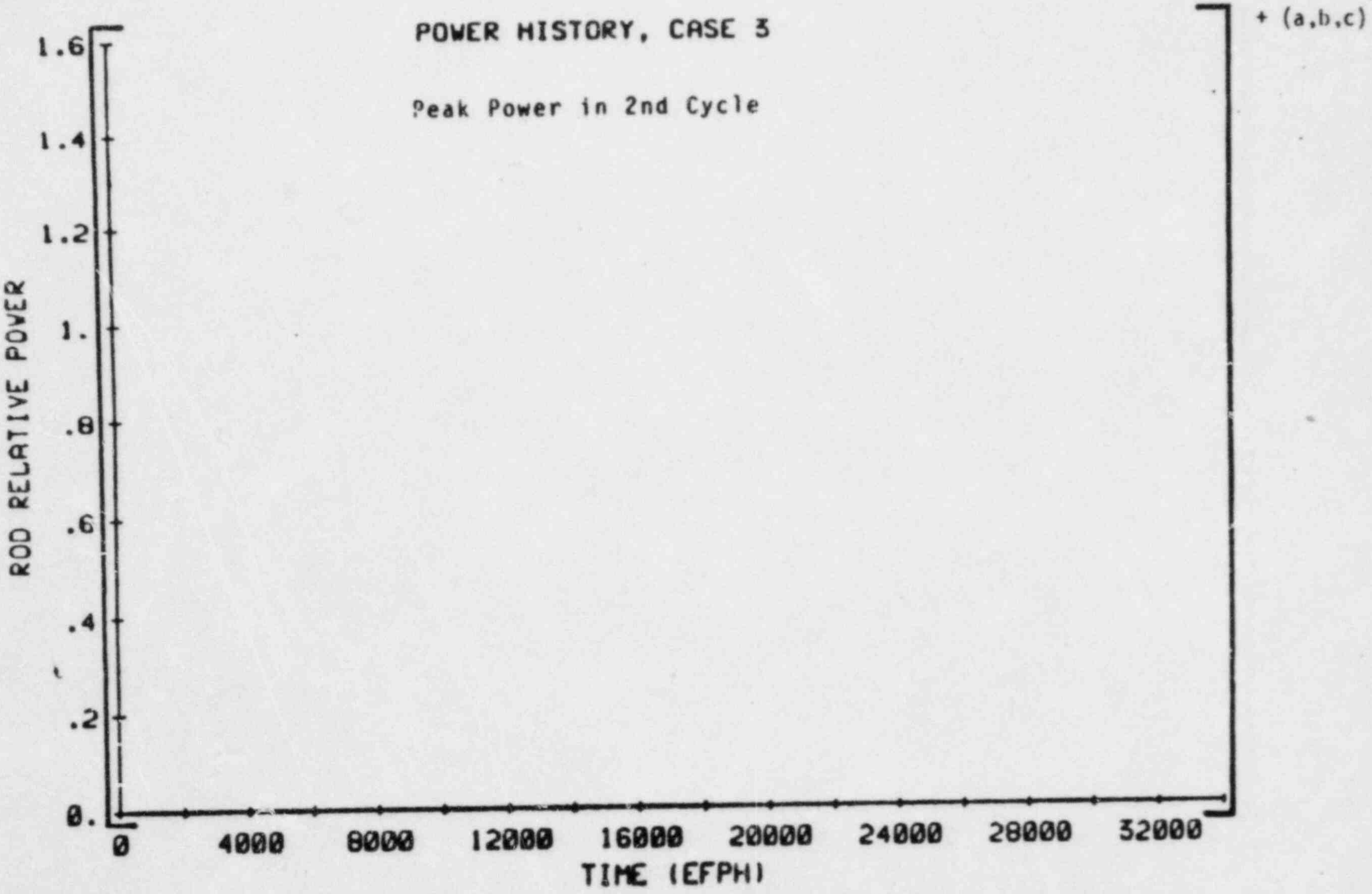
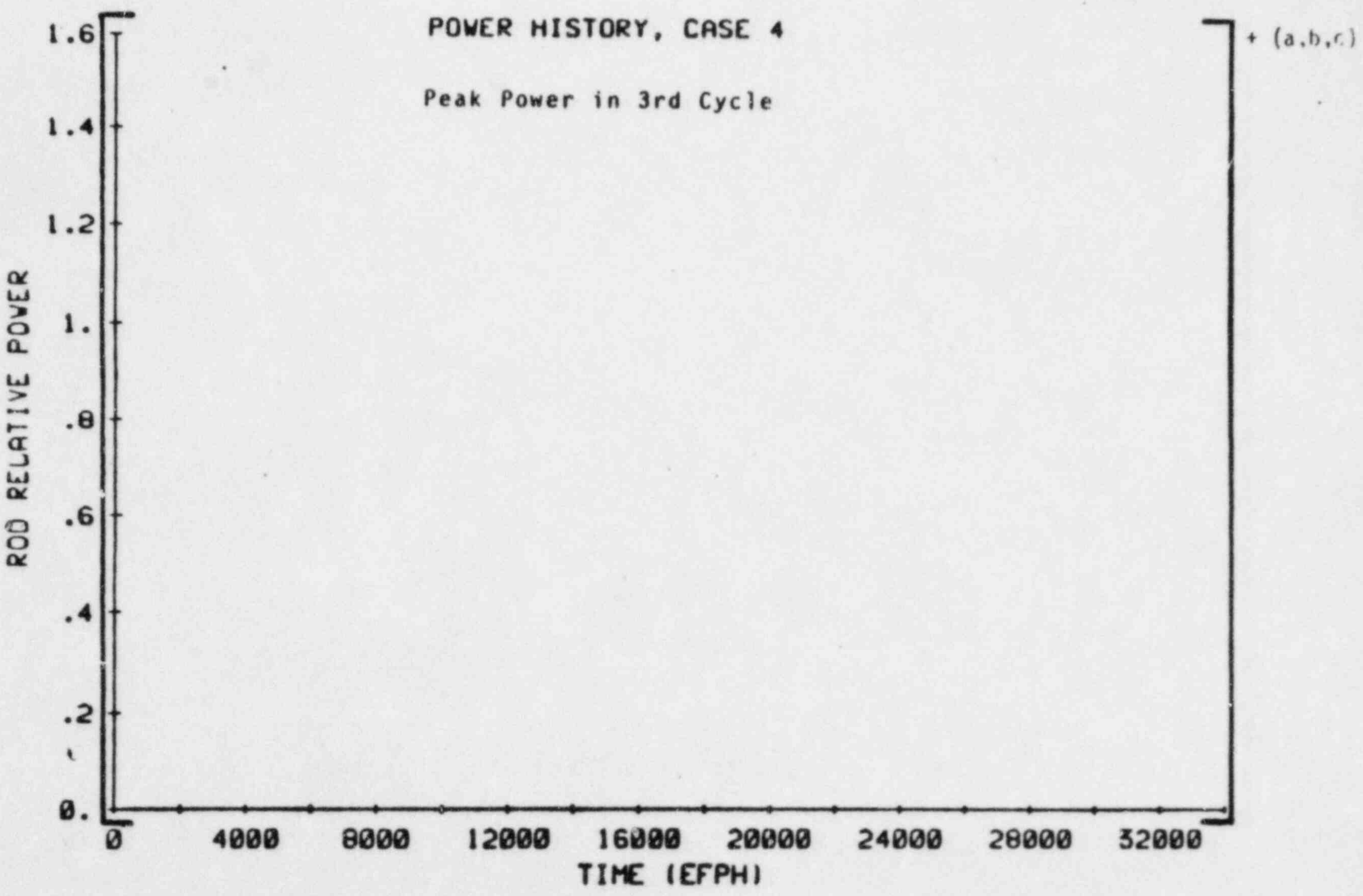

POWER HISTORY, CASE 3
Peak Power in 2nd Cycle
+ (a,b,c)
ROD RELATIVE POWER
1.6
1.4
1.2
1.
.8
.6
.4
.2
0.
0
4000
8000
12000
16000
20000
24000
28000
32000
TIME (EFPH)

FIGURE 4



POWER HISTORY, CASE 4
Peak Power in 3rd Cycle
+ (a,b,c)
ROD RELATIVE POWER
1.6
1.4
1.2
1.
.8
.6
.4
.2
0.
0
4000
8000
12000
16000
20000
24000
28000
32000
TIME (EFPH)

FIGURE 5
Rod Internal Pressure Versus Rod Average Burnup for Power History Case 1

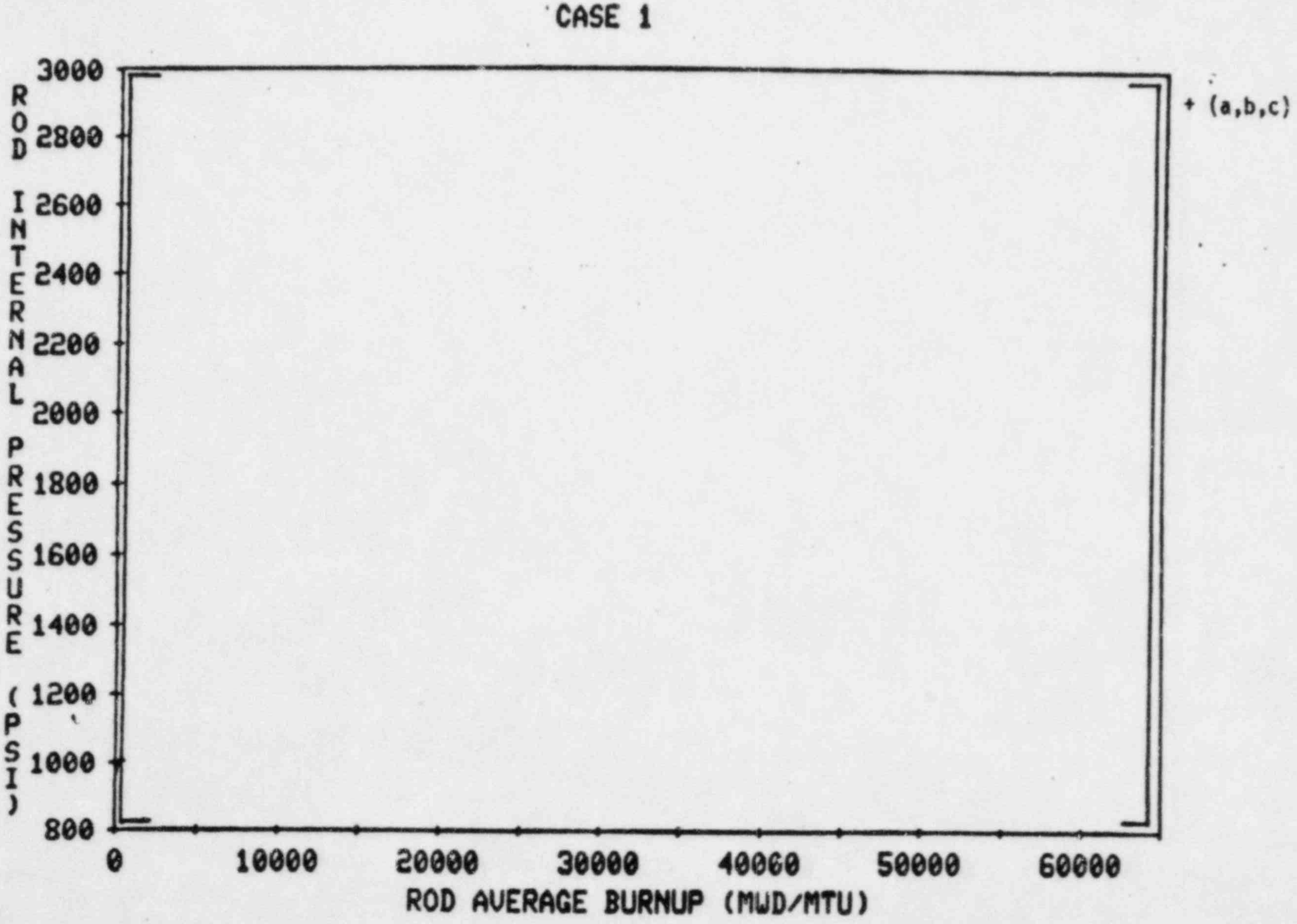


FIGURE 6

Rod Internal Pressure Versus Rod Average Burnup for Power History Case 2

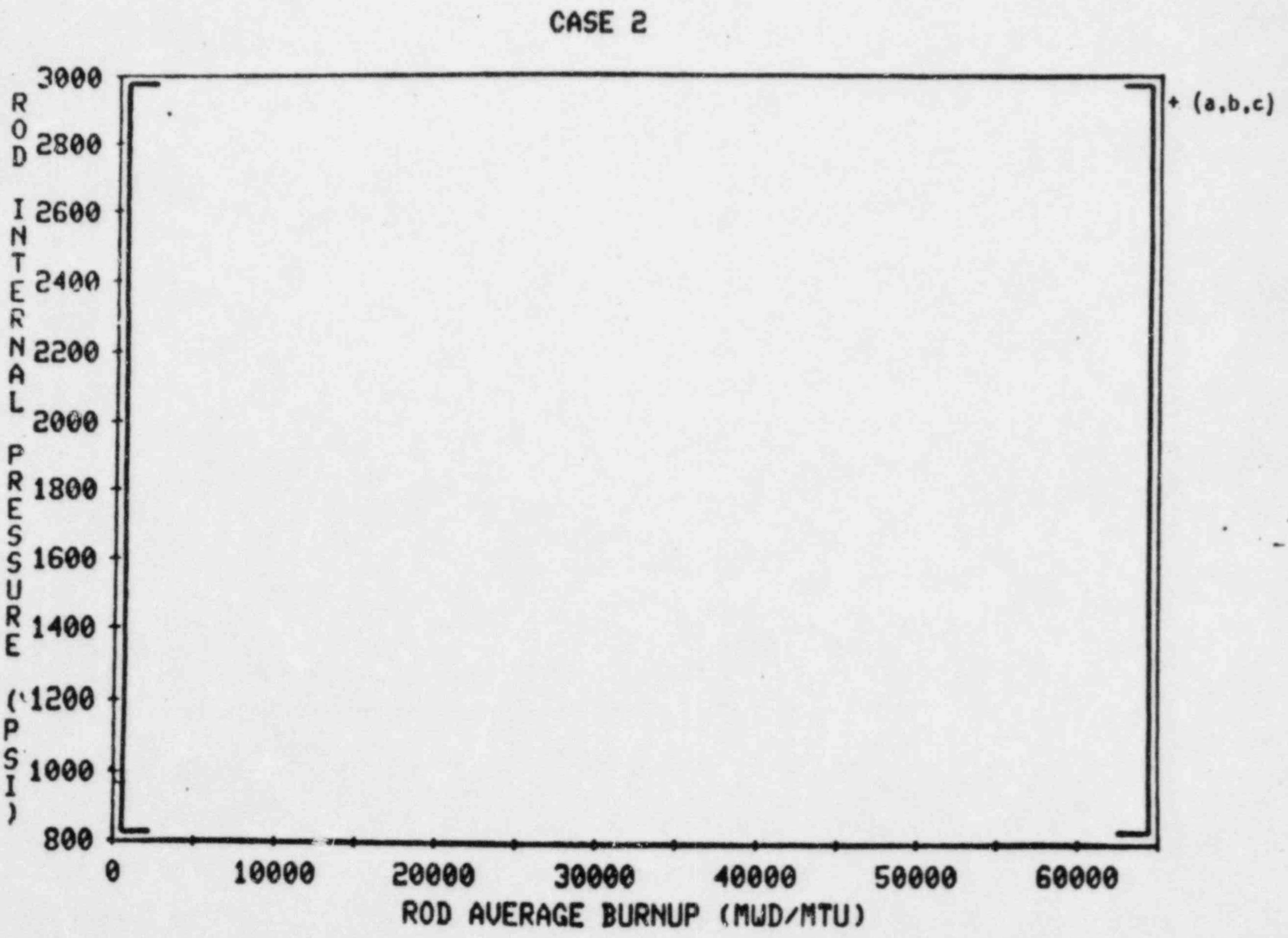


Figure 7

Rod Internal Pressure Versus Rod Average Burnup for Power History Case 3

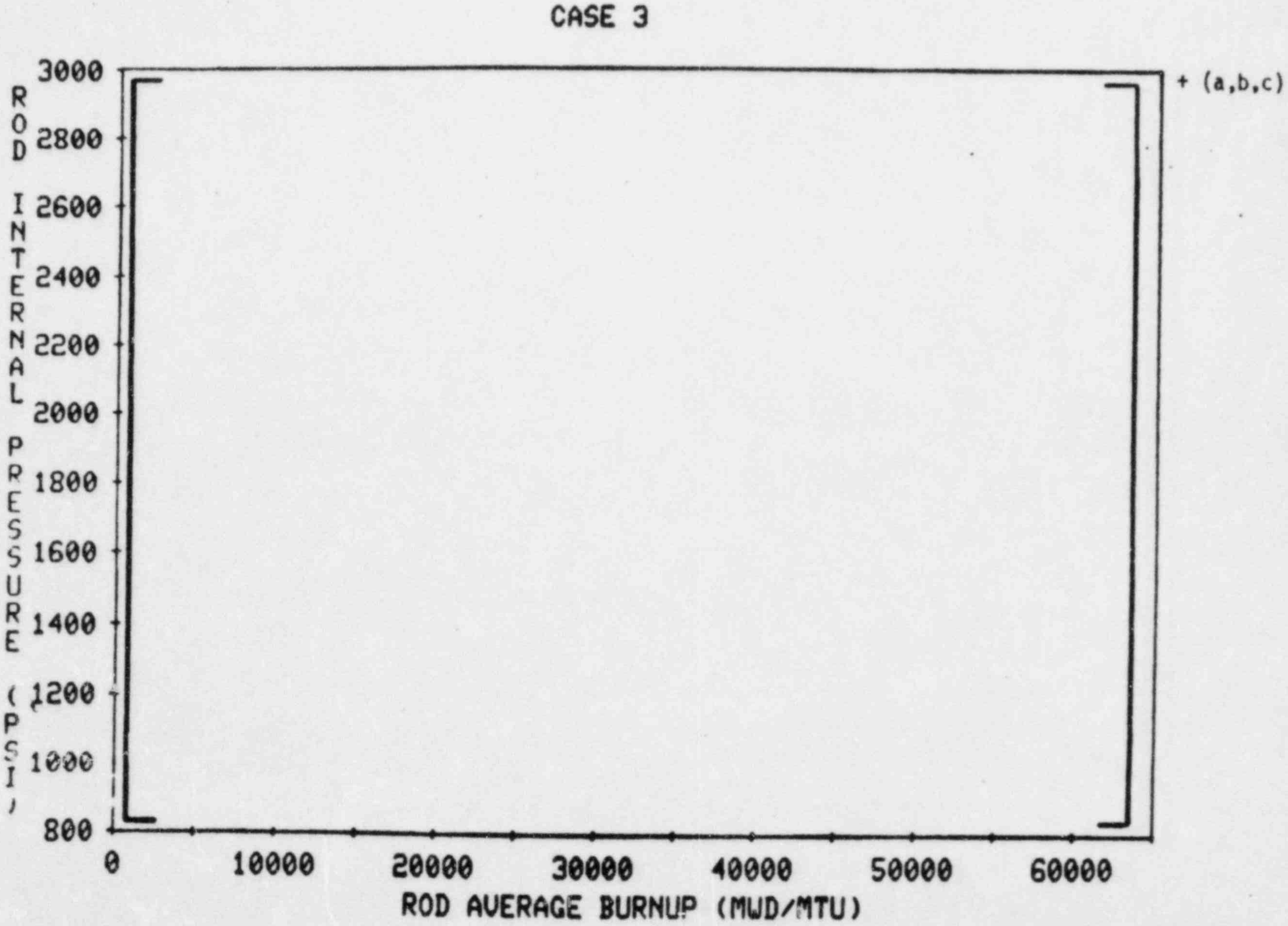


Figure 8

Rod Internal Pressure Versus Rod Average Burnup for Power Histor Case 4

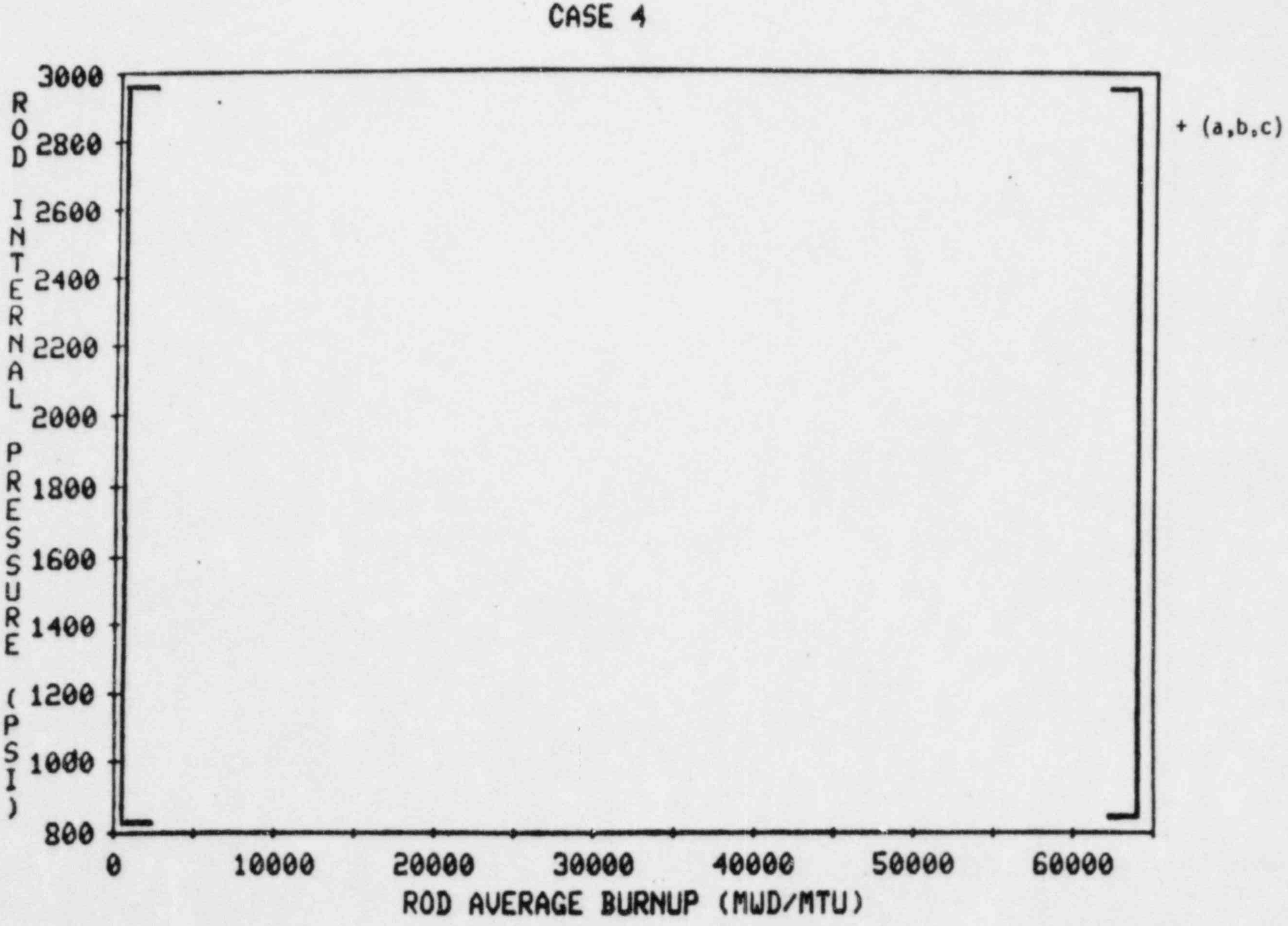


FIGURE 9

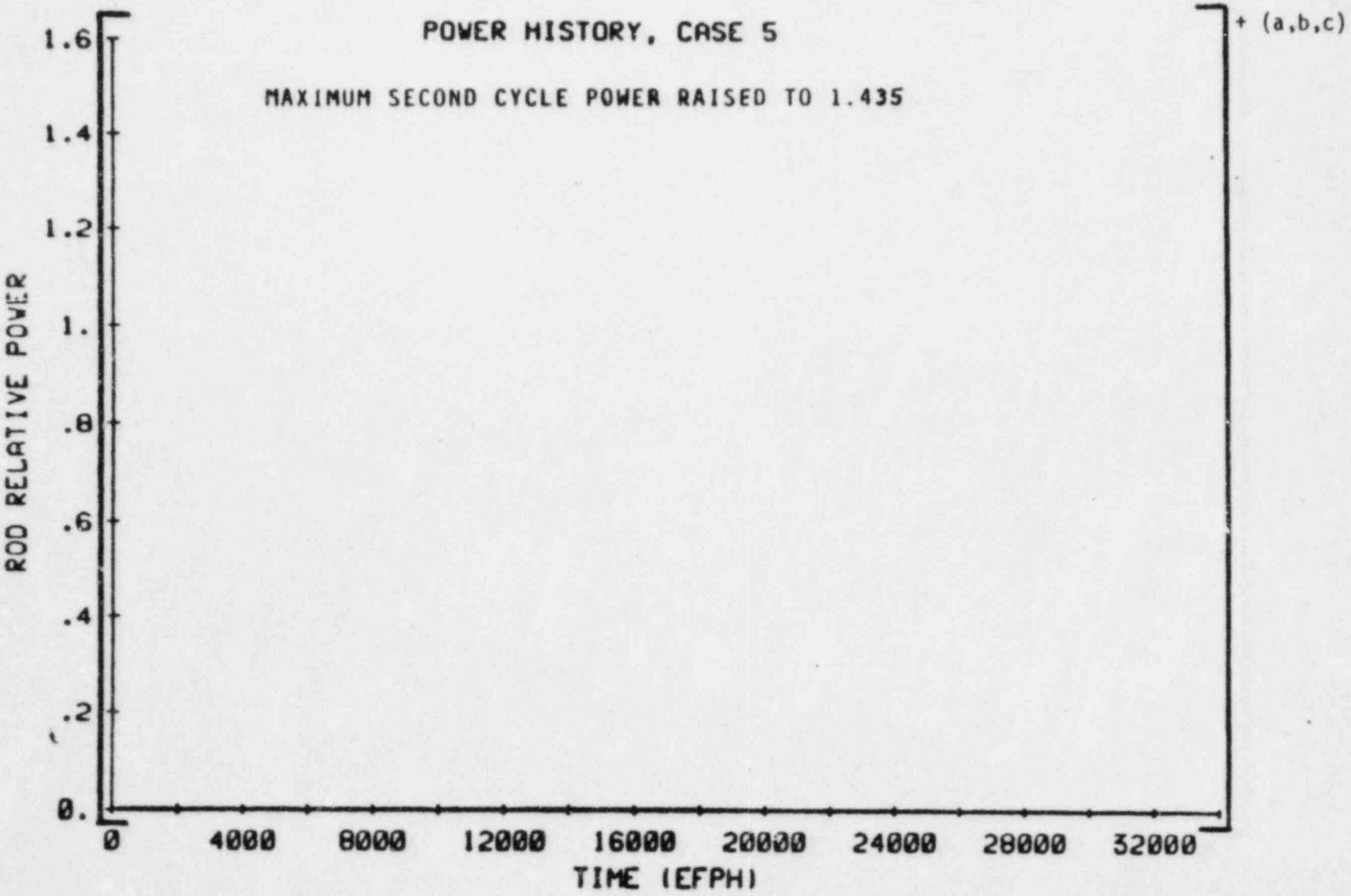
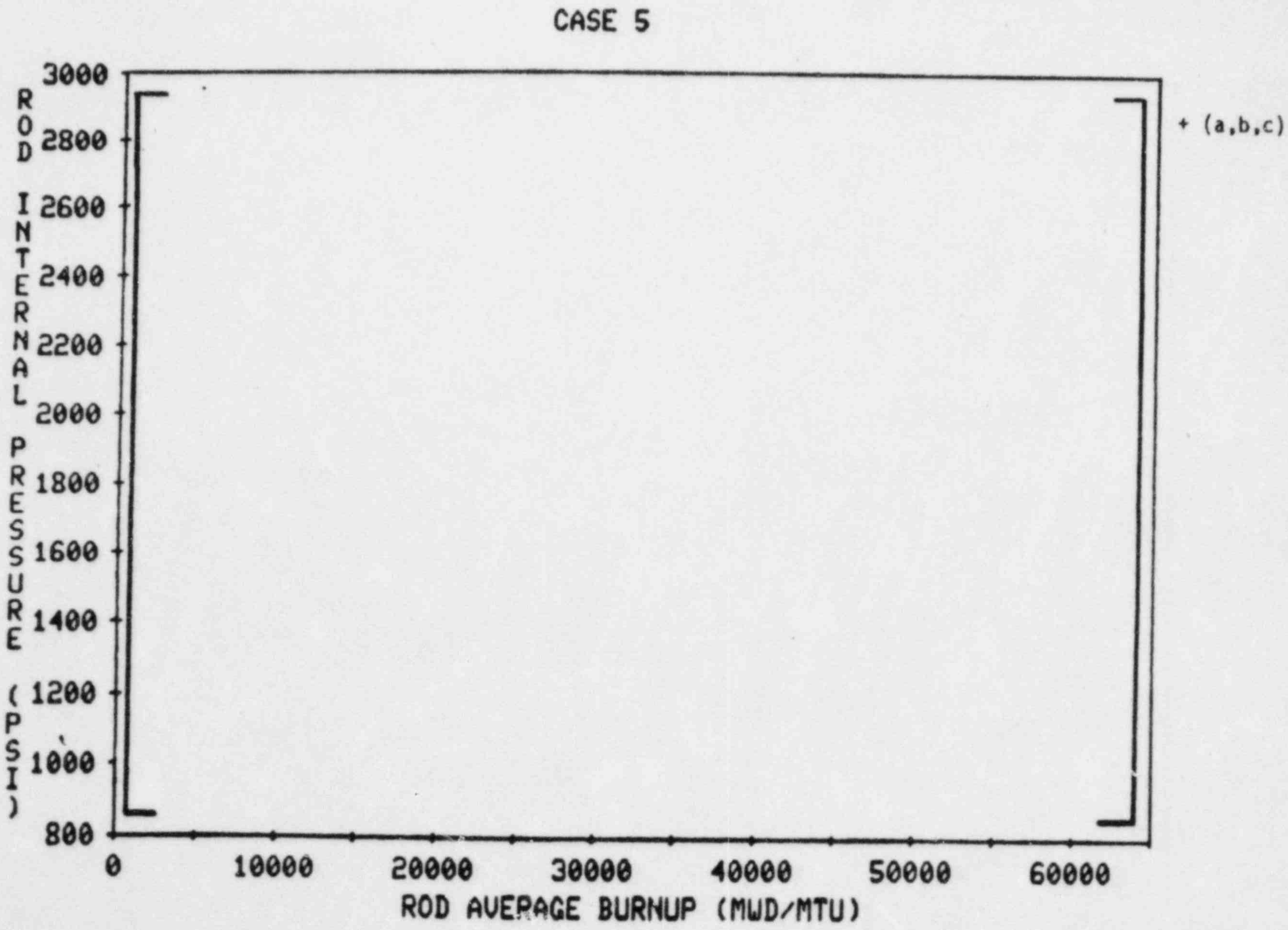

POWER HISTORY, CASE 5
MAXIMUM SECOND CYCLE POWER RAISED TO 1.435
+ (a,b,c)
ROD RELATIVE POWER
1.6
1.4
1.2
1.
.8
.6
.4
.2
0.
0
4000
8000
12000
16000
20000
24000
28000
32000
TIME (EFPH)

Figure 10

Rod Internal Pressure Versus Rod Average Burnup for Power History Case 5



+(a,

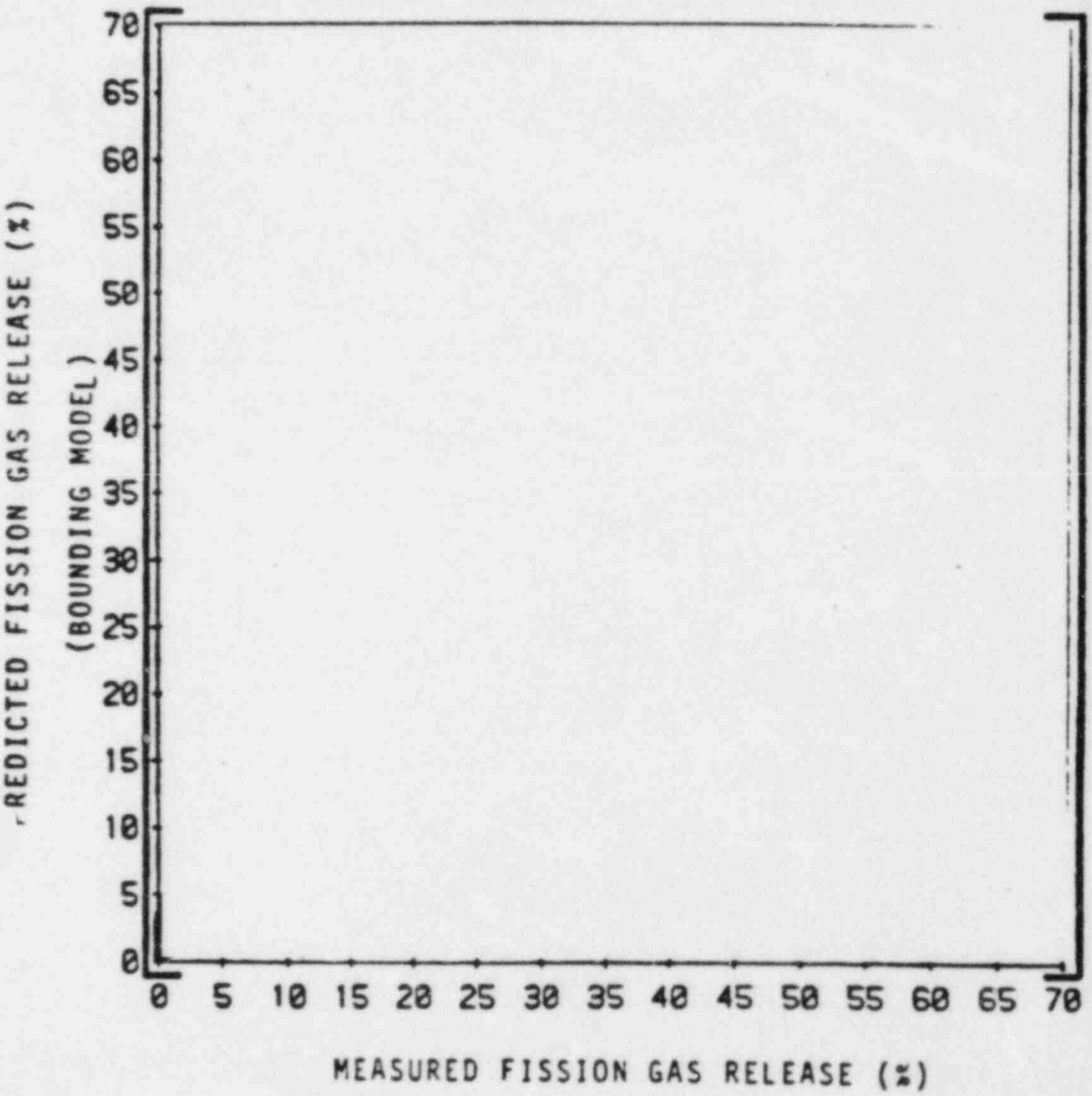


Fig. 11 Predicted vs. measured fission gas release, measured fission gas release All Data

+ (a,b,c)

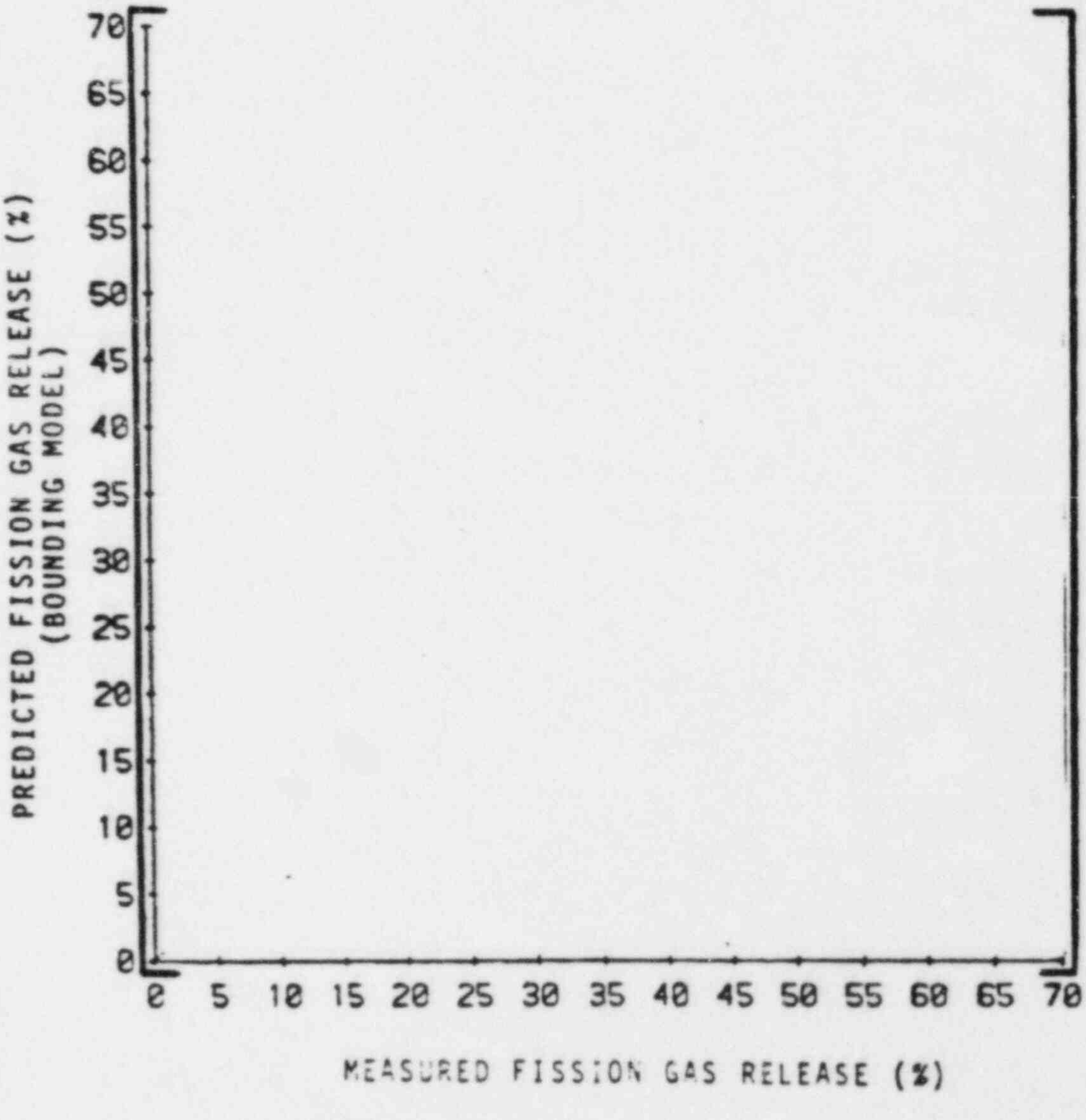


Fig. 12 Predicted vs. measured fission gas release, measured fission gas release $\geq$ 3%

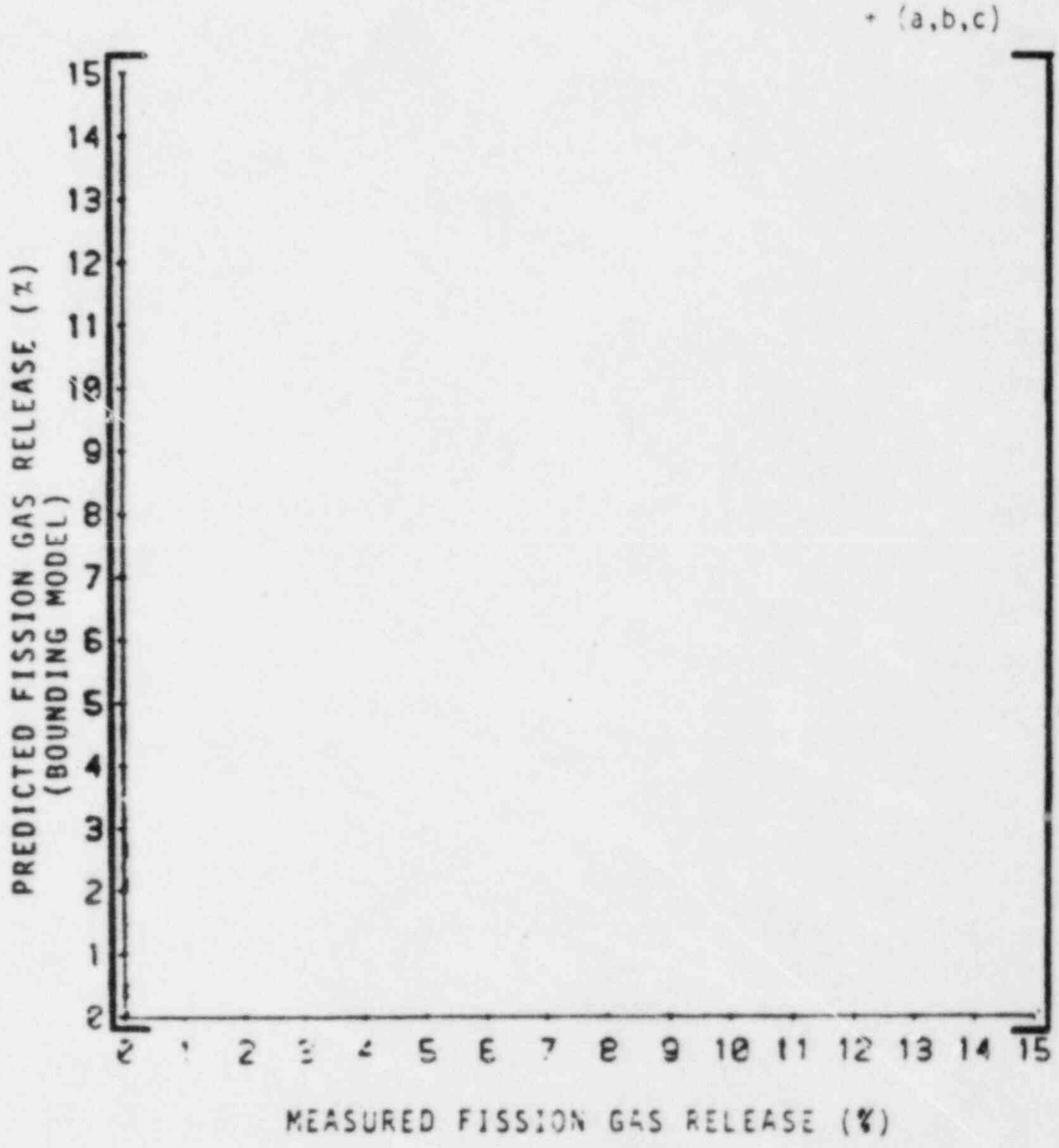


Fig. 13 Predicted vs. measured fission gas release, measured fission gas release < 3%

SECTION K

WESTINGHOUSE LETTER FROM E. P. RAHE,
SLIDES FOR TRANSIENT FISSION GAS RELEASE
AT EXTENDED BURNUP,
PRESENTATION TO NRC ON JANUARY 17, 1985 (PROPRIETARY),
NS-NRC-85-3003
DATED FEBRUARY 6, 1985
TO NRC, C. O. THOMAS

Westinghouse Electric Corporation
Water Reactor Divisions
Box 355
Pittsburgh Pennsylvania 15230

February 6, 1985
NS-NRC-85-3003

Dr. Cecil O. Thomas, Branch Chief
Standardization and Special Projects Branch
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: Slides for Transient Fission Gas Release at Extended Burnup Presentation to NRC on January 17, 1985 (Proprietary)

Dear Dr. Thomas:

Per your request, enclosed are:

1. Three (3) copies of slides for Transient Fission Gas Release at Extended Burnup Presentation to the NRC at Pittsburgh on January 17, 1985 (Proprietary).

2. Three (3) copies of slides for Transient Fisison Gas Release at Extended Burnup Presentation to the NRC at Pittsburgh on January 17, 1985 (Non-Proprietary).

3. One (1) copy of an Application for Withholding, AW-85-011 (Non-Proprietary).

4. One (1) copy of an original Affidavit, AW-76-21 (Non-Proprietary).

This submittal contains proprietary information of Westinghouse Electric Corporation. In conformance with the requirements of 10CFR2.790, as amended, of the Commission's regulations, we are enclosing with this submittal an application for withholding proprietary information from public disclosure and an affidavit. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission.

Correspondence with respect to the affidavit or application for withholding should reference AW-85-011 and should be addressed to R. A. Wiesemann, Manager of Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230.

Very truly yours,

E. P. Rahe, Jr., Manager
Nuclear Safety Department

cc: M. Dunenfeld
R. Lobel

GWH/kk
Attachments

Westinghouse Electric Corporation

Water Reactor Divisions

Box 355
Pittsburgh Pennsylvania 15230

February 6, 1985
AW-85-011

Dr. Cecil O. Thomas, Chief
Standardization and Special Projects Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Slides for Transient Fission Gas Release at Extended Burnup Presentation to NRC on January 17, 1985 (Proprietary)

Ref.: Westinghouse Letter No. NS-NRC-85-3003, Rahe to Thomas, dated February 6, 1985

Dear Dr. Thomas:

This application for withholding is submitted by Westinghouse Electric Corporation ("Westinghouse") pursuant to the provisions of paragraph (b) (1) of Section 2.790 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is of the same technical type as that proprietary material previously submitted with affidavit, AW-76-21, signed by the owner of the proprietary information, Westinghouse Electric Corporation.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-85-011 and should be addressed to the undersigned.

Very truly yours,

Robert A. Wiesemann

Robert A. Wiesemann, Manager
Regulatory & Legislative Affairs

/kk
cc: E. C. Shomaker, Esq.
Office of the Executive Legal Director, NRC

PROPRIETARY INFORMATION NOTICE

TRANSMITTED HEREWITH ARE PROPRIETARY AND/OR NON-PROPRIETARY VERSIONS OF DOCUMENTS FURNISHED TO THE NRC IN CONNECTION WITH REQUESTS FOR GENERIC AND/OR PLANT SPECIFIC REVIEW AND APPROVAL.

IN ORDER TO CONFORM TO THE REQUIREMENTS OF 10CFR2.790 OF THE COMMISSION'S REGULATIONS CONCERNING THE PROTECTION OF PROPRIETARY INFORMATION SO SUBMITTED TO THE NRC, THE INFORMATION WHICH IS PROPRIETARY IN THE PROPRIETARY VERSIONS IS CONTAINED WITHIN BRACKETS AND WHERE THE PROPRIETARY INFORMATION HAS BEEN DELETED IN THE NON-PROPRIETARY VERSIONS ONLY THE BRACKETS REMAIN, THE INFORMATION THAT WAS CONTAINED WITHIN THE BRACKETS IN THE PROPRIETARY VERSIONS HAVING BEEN DELETED. THE JUSTIFICATION FOR CLAIMING THE INFORMATION SO DESIGNATED AS PROPRIETARY IS INDICATED IN BOTH VERSIONS BY MEANS OF LOWER CASE LETTERS (a) THROUGH (g) CONTAINED WITHIN PARENTHESES LOCATED AS A SUPERSCRIPT IMMEDIATELY FOLLOWING THE BRACKETS ENCLOSING EACH ITEM OF INFORMATION BEING IDENTIFIED AS PROPRIETARY OR IN THE MARGIN OPPOSITE SUCH INFORMATION. THESE LOWER CASE LETTERS REFER TO THE TYPES OF INFORMATION WESTINGHOUSE CUSTOMARILY HOLDS IN CONFIDENCE IDENTIFIED IN SECTIONS (4)(ii)(a) through (4)(ii)(g) OF THE AFFIDAVIT ACCOMPANYING THIS TRANSMITTAL PURSUANT TO 10CFR2.790(b)(1).

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Robert A. Wiesemann, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Corporation ("Westinghouse") and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Robert A. Wiesemann

Robert A. Wiesemann, Manager
Licensing Programs

Sworn to and subscribed before me this ___ day of ______ 1976.

Notary Public

(1) I am Manager, Licensing Programs, in the Pressurized Water Reactor Systems Division, of Westinghouse Electric Corporation and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing or rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Water Reactor Divisions.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse Nuclear Energy Systems in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(g) It is not the property of Westinghouse, but must be treated as proprietary by Westinghouse according to agreements with the owner.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition in those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information is not available in public sources to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in the attachment to Westinghouse letter number NS-CE-1100, Eicheldinger to Vassallo, dated June 11, 1976, concerning THINC II/THINC IV code comparisons. The letter and attachment are being submitted in response to the NRC request at the May 28, 1976 NRC/Westinghouse meeting.

This information enables Westinghouse to:

(a) Justify the Westinghouse design correlations.

(b) Assist its customers to obtain licenses.

(c) Obtain preliminary design approvals.

(d) Meet warranties.

(e) Provide greater flexibility to customers assuring them of safe and reliable operation.

(f) Reduce plant and fuel costs.

(g) Optimize performance while maintaining high level of fuel integrity.

Further, the information gained from the THINC IV development program is of commercial value and is sold for considerable sums of money as follows:

(a) Westinghouse sells the use of this information to foreign licensees.

(b) Westinghouse uses the information to perform and justify analyses which are sold to customers.

(c) Westinghouse sells testing services based upon the experience gained and the test equipment and methods developed.

Public disclosure of this information concerning THINC II/ THINC IV code comparisons is likely to cause substantial harm to the competitive position of Westinghouse because competitors could utilize this information to assess and justify their own designs without commensurate expense.

The comparisons performed and their evaluation represent a considerable amount of highly qualified development effort. This work was contingent upon a THINC IV development program which has been underway during the past six years. Altogether, a substantial amount of money and effort has been expended

by Westinghouse which could only be duplicated by a competitor if he were to invest similar sums of money and provided he had the appropriate talent available.

Further the deponent sayeth not.

WESTINGHOUSE/NRC MEETING

ON THE

WESTINGHOUSE

EXTENDED BURNUP

TOPICAL REPORT

WCAP-10125

JANUARY 17, 1985

NON-PROPRIETARY VERSION

AGENDA

- W EXTENDED BURNUP TOPICAL OBJECTIVES
- W TRANSIENT FISSION GAS RELEASE METHODOLOGY
  - ROD INTERNAL PRESSURE DESIGN CRITERIA
  - ROD INTERNAL PRESSURE DESIGN METHODOLOGY
  - TRANSIENT POWER DUTY
  - TRANSIENT FISSION GAS RELEASE MODEL
  - TRANSIENT FISSION GAS RELEASE IMPACT ON ROD INTERNAL PRESSURE
- CONCLUSIONS

W EXTENDED BURNUP TOPICAL REPORT

OBJECTIVE

- CONFIRM NRC APPROVED FUEL DESIGN/ SAFETY ANALYSIS APPROACH APPLICABLE FOR EXTENDED BURNUP
  - METHODS
  - MODELS
  - CRITERIA

NRC APPROVED W TRANSIENT FISSION GAS RELEASE ANALYSIS

ROD INTERNAL PRESSURE DESIGN CRITERIA

- PRESSURE LIMITED TO PRECLUDE
  - FUEL/CLAD GAP INCREASE DUE TO OUTWARD CLAD CREEP
  - EXTENSIVE DNB PROPAGATION
- PERMITS INTERNAL PRESSURE TO EXCEED SYSTEM PRESSURE

NRC APPROVED W TRANSIENT FISSION GAS RELEASE ANALYSIS

W ROD INTERNAL PRESSURE DESIGN METHODOLOGY

+(a,c)

- [ ]

- PAD3.3 CODE USED IN DESIGN ANALYSIS

- PAD3.3 APPLICABILITY FOR CONDITION. II TRANSIENT ANALYSIS IS ADDRESSED IN WCAP-8720, ADDENDUM 1

W TRANSIENT FISSION GAS RELEASE METHODOLOGY

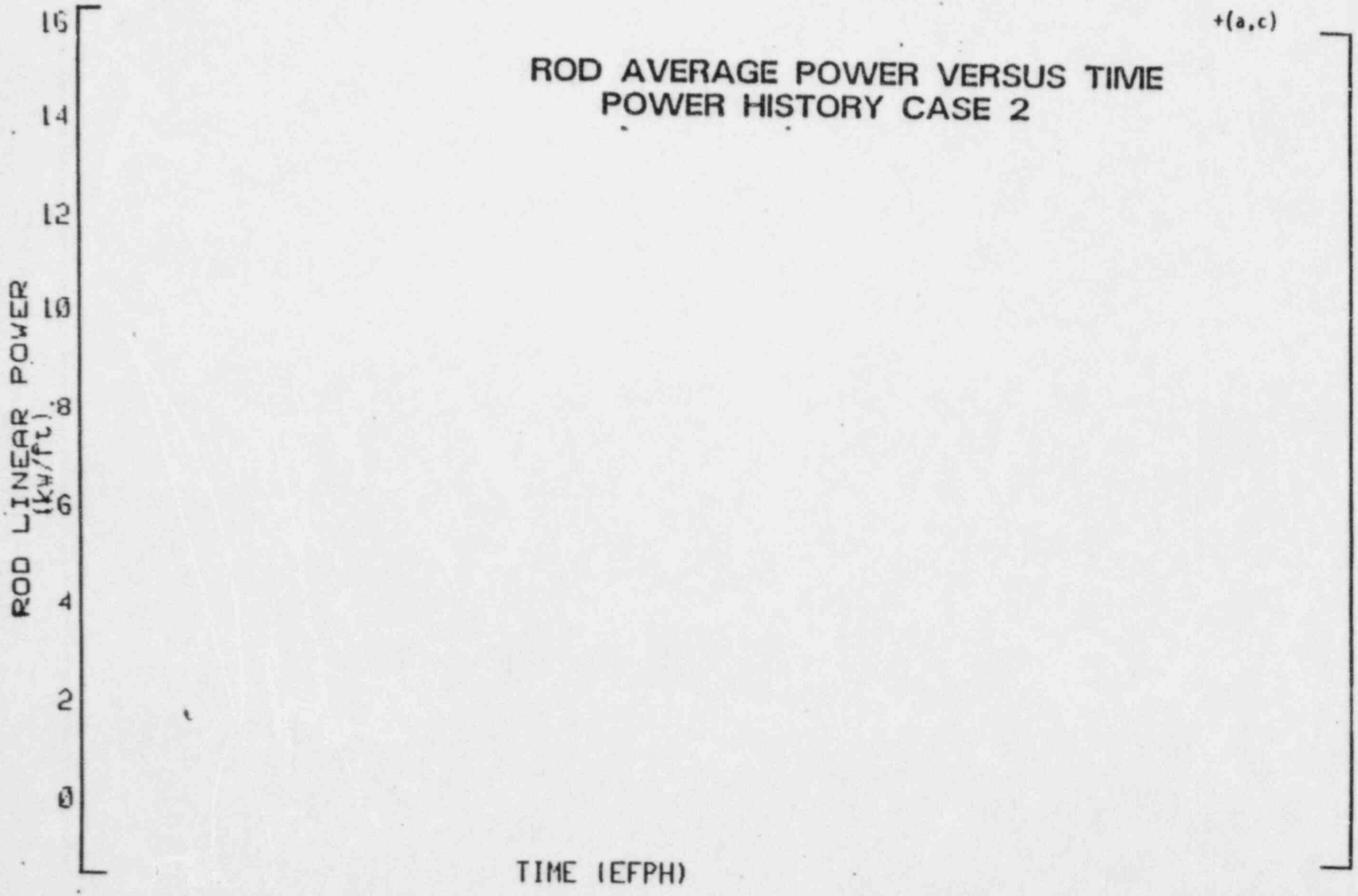
TRANSIENT POWER DUTY

- ANSI CONDITION II TRANSIENTS
  - TYPICALLY SHORT - LESS THAN 15 MINUTES
- MAXIMUM LOCAL POWER INCREASE ATTAINABLE AT EXTENDED BURNUP IS LIMITED BY
  - DEPLETION OF FISSIONABLE ISOTOPES
  - FISSION PRODUCT BUILDUP

W TRANSIENT FISSION GAS RELEASE METHODOLOGY

TRANSIENT POWER LIMITS

- PEAK LOCAL POWER DURING A CONDITION II TRANSIENT IS A FUNCTION OF
  - INITIAL POWER
  - BURNUP
- MAXIMUM TRANSIENT POWERS OCCUR EARLY IN LIFE


+(a,c)
ROD AVERAGE POWER VERSUS TIME
POWER HISTORY CASE 2
ROD LINEAR POWER (kw/ft)
16
14
12
10
8
6
4
2
0
TIME (EFPH)

W TRANSIENT FISSION GAS RELEASE METHODOLOGY

WCAP-8720, ADDENDUM 1 REVIEW

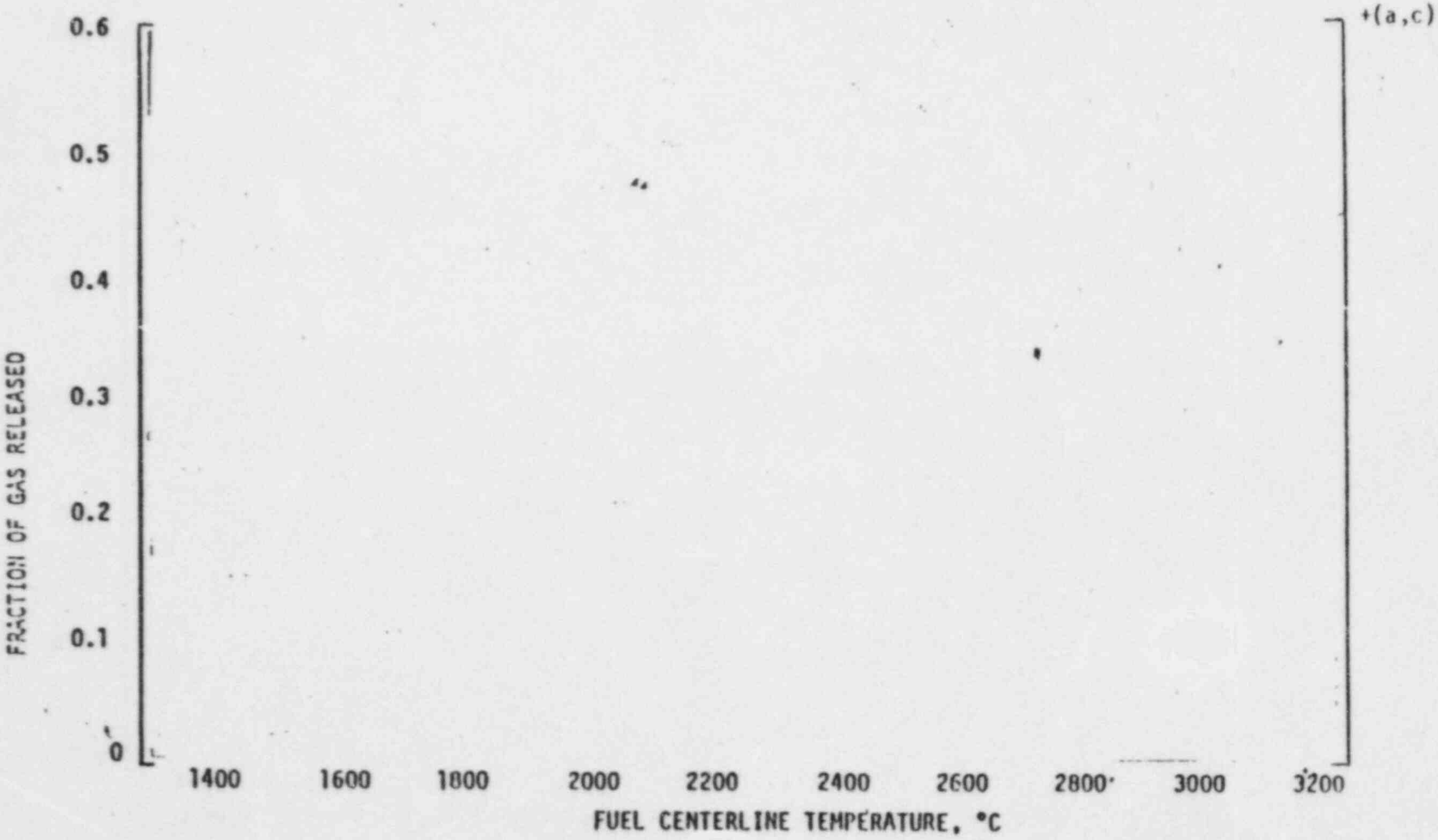
- PAD3.3 MODELS VALID FOR ANALYSIS OF CONDITION II TRANSIENTS
  - EXCEPT FISSION GAS RELEASE MODEL
  - MAY UNDERPREDICT TRANSIENT FISSION GAS RELEASE FOR HIGH TEMPERATURE TRANSIENTS
- TRANSIENT FISSION GAS RELEASE MODELLED IN PAD3.3
  - ANSI CONDITION II TRANSIENTS
  - 50 HOUR TIME STEP ASSUMED AT PEAK TRANSIENT POWER LEVEL

W TRANSIENT FISSION GAS RELEASE METHODOLOGY

50 HOUR HOLD TIME METHODOLOGY

- VALIDATED FOR SHORT TERM ANSI CONDITION II TRANSIENTS IN PWR CORES
- BASED ON COMPARISON AGAINST AVAILABLE SHORT TERM TRANSIENT FISSION GAS RELEASE DATA
  - W OVERRAMP RODS
  - RISO RAMP TEST RODS
  - SAXTON II PU RODS
  - ARGONNE DIRECT ELECTRICAL HEATING TESTS

TRANSIENT FISSION GAS RELEASE VERSUS PEAK CENTERLINE TEMPERATURE

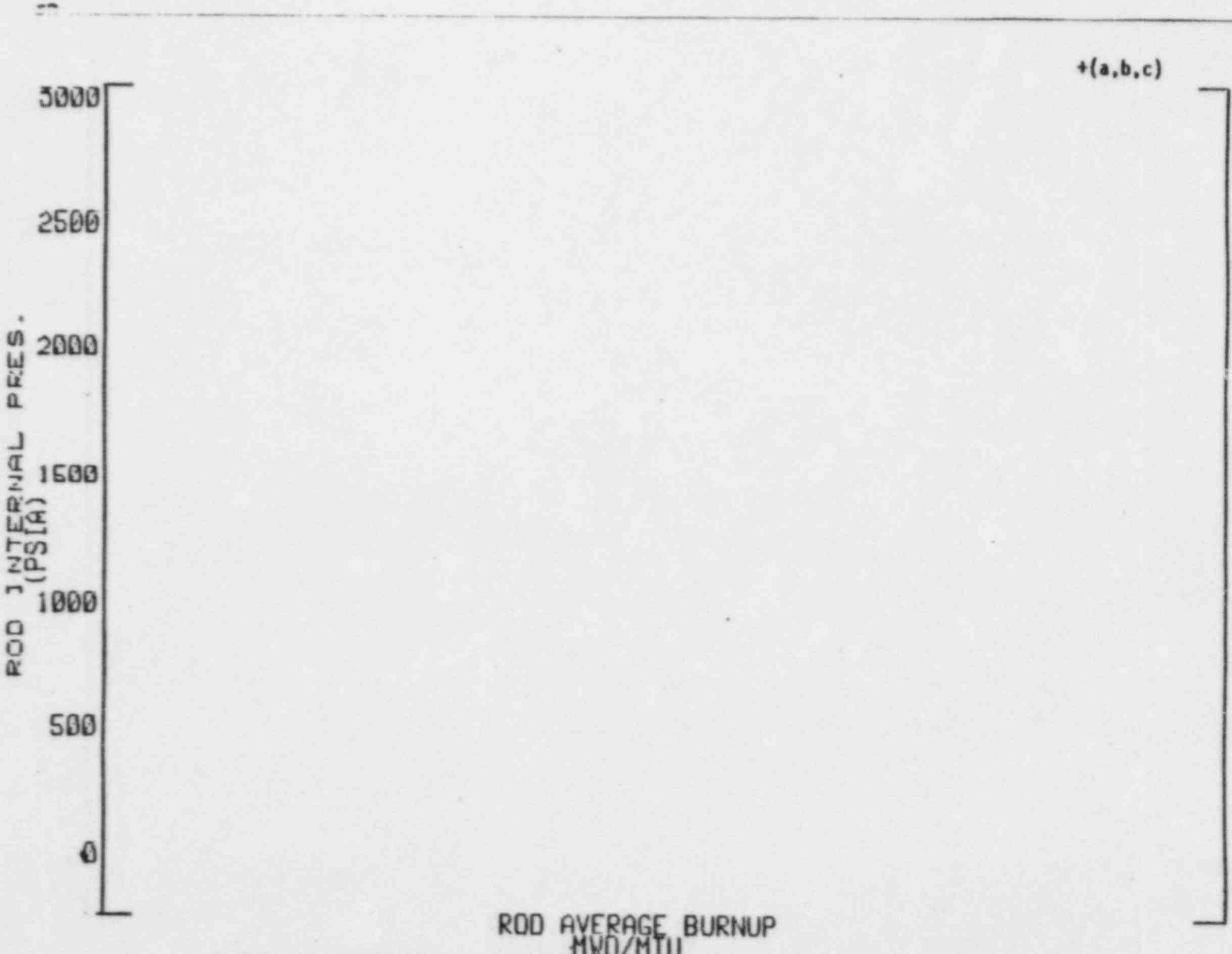


TRANSIENT GAS RELEASE MODELLING

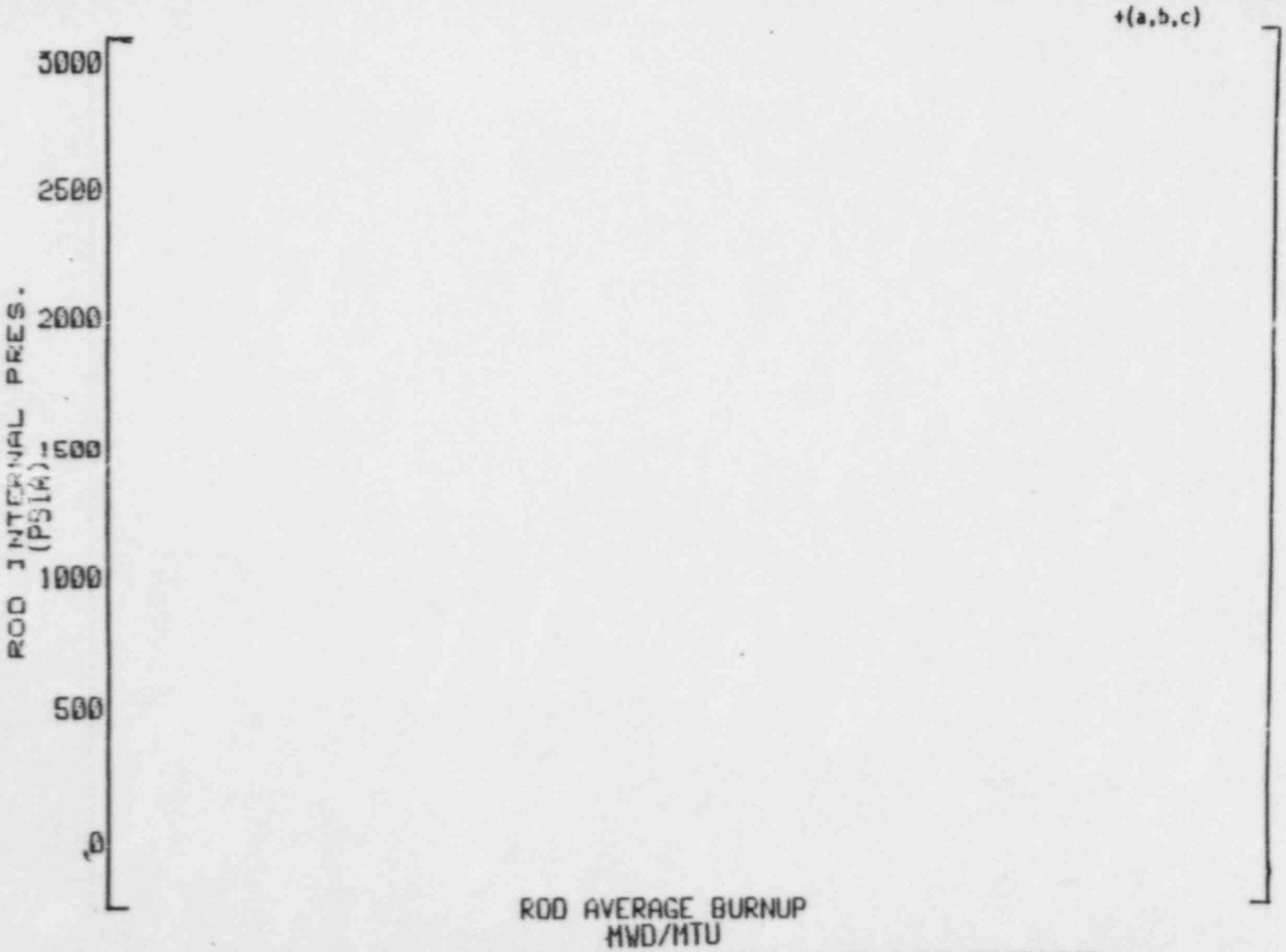
- DATA INDICATE LOW TRANSIENT FISSION GAS RELEASE FOR
  - SHORT HOLD TIMES - NEAR 15 MINUTE
  - PEAK CENTERLINE TEMPERATURES OF 1330 TO 1570 DEGREES C
- 50 HOUR HOLD TIME MODEL IS A REASONABLE APPROXIMATION OF THE CONDITION II TRANSIENT FISSION GAS RELEASE

TRANSIENT FISSION GAS RELEASE IMPACT ON ROD INTERNAL PRESSURE

- RESULTS OF 50 HOUR HOLD TIME MODEL
  - WORST CASE TRANSIENT AT END OF CYCLE 3
  - WORST CASE TRANSIENT AT END OF CYCLE 1
  - WORST CASE TRANSIENTS AT END OF CYCLES 1, 2, AND 3



ROD INTERNAL PRESSURE VS ROD AVERAGE BURNUP
POWER HISTORY CASE 2
WORST CASE CONDITION II TRANSIENTS AT
EOC1, ECC2, AND EOC3



ROD INTERNAL PRESSURE VS ROD AVERAGE BURNUP
POWER HISTORY CASE 2
SINGLE WORST CASE CONDITION II TRANSIENT AT EOC3

ROD INTERNAL PRESSURE VS ROD AVERAGE BURNUP
POWER HISTORY CASE 2
SINGLE WORST CASE CONDITION II TRANSIENT AT EOC1

CONCLUSIONS

- TRANSIENT ROD DUTY AT EXTENDED BURNUP LIMITED BY FUEL DEPLETION AND FISSION PRODUCT BUILDUP

- NRC APPROVED WESTINGHOUSE CONDITION II TRANSIENT METHODOLOGY IS VALID FOR EXTENDED BURNUP

- CONDITION II TRANSIENT FISSION GAS RELEASE DOES NOT SIGNIFICANTLY IMPACT EOL ROD INTERNAL PRESSURE

SECTION L

NRC LETTER FROM C. O. THOMAS,
REQUEST NUMBER THREE FOR ADDITIONAL INFORMATION
ON WCAP-10125 (P),
DATED FEBRUARY 20, 1985
TO WESTINGHOUSE, E. P. RAHE

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

February 20, 1985

Mr. E. P. Rahe, Jr., Manager
Nuclear Safety Department
Westinghouse Electric Corporation
Box 355
Pittsburgh, Pennsylvania 15230

Dear Mr. Rahe:

SUBJECT: REQUEST NUMBER THREE FOR ADDITIONAL INFORMATION ON WCAP-10125(P)

We are currently reviewing the Westinghouse Licensing Topical Report, WCAP-10125(P) entitled "Extended-Burnup Evaluation of Westinghouse Fuel".

The initial review reveals the need for the additional information indicated in the enclosure. In order to complete this review within the currently scheduled time, responses to all questions should be received by NRC by March 15, 1985. Please advise D. H. Moran at (301) 492-9409 if you cannot meet this date.

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OBM clearance is not required under P.L. 96-511.

Sincerely,

Cecil O. Thomas

Cecil O. Thomas, Chief
Standardization & Special
Projects Branch
Division of Licensing

Enclosure:
As stated

ENCLOSURE

According to WCAP 10125, Westinghouse is proposing to apply a new criterion for axial fuel rod irradiation growth. It is not clear from the report exactly what the new criterion is or how it is to be applied. Therefore, please respond to the following questions:

(1) State the new Westinghouse criterion for rod-to-nozzle gap closure.

(2)(a) Describe how this criterion is to be applied in safety and design analysis.

(b) Is the proposed bound a confidence limit on the mean of future observations or is it a tolerance limit for the population of future observations of gap size.

What uncertainties are explicitly considered in the statistical evaluation of gap size, e.g., fabrication, the irradiation growth model (both rod and guide thimbles), fluence, guide thimble loads and other uncertainties that could affect gap closure? Could these uncertainties distort either the tolerance limit or the confidence limit? What is the probability of any individual rod within a typical fuel batch with a lead rod average burnup of 60 Mwd/kgM experiencing gap closure?

(c) Axial gap closure is known to be a function of several fuel design parameters including fuel pellet geometry, as built cold gap, hold-down spring force, cladding heat treatment, etc.. Describe how the differences in these fuel design variables are taken into account in applying a statistical criterion to gap closure.

(d) How is the difference between dimensions during operation when the fuel is hot and the measurement of the gap when the fuel is cold taken into account?

3. Provide a description of the evidence which supports the conclusion that potential contact of the fuel rod to the nozzle will not have deleterious effects in terms of:

   (a) increasing the incidence of rods bowed to contact

   (b) potentially introducing a new cause of fuel rod failure by mechanical failure due to compression of the fuel rod or by contact with adjacent rods

4. Will this new criterion be applied to VANTAGE-5 and QUAD + and other future Westinghouse fuel bundle designs?

5. State how the use of this proposed criterion for fuel rod-to-nozzle growth gap is in compliance with 10CFR50, Appendix A, GDC 10 with respect to exceeding specified acceptable fuel design limits during normal operation and anticipated operational occurrences.

SECTION M

WESTINGHOUSE LETTER FROM E. P. RAHE,
RESPONSE TO NRC REQUEST NUMBER 3
FOR ADDITIONAL INFORMATION
ON WCAP-10125 (PROPRIETARY), NS-NRC-3018,
DATED MARCH 14, 1985,
TO NRC, C. O. THOMAS

Westinghouse Electric Corporation
Water Reactor Divisions
Box 355
Pittsburgh Pennsylvania 15230-0355

NS-NRC-85-3018
March 14, 1985

Dr. Cecil O. Thomas, Chief
Standardization and Special Projects Branch
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: Response to NRC Request Number 3 for Additional Information on WCAP-10125 (Proprietary)

REFERENCE: Letter from C. O. Thomas (NRC) to E. P. Rahe, Jr. (W), dated February 20, 1985

Dear Dr. Thomas:

In accordance with the referenced letter, enclosed are:

1. One (1) copy of the response to the NRC request number 3 for additional information on WCAP-10125 (Proprietary).

2. One (1) coy of Application for Withholding, AW-85-024 (Non-Proprietary).

The enclosed response is submitted for your review and approval in compliance with obtaining a forthcoming SER on WCAP-10125.

Correspondence with respect to the affidavit or application for withholding should reference AW-85-024 and should be addressed to R. A. Wiesemann, Manager of Regulatory and Legislative Affairs, Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230.

Very truly yours,

E P Rahe Jr.

E. P. Rahe, Jr., Manager
Nuclear Safety Department

cc: R. Lobel

GWH/kk
Attachments

Westinghouse Electric Corporation

Water Reactor Divisions

Box 355
Pittsburgh Pennsylvania 15230 0355

March 14, 1985
AW-85-024

Dr. Cecil O. Thomas, Chief
Standardization and Special Projects Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Response to NRC Request Number 3 for Additional Information on WCAP-10125 (Proprietary)

Ref.: Westinghouse Letter No. NS-NRC-85-3018, Rahe to Thomas, dated March 14, 1985

Dear Dr. Thomas:

The enclosed material transmitted by the reference letter contains information proprietary to the Westinghouse Electric Corporation.

The material is not intended to be employed as part of a license application or other action identified in 10CFR2.790(a). It will be separately submitted with an Application for Withholding Proprietary Information from Public Disclosure accompanied by an Affidavit meeting the requirements of 10CFR2.790(b) prior to such use.

Accordingly, we request the material be treated as proprietary information within the provisions of 10CFR9.5(4), "Freedom of Information Act Regulations." If there is a need to make public disclosure of the material prior to a separate Westinghouse submittal for docket in accordance with the provisions of 10CFR2.790(a), please notify Westinghouse prior to making a disclosure determination.

Correspondence with respect to the proprietary aspects of this submittal should reference AW-85-024 and should be addressed to the undersigned.

Very truly yours,

Robert A. Wiesemann

R. A. Wiesemann, Manager
Regulatory and Legislative Affairs

/kk

cc: E. C. Shomaker, Esq.
Office of the Executive Legal Director, NRC

Response to NRC Request Number 3 for Additional Information on WCAP-10125 (Proprietary)

The current Westinghouse design basis for interactive fuel rod/assembly irradiation growth effects is that the fuel rods will be designed with adequate clearance between the fuel rod ends and the top and bottom nozzles to accommodate the differences in growth of the fuel rods and the skeleton structure. The calculated minimum gap which allows for differential growth between fuel rods and the fuel assembly has been established for Westinghouse fuel designs to be slightly greater than [ ]+. (a,c) This required allowance for irradiation growth is established to preclude fuel rod-to-nozzle interference during projected operation on the basis of the assumption of worst case fuel rod and fuel assembly growth combined with worst case fabrication tolerances. This methodology has been used for all Westinghouse fuel assembly designs including the Westinghouse Optimized Fuel Assembly (OFA) and Vantage-5 designs (References 1 and 2).

The planned revision of the Westinghouse design criterion for fuel rod growth gap which was described in WCAP-10125 recognizes the statistical nature of fuel rod and fuel assembly growth correlations, and replaces the criterion of no interference between the rods and the top and bottom nozzles with a statistical criterion. This revised criterion will assure with high confidence that only a very small fraction of the fuel rods in a region could be in contact with both the top and bottom nozzles at the end of design life. The design methodology to be used with the revised criterion will use a rigorous statistical convolution of the uncertainties associated with the fuel rod and fuel assembly growth relationships, the as-fabricated rod-to-nozzle gap distribution, and the expected fuel rod fluence distribution in the fuel region at the design discharge burnup.

The details of the proposed Westinghouse statistical fuel rod growth criterion and design methodology cannot be provided to the NRC in sufficient time for incorporation in the current review of WCAP-10125. Rather than delay the SER for WCAP-10125, Westinghouse will continue to use the current design methodology and criterion for interactive fuel rod/fuel assembly growth in design to extended burnups until the detailed supporting information for the proposed revised criterion has been submitted to the NRC, reviewed and approved for use in design applications.

<u>References</u>

1) WCAP-9500-A, "Reference Core Report 17X17 Optimized Fuel Assembly," May, 1982.

2) WCAP-10444, "Reference Core Report Vantage 5 Fuel Assembly," December, 1983.